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INTRODUCTION

Journal of Energy special issue: Papers from 11th International Conference of the Croatian Nuclear Society "Nuclear Option in Countries with Small and Medium Electricity Grids"

Welcome to this special issue, which is based on selected papers presented at the 11th International Conference of the Croatian Nuclear Society "Nuclear Option in Countries with Small and Medium Electricity Grids", held in Zadar, Croatia, on June 5th–8th, 2016.

This International Conference was organized by the Croatian Nuclear Society in cooperation with International Atomic Energy Agency (IAEA), Croatian State Office for Nuclear Safety and University of Zagreb, Faculty of Electrical Engineering and Computing. The goal of the Conference was to address the various aspects of the implementation of nuclear energy for electricity production in the countries with small and medium electricity grids and in power system in general. The conference also focuses on the exchange of experience and co-operation in the fields of the plant operation, nuclear fuel cycle, nuclear safety, radioactive waste management, regulatory practice and environment protection.

The conference was organized in eight main topics covered in ten oral sessions and one poster session. In three Conference days authors presented 49 papers orally and 23 papers in poster session. 102 participants came from 16 countries representing equipment manufacturers and utilities, universities and research centres, and international and government institutions. Eight invited lectures were held and 72 papers were accepted by international programme committee.

The importance of international cooperation for the assessment of the nuclear option has been recognized by everybody planning to introduce nuclear power plant to the grid. That is even more important for small and medium countries having limited resources and specific requirements due to limited grid size. The Conference topics reflect some current emphasis, such as country energy needs, new reactor technologies (especially small reactors), operation and safety of the current nuclear power plants, move of the focus in nuclear safety toward severe accidents and accident management strategies, improvement in nuclear safety, reactor physics and radiation shielding calculation tools and ever increasing requirements for minimization of environmental impact.

From 72 papers presented at the Conference, 14 papers were accepted for publication in this number of Journal of Energy after having undergone the additional peer-review process. We would like to thank the authors for their contributions and the reviewers who dedicated their valuable time in selecting and reviewing these papers, both during the Conference and during the preparation of this special issue of Journal of Energy. It was very challenging to collect a balanced overview of the entire Conference. We decided to select 14 papers for this issue what together with already selected 16 papers in previous number covered most important contributions of the Conference. We believe that the papers which were selected for this number represent some of the best research related to nuclear safety, severe accident management, equipment reliability, and probabilistic risk analyses. We hope this special issue will provide a valuable insight into different aspects of nuclear safety modelling and assessment, as well as a pleasant and inspiring reading.

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Enhancements to PWR SAMG since Fukushima

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ABSTRACT

Plant specific Severe Accident Management Guidelines are provided to give guidance to plant staff to respond to a severe accident, to protect containment, minimise fission product releases and prioritise equipment recovery efforts while bringing the plant to a controlled stable condition.

This paper presents and discusses the enhancements that have been made particularly to widely implemented SAMG based on the Westinghouse Owners Group generic model (now part of the PWR Owners Group) since the Fukushima accident. Although U.S. and European responses to the accident may seem at first sight to be different, in fact, the enhancements needed to SAMG in order to address lessons learned were quite consistent. How to deal with loss of d.c. power and / or instrumentation, long term loss of a.c. power (much longer than had previously been considered), multiple unit severe accidents, accidents in the fuel pool, accidents from shutdown initial plant conditions and other issues and conditions that occurred during the event and which we must learn from?

This paper presents results of a number of projects aimed at providing SAMG enhancements to address these issues, some originating in the U.S. and some in Europe. Main enhancements addressing lessons learned from Fukushima are described.

The paper also summarises and describes an ongoing project to develop a fully revised generic SAMG specifically applicable to PWR plants in Europe (and also applicable to other designs, such as VVER) which benefits from all these efforts and will provide a single integrated and updated basis for plants with mitigation systems typical in Europe (often different from the US) to update their SAMG.

Keywords: *PWR SAMG, Development, Fukushima*

1 BACKGROUND AND EARLY DEVELOPMENT

Before the early nineties, guidance for plant staff to recover from an accident was limited to instructions contained within Emergency Operating Procedures (EOP). However, it was recognized that once core damage has occurred, the EOP actions (which place priority on protecting the core) may not be appropriate when priority needs to shift to protecting containment and minimizing fission product releases. It was also recognized that it would be hard to formulate specific procedure-based guidance for recovery from a severe accident (an event in which core damage has occurred), and that some sort of evaluation process would be needed in order to select appropriate recovery actions. This suggested that the role of determining strategies may be better suited to a Technical Support Center (TSC) rather than being solely determined by operations staff.

SAMG were developed to provide structured guidance to plant staff to stabilize the plant and return it to a controlled state following a severe accident involving core damage. Early on, it was recognized that SAMG deal with a situation far outside the plant design basis, and as such, any available means to mitigate the situation would be considered and included in the guidance, even if equipment was used outside it intended use.

In the US, SAMG development was performed as an industry initiative in the early nineties. A Technical Basis Report (TBR) was developed by EPRI[1][1] , and the various utility owners groups developed generic guidelines packages, based on the technical state of the art provided in the TBR. Individual utilities then used the generic guidelines to develop plant specific guidelines.

The original WOG approach to PWR SAMG, which is briefly outlined below, has also been extensively used to develop plant specific SAMG in plants outside the US, including many PWR plants, but also plants of differing design including VVER and CANDU reactors.

2 WOG SAMG OVERVIEW

The original generic WOG SAMG were developed during 1991 to 1994 with the aim of providing a basis for individual PWR plant owners to implement plant specific guidance material. The goals for severe accident management that form the foundation of the WOG SAMG are:

1. Terminate any release of radioactivity to the environment;
2. Prevent the failure of any containment fission product boundary as a result of the further progression of a core damage accident, and
3. Return the plant to a controlled stable condition where containment fission product boundaries would not be threatened in the long term.

In reaching these goals, the release of radioactivity to the environment must be minimized, and the availability of equipment and instrumentation must be maximized.

The WOG SAMG Revision 0 [2] provides severe accident management guidelines based on a structured decision making process, and use of step-wise structured guidelines and flowcharts. This approach minimises training requirements and provides a tool that is easy to use under the potentially high stress conditions of a severe accident. The structured process includes:

- Assess plant conditions by monitoring key plant parameters
- Prioritize response
- Assess equipment availability – and prioritize recovery actions for unavailable equipment
- Identify and assess negative impacts of potential actions
- Determine whether to implement available equipment
- Determine whether implemented actions are effective
- Identify long term concerns for implemented strategies

The applicable accident management actions are independent of the details of the severe accident phenomena and/or the progression of severe accidents. The principal actions contained in the guidelines are:

[1] The original EPRI TBR was a proprietary document. The TBR was subsequently updated following the Fukushima accident (EPRI2012), and is no longer proprietary. It is available at : http://my.epri.com/portal/server.pt?Abstract_id=000000000001025295

1. Fill the steam generators
2. Depressurize the reactor system.
3. Inject water into the reactor system to cool the core
4. Inject water into the containment
5. Depressurise the containment
6. Reduce the containment hydrogen concentration

The generic WOG SAMG is contained in two control room guidelines (implemented by the operations staff in the main control room), a diagnostic flow chart and a severe challenge status tree (the primary diagnostic tools for severe accident management), and fourteen Severe Accident Management Guidelines which provide a structured evaluation for implementing severe accident management strategies.

The diagnostic tools and the guidelines are intended primarily for use by the on-site Technical Support Centre (TSC). TSC organization and staff training in the use of new SAMG are therefore important issues which must be addressed during plant specific implementation..

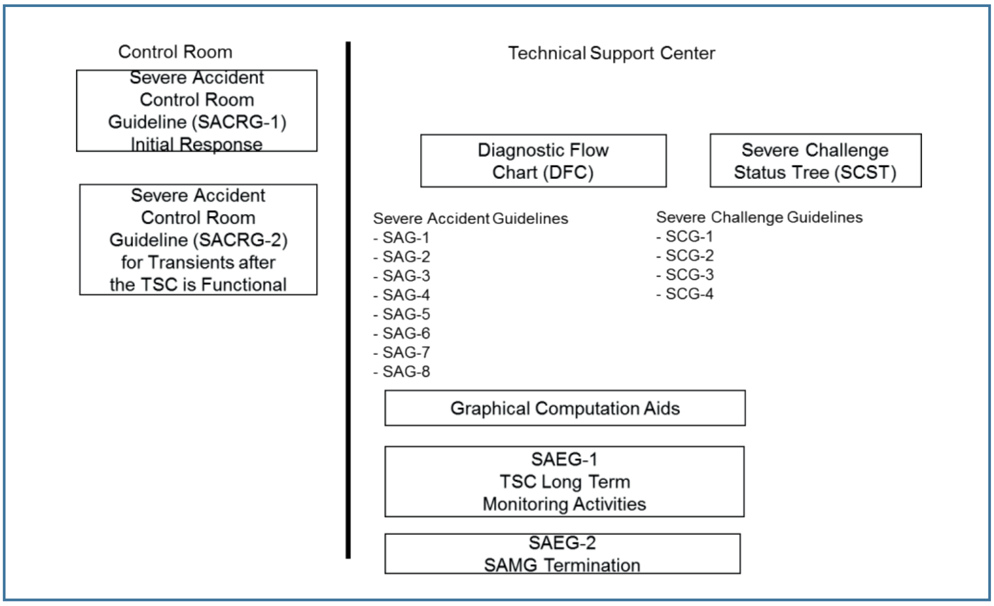

Figure 1: WOG SAMG Rev. 0 - Components

3 REVISIONS AND DEVELOPMENT OF THE WOG SAMG

Revisions to the generic guidance were performed when improvements were made to the understanding of severe accident phenomena, plant modifications were introduced to facilitate mitigation of severe accidents, and lessons were learned from implementation and performance of drills and exercises.

Revision 1 of the WOG SAMG was issued in 2001 [3], and incorporated many detailed changes/modifications to the generic WOG SAMG which had been accumulated from several years of experience with training, validation and exercises of plant specific guidelines.

Revision 2 was issued for implementation in December 2012 [4], following the Fukushima accident. This revision was intended to address short term post-Fukushima modifications to the SAMG. Since the development of consolidated PWROG SAMG was ongoing in parallel (see below), this revision addressed only specific issues raised by the revised TBR [1], mainly in the following areas:

- Spent Fuel Pool
- Auxiliary Building Ventilation
- Containment Venting
- Use of Raw Water (e.g., sea water, brackish water, river water)
- External cooling of the reactor vessel lower head

PWROG consolidated SAMG: Driven mainly by the desire in the US to consolidate the three original Owners Group SAMG approaches into a single approach applicable to all PWRs, the PWR Owners Group has performed a major revision of the generic SAMG which is intended to be applicable for Westinghouse, Combustion Engineering and B&W PWRs, and which can also be adapted (as was the original WOG SAMG) to other designs. This development drew on the strong features of the individual SAMG approaches, on the various revisions described above, and on the lessons learned from the Fukushima accident, to provide a restructured an improved SAMG package. The basic structure and components of the PWROG consolidated SAMG are shown in Figure 2. The PWROG SAMG was finalized in early 2016 after undergoing validation. Utilities are expected to start implementing revised plant-specific SAMG based on the new approach during 2016.

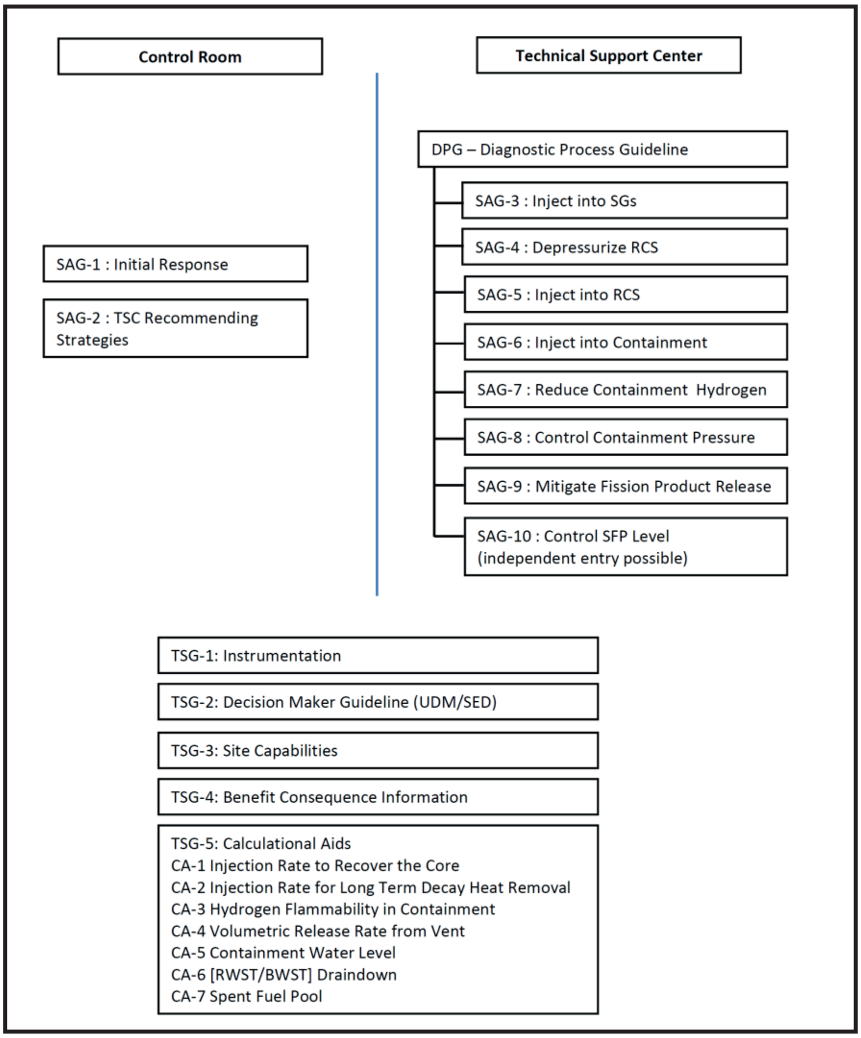

Figure 2: Structure and Components of the PWROG Consolidated SAMG

European work / revisions for international plants: Many plants in Europe and elsewhere (but outside the US) have implemented systems upgrades and backfits to enhance severe accident mitigation. In addition, certain issues (such as severe accidents occurring from shutdown initial plant states) have always received more attention in Europe than in the US. This has led to specific needs for international plants implementing SAMG which were originally developed in the US. To address some of these issues, in the period following the Fukushima accident, a European PWROG project was performed in 2012-2013, which complemented the development efforts ongoing in the US. The project provided international plants with:

- Modifications to SAMG for plants with PAR-based hydrogen control systems
- Modifications to SAMG for plants with severe accident filtered vent system
- Modifications to SAMG for plants without containment hydrogen monitoring system
- A specific procedure for severe accident combined with loss of all D.C./Instrumentation
- A review of the benefits and disadvantages of diagnosing reactor vessel failure in SAMG
- Extension of SAMG to cover shutdown initial plant states
- Review of multiple unit issues for SAMG (consideration of simultaneous severe accidents in multiple units)
- Interfaces with US development programs to ensure consistency.

4 EXAMPLE OF NEW FEATURES: TSGS

TSGs are a new feature of the PWROG SAMG. “TSG” is consistent with the BWROG terminology for additional SAMG tools used by the TSC. They address different areas of severe accident response, and may be used by both control room and Technical Support Center. The generic PWROG SAMG contains five TSGs:

TSG-1: Instrumentation: This TSG provides guidance for evaluating the accuracy and reliability of plant instrumentation, contains instrumentation information (design limits, calibration details, operation details, calculated biases), guidance on cross-checking of indications with other instruments, obtaining local readings. It covers loss of all DC as well as loss of individual instrumentation

TSG-2: Decision Maker Guideline (UDM/SED): This TSG provides the ERO Ultimate Decision Maker with guidance related to SAMG actions and potential conflicts. TSG-2 is discussed in more detail below.

TSG-3: Site Capabilities TSG-3: This TSG contains details of site mitigation equipment, both installed and portable equipment (e.g. air compressors, electrical generators and d.c. power sources). Details are pre-filled by the plant for equipment availability and location, capacity, operating requirements, etc. Site water resources and fuel resources are also be tracked.

TSG-4: Benefit Consequence Information: Benefit consequence evaluation has been simplified in the new SAMG by including the recommended action for most likely plant conditions directly in the SAGs. This TSG provides additional information regarding benefit/consequences of strategies.

TSG-5: Calculational Aids: Calculational aids are grouped together in TSG-5.

5 DECISION MAKER GUIDELINE: TSG-2

The ultimate decision making regarding a SAM strategy recommended by the TSC often lies with the Site Emergency Director (SED). The SED is the only member of the emergency response organization (ERO) with a sufficiently broad understanding of the situation to be able to evaluate the overall potential impacts of a proposed strategy. However, in the past there was no guidance from within the SAMG to assist the SED in this role. This has been addressed within the PWROG consolidated SAMG by the provision of a specific Technical Support Guideline for use by the SED. TSG-2 provides the ultimate decision maker (the SED) with a tool that provides information that may be beyond TSC recommended strategies (use not required), and helps him to evaluate the appropriateness of proposed strategies in the context of:

- personnel safety
- physical plant damage
- site resources
- nuclear safety (single and multi-unit decisions)
- regional impacts
- regaining mitigation capability.

6 ADDRESSING FUKUSHIMA LESSONS LEARNED

The way in which the new developments described above address some of the major lessons learned from the Fukushima accident is summarized below.It should be noted that addressing these issues relies on a fully integrated and self-consistent set of guidelines and procedures, of which SAMG form a part.

Extended Station Blackout:
- Implementation of FLEX (or equivalent) equipment and capabilities;
- Implementing the corresponding FLEX Support Guidelines (FSG) (or associated operating procedures);
- Ensuring consistent links between procedures and guidelines (including EOPs, SAMGs, EDMGs, and FSGs).

Loss of instrumentation and control
- FSG-7 and SACRG-0.0 for loss of d.c.;,
- Instrumentation Technical Support Guideline TSG-1.

Severe accident occurring from shutdown initial condition:
- Extension of generic SAMG to cover shutdown states (PSC-1081)

Loss of Spent Fuel Pool cooling:
- FSG and FLEX equipment to makeup to SFP
- Extension of generic SAMG to cover spent fuel pool accidents

Use of seawater – potential precipitation issues:
- EPRI TBR update and WOG SAMG rev. 2 guidance

Multi-unit severe accident:

- TSGs for Decision Maker
- N+1 FLEX equipment

Site disruption – TSC unavailable/late :

- Restructuring of control room SAGs – some actions systematic by operators
- EDMG for loss of command and control

7 PSC-1413 AND THE WAY AHEAD

As described, there have been many developments since the first SAMG of the mid-nineties, both pre- and post-Fukushima. But what is important to plant owners is to have a single applicable set of generic SAMG that can be used to develop plant specific guidelines and which incorporate all the development TD since the original SAMG were developed.

The development of the new PWROG SAMG in the US has incorporated all these developments, but was also driven by a US desire to "consolidate" the three PWR owners' groups (W, CE and B and W) approaches into a single approach usable by all US PWRs.

In Europe, many plants implemented WOG type SAMG. (The WOG SAMG have been used as the basis for plant specific SAMG in Westinghouse PWRs (eg Beznau, Ringhals, Asco, Vandellos, Almaraz, Tihange and others), at Framatome/AREVA plants (eg Koeberg), at Siemens/AREVA plants (eg Borssele) and in VVER reactors in Central Europe (Bohunice, Mochovce, Temelin, Dukovany). However, the approach to systems upgrades for the mitigation of severe accidents has differed between the U.S. and Europe, with European plants generally opting to backfit mitigation systems such as passive hydrogen control and filtered containment vents. Also, plant features are often different and regulation though generally consistent has produced some differences in expected scope/expectations (NRC/NEI vs IAEA/WENRA) (for a review of the regulatory and industry response to the Fukushima accident in U.S. and in Europe, see [6]). These differences can have a major impact on SAMG, and have meant that developing and maintaining plant specific SAMG based on U.S. generic reference guidelines has become more and more complex for European plants. In addition, the motivation to consolidate different U.S. owners groups' SAMG mainly did not exist in many European plants, which often followed the WOG approach, though of course new technical developments/enhancements need to be included. While the international PWROG project PSC-1081 described above addressed many of these differences, it did not produce a new dedicated set of generic guidelines for plants with additional SA mitigation features.

To address this, PWROG authorised a currently ongoing project specifically for international plants which uses a reference plant design typical in Europe (with large dry containment, PARs, filtered vent, etc). The project will deliver an updated set of generic SAMG applicable to the reference plant, and taking maximum benefit of the enhancements resulting from the previous development projects described earlier in this paper. This project, PSC-1413, is currently underway and is scheduled to complete in the second half of 2016.

This results in two sets of generic PWROG SAMG - one for US reference plant and one for European. The two resulting sets are fully consistent but are tailored to enable utilities in both the US and Europe to adapt more easily their applicable generic set to their plant.

Thus in the US, during 2016/2017 plants are expected to begin implementing plant specific SAMG upgrades based on the consolidated PWROG SAMG. And in Europe, the generic guideline set from PSC-1413 can be used from mid 2016 as a basis for developing or updating plant specific SAMG.

8 CONCLUSIONS

There is wide experience in implementing plant specific SAMG based on the original WOG severe accident management guidelines. While generic SAMG were revised periodically, the Fukushima accident revealed additional areas requiring attention.

Following Fukushima, PWROG programs were launched to address these areas, to integrate PWR SAMG (in US) and to provide specific guidance for International Plants.

Integrating revised SAMG with other plant specific guidance and procedures (including EOP, FSG and EDMG) provides for a comprehensive accident management capability, and special effort has gone into ensuring such integration.

During 2016, new generic SAMG will be available for both U.S. and European reference PWR plant designs which can form the basis of plant specific SAMG upgrades which will take full advantage of the extensive developments since Fukushima.

9 ABBREVIATIONS

B&W	Babcock and Wilcox
BWROG	Boiling Water Reactor Owners Group
CA	Computational Aid
CE(OG)	Combustion Engineering (Owners Group)
DPG	Diagnostic Process Guideline
EDMG	Extensive Damage Mitigation Guideline
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
FSG	FLEX Support Guideline
FLEX	Diverse and Flexible Coping Strategies
IAEA	International Atomic Energy Agency
NRC	Nuclear Regulatory Commission
NEI	Nuclear Energy Institute
PWROG	Pressurized Water Reactor Owners Group
TBR	Technical Basis Report
TSC	Technical Support Center
TSG	Technical Support Guideline
SACRG	Severe Accident Control Room Guideline
SAG	Severe Accident Guideline
SAMG	Severe Accident Management Guideline(s)
SED	Site Emergency Director
SFP	Spent Fuel Pool
UDM	Ultimate Decision Maker
WENRA	Western European Regulators Association
WOG	Westinghouse Owners Group

10 REFERENCES

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11 ACKNOWLEDGEMENTS

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Development of an Evaluation Methodology for Loss of Large Area induced from Extreme Events with malicious origin

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ABSTRACT

Event of loss of large area (LOLA) induced from extreme external event at multi-units nuclear installation has been emerged a new challenges in the realm of nuclear safety and regulation after Fukushima Dai-Ichi accident. The relevant information and experience on evaluation methodology and regulatory requirements are rarely available and negative to share due to the security sensitivity. Most of countries has been prepared their own regulatory requirements and methodologies to evaluate impact of LOLA at nuclear power plant. In Korea, newly amended the Nuclear Safety Acts requires to assess LOLA in terms of EDMG (Extended Damage Mitigation Guideline). Korea Institute of Nuclear Safety (KINS) has performed a pilot research project to develop the methodology and regulatory review guidance on LOLA at multi-units nuclear power plant since 2014. Through this research, we proposed a methodology to identify the strategies for preventive and mitigation of the consequences of LOLA utilizing PSA techniques or its results. The proposed methodology is comprised of 8 steps including policy consideration, threat evaluation, identification of damage path sets, SSCs capacity evaluation and identification of mitigation measures and strategies. The consequence of LOLA due to malevolent aircraft crash may significantly susceptible with analysis assumptions including type of aircraft, amount of residual fuel, and hittable angle and so on, which cannot be shared overtly. This paper introduces a evaluation methodology for LOLA using PSA technique and its results. Also we provide a case study to evaluate hittable access angle using flight simulator for two types of aircrafts and to identify potential path sets leading to core damage by affected SSCs within damaged area.

Keywords: *BDBEEE, LOLA, EDMG*

1 INTRODUCTION

After September 11 event in 2001 and Fukushima nuclear disaster in 2011, the landscape of nuclear safety paradigm has been changed drastically. Before September 11 event, malevolent man-made hazards have rarely taken into consideration for safety design of nuclear installations. Fukushima catastrophic disaster gave us a wakeup call for re-consideration of robustness of current accident management framework against the event of loss of large area induced from beyond design basis extreme external events (BDBEEE). USNRC announced several regulatory requirements and guidance documents regarding the event of loss of large area including 10CFR 50.54(hh)[1], Regulatory Guide 1.214[2] and SRP 19.4[3]. In Korea, the consideration of loss of large area has been limitedly taken into account for newly constructing NPPs as a voluntary basis. In general, it is hardly possible to find available information on methodology and key assumptions for the assessment of LOLA due to “need to know based approach”. Urgent needs exists for developing country specific regulatory requirements, guidance and evaluation methodology by themselves with the consideration of their own geographical and nuclear safety and security environments. Korea Hydro and Nuclear Power Company (KHNP) has prepared an Extended Damage Mitigation Guideline for APR-1400 as a near-term post-Fukushima action plan. However, accident management during the event of loss of large area at multi-unit site requires cross-cutting and interdisciplinary coordination and cooperating among in-house organizations or inter-organizations. The submittal guidance NEI 06-12[4] related to B.5.b Phase 2&3 focused on unit-wise mitigation strategy instead of site level mitigation or response strategy. Phase I mitigating strategy and guideline for LOLA provides emphasis on site level arrangement including cooperative networking outside organizations and agile command and control system. Korea Institute of Nuclear Safety has carried out a pilot in-house research project to develop the methodology and guideline for evaluation of loss of large area since 2014. This paper introduces the summary of the results and outcomes of the aforementioned research project [5].

2 METHODOLOGY

The main purpose of LOLA evaluation is to delineate potential mitigation strategies amd measures through identifying vulnerability and anticipated offsite consequences induced from the chosen hazards or threats scenarios. Figure 1 provides an overall outlook of a methodology for the LOLA assessment.

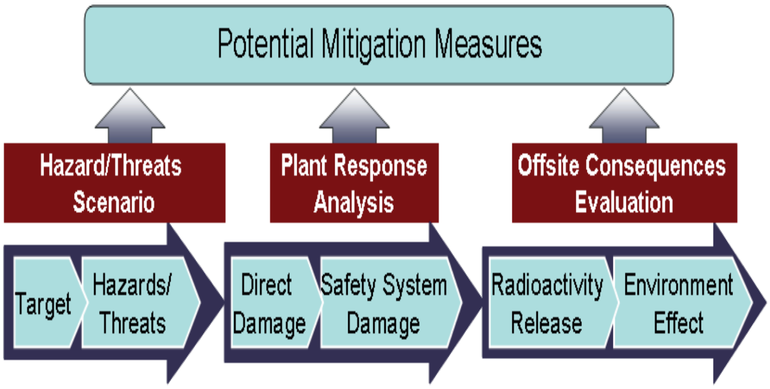

Figure 1: The framework of LOLA assessment

In case of LOLA events induced from fire and explosion with malicious origin, most of countries have dealt the selection of target scenarios and major analysis assumptions as “need to know” basis or safeguard information approach. The detailed information on the aforementioned should be dealt as sensitive information in terms of nuclear security due to the possibility of misuse to identify vulnerability of being targeted facilities for malicious acts such as radiological sabotage.

Figure 2: A methodology for LOLA Assessment

Figure 2 provides an evaluation methodology for the event of LOLA induced from explosion or fire with malicious origin. The approach of need-to-know requires policy consideration step to select the target scenarios and assumptions for the analysis reflecting following aspects:

- Type and size of aircraft being crashed
- Hitting point and angle
- Terminal velocity to crash
- Amount of residual fuel

Target scenarios and analysis assumptions affect significantly to characterize the scope of analysis and potential mitigation strategies and measures.

The characterization of target scenarios and major assumptions based on the policy consideration are followed by specifying damage area print as shown in Figure 2. Damage area footprint provides visualization of damaged area and list of affected rooms and structures, system and components (SSCs). Figure 3 gives an example of visualization of affected area. Magnitude of damage area varies with hitting point and size of fireball generated by fire and explosion. The size of fireball specifies the number of SSCs to be considered for the assessment. Identification of damage area can be made by computational fluid dynamics, fire analysis and empirical correlation of damage functions considering following aspects:

- Fireball overpressure
- Cable fragility
- Fire propagation effect
- Available firefighting assets
- Fire-induced failure of SSCs
- Burning liquid fuel spread in multi-level structures

S. C. Kim, J. S. Park, D. J. Chang, D. H. Kim, S. W. Lee, Y. J Lee, H. W. Kim, Development of an Evaluation Methodology for Loss of Large Area induced from Extreme Events with malicious origin, Journal of Energy, vol. 65 Number 3–4 (2016) Special Issue, p. 12-20

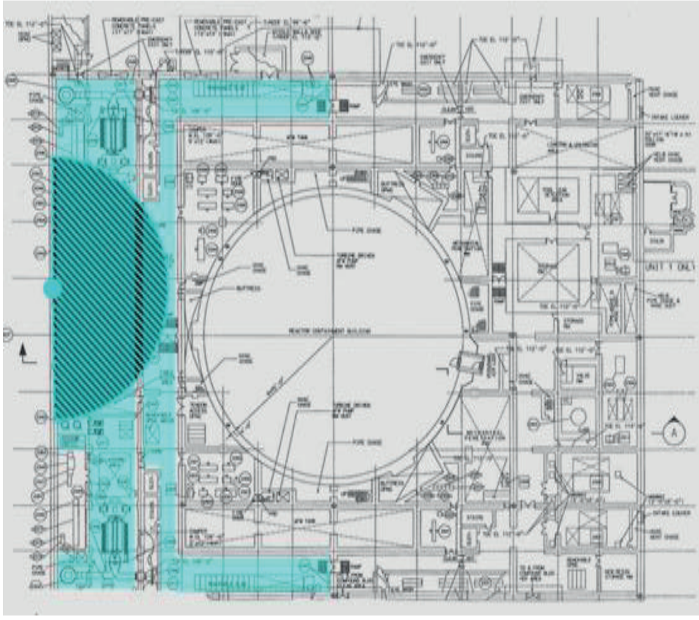

Figure 3: Example of damage footprint

We can utilize various mechanistic models as a function of amount of fuel and duration time of fireball to calculate maximum diameter of fireball [6] [7].

Identified damage area footprint and list of affected SSCs make possible to specify the path sets to core damage using PSA(Probabilistic Safety Assessment) models and existing PSA result. For the conservative approach, a conservative assumption with entire failure of SSCs included in fireball diameter can be made. Through the site walk-down and detailed evaluation of survivability of SSCs, unnecessarily pessimistic sequences of events or SSCs can be screened out from the list of target analysis. Based on finalized path sets to core damage and listed SSCs, consequence analysis should be made by utilization of existing severe accident analysis codes such as MELCOR or MAAP.

However, when we utilize existing PSA and severe accident analysis results, careful attention should be given due to the possible alien mechanism of containment failure, which is screened out in existing PSA framework.

Final outcomes of LOLA evaluation is identification of candidate strategies for EDMG (Extended Damage Mitigation Guides). Through this pilot research, we proposed a draft EDMG guideline for domestic nuclear power plants with emphasis on following aspects at the strategical point of view:

- Firefighting response strategy
- Response strategies for mitigating core damage
- Response strategies for mitigating fuel damage at spent fuel pool

3 CASE STUDY

As a case study, a comparative assessment for identify hittable angle to bring most significant consequence for two types of aircrafts, which are fighter plane and carrier plane, was carried out with empirical evaluation utilizing full-scoped real flight simulator and quantification model. The simulations are aimed to investigate the hittable angles depends on velocity and size of physical dimensions. The case of fighter plane gives us insights the impacts of high terminal velocity with small physical dimension in terms of mass. Detailed information related to simulation and target aircrafts are given to Table 1.

Table 1 Physical characteristics of aircrafts used for the simulation

	Engine	Physical Dimension(m)	Max Velocity	Termin. Velocity
Fighter	Single Jet	15m(L) x 10m(W) x 5m(H)	2500km/h	900-1200km/h
Carrier	Double Turbo-Prop Jet	21m(L) x 26m(W) x 8m(H)	509km/h	300-400km/h

As the target facility for the simulation, an imaginary nuclear power plant with two units of 1000Mwe PWR located in seashore. To identify the access angle giving significant impact to the facility, 9 representative angles as shown in Table 2. For each representative angles, 20 times simulations by real military pilots has been done with real flight simulator. Figure 4 and 5 provide sketches of simulated inside view and outside view during the access.

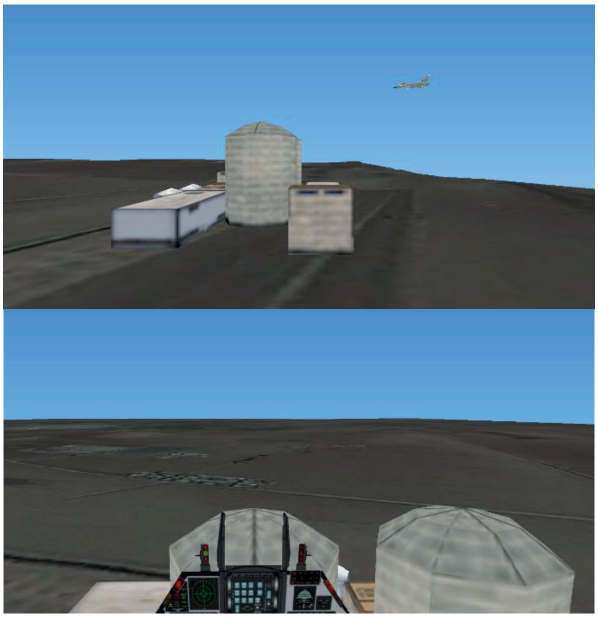

Figure 4: Inside and outside sketch view of simulation for fighter plane

Figure 5: Inside and outside sketch view of simulation for carrier

Table 2 summarizes the empirical results of flight simulator for identifying hittable angles.

Ang / Prob.	0 °	5°	10°	15°	20°	30°	45°	60°	90°
Fighter	≥85 %	≤95%	≥95%	100%	100%	≤95%	≤75%	≤25%	0%
Carrier	≥90%	≤95%	≥95%	100%	100%	≤75%	0%	0%	0%

The samples of simulated flight profiles including access angles of aircrafts provided from Figure 6 to 8.

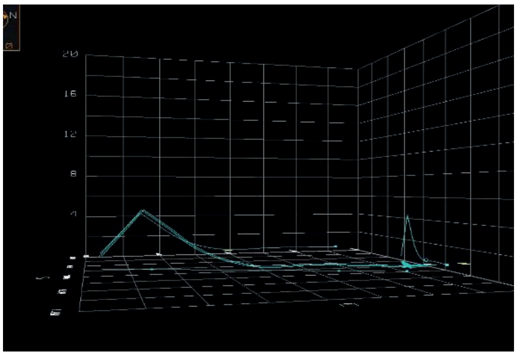

Figure 6: Flight profiles and access angle (0°)

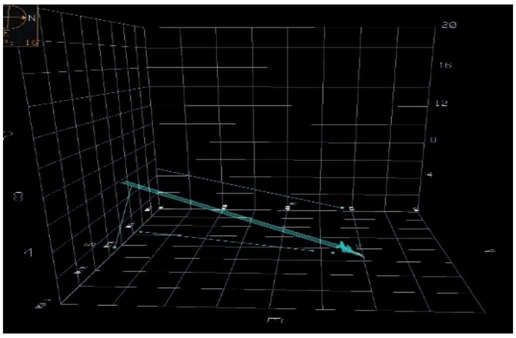

Figure 7: Flight profiles and access angle (30°)

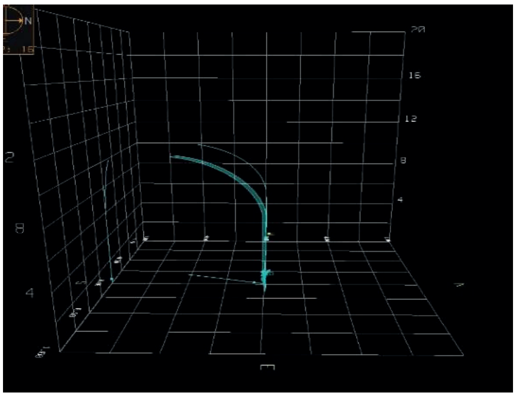

Figure 8: Flight profiles and access angle (90°)

Simulation results demonstrated that fighter plane make possible drastic changing access angle even higher angles than 60°, while the carrier was not possible to change access angle abruptly higher than 60° as shown in Table 2. However, both types of aircraft is not possible to access perpendicular direction to the facility due to the difficulties of posturing the planes. For identification of optimal angle for the evaluation, we quantified impact momentum depending on access angle using following equation 1[8]:

$$F_R = \sqrt{\left(\frac{4k}{\pi}\right)\left(\frac{M+\frac{k\pi\rho_c d^3}{4}}{\frac{4d^3 f_c^{0.456}}{82.6V^2cos^2(k\pi sin\beta)+\rho_c d^3}}\right)} \times \frac{121.7\rho_c d^3\sqrt{f_c}}{2M}\left(\frac{V}{1000d}\right)^{0.2} \times cos\beta \tag{1}$$

Where,

M: Mass of aircraft

K: dimensionless constant for

ρ_c: Density of concrete structure

d: diameter of projectile (Aircraft's head)

V: Access Velocity

Quantification result of impact momentum demonstrate that high impact momentum appears at the range of 0° to 30° of access angle as shown in Figure 9. Considering piloting difficulties and impact momentum, access angle to be taking into account for the assessment would be zero to 30° depending on terminal velocity and mass of target aircraft.

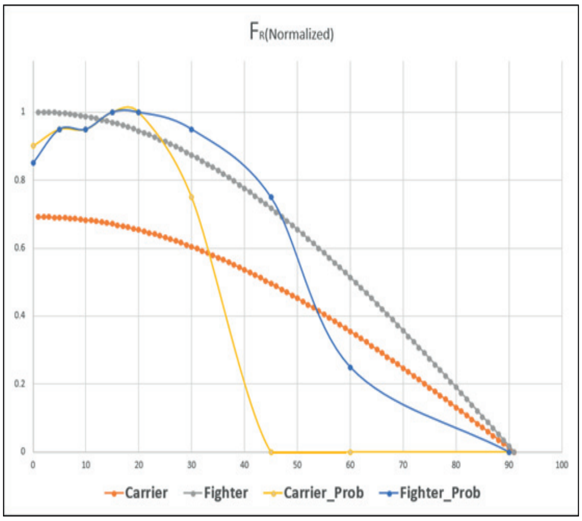

Figure 9: Impact momentum and hitting probability depending on access angle (normalized)

Identification of impact momentum and target access angle call for estimating fireball size characterized by the amount of residual fuel inventory of aircrafts to be assessed. The residual fuel inventory on the verge of crashing are a critical factor to estimate the fireball size which gives significant influence to damaged area footprint. We assumed 82% of residual fuel inventory conservatively, which is same inventory at the beginning of cruising altitude. We estimated fireball size for those of two aircrafts based on the assumed amount of residual fuel and using Abassi's equation 2[9].

$$D = 5.8(m_{feul})^{\frac{1}{3}} \tag{2}$$

We got fireball size of around 20m for fighter and around 50m for carrier from the calculation.

Fighter plane case was screened out from the list of target scenario for identification damaged area footprint due to that carrier case having 2.5 times higher fireball size can encompass fighter plane's impact.

Damaged area footprint and identification of affected SSCs within damaged area can be made by reviewing the general arrangement drawing and walk-down of target facility to be assessed.

Based on the affected SSCs, the path sets leading to core damage can be identified using existing PSA model and Quantification tools. In this study, quantification was made by VIPEX/FTREX[1]® code.

Through the quantification of path sets using the aforementioned code and PSA model, we identified that 1448 sets out of 112,466 sets are exposure as susceptible sets leading to core damage. Figure 10 gives a sample of results of automatic identification of basic event including safety critical functions related to target sets with VIPEX.

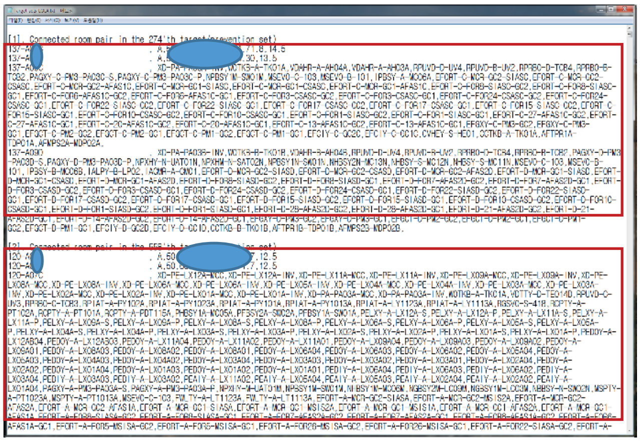

Figure 10: List of basic events affected by SSCs within fireball (example)

The results of case study demonstrated that containment structure secures robustness against target scenarios and SSCs located with damaged area at auxiliary building may be susceptible with breaching primary heat removal and RCS inventory control.

The result of case study demonstrated that the proposed methodology secure enough technical feasibility and applicability for the evaluation of beyond design basis extreme events in regard with preparing strategies of EDMG.

4 CONCLUSION AND REMARKS

After Fukushima Dai-Ichi accident, the awareness on countering the event of loss of large area induced from extreme man-made hazards or extreme beyond design basis external event at nuclear installations. Urgent need exists to develop regulatory guidance for coping with this undesirable situation, which has been out of consideration at existing nuclear safety regulatory framework due to the expectation of rare possibility of occurrence. This paper introduces a methodology and consideration to be given for evaluating the event of loss of large area at nuclear power plant with regard to prepare extended damage mitigation guide (EDMG). As a case study, empirical evaluation for identifying access angle related to analysis assumption with real flight simulator and pilots. Based on identified access angle, type of aircraft and the amount of residual fuel, damage area which is characterized by fireball size and path sets leading to core damage was quantified using PSA tools and results. Analysis results demonstrated that significant impact took a place at the around 20 degree of access angle and 1448 basic events are closely tied with affected SSCs within fireball. This paper proposed a systematic approach for evaluating loss of large area induced from beyond design basis external event with malicious origin. The refining the proposed methodology and its demonstration of the feasibility will be continued through the consecutive research work.

[1] VIPEX :Vital area Identification Package Expert
FTREX: Fault Tree Reliability Evaluation eXpert

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Accident Management under Extreme Events

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ABSTRACT

Most nuclear power plants have extensive sets of Emergency Operating Procedures and Severe Accident Management Guidelines. These offer protection for a large series of events, both inside and outside the licensed design basis of the plant. For Extreme Events, which are characterised by a large destruction on-site and may include loss of command and control, damage to multiple units on-site, loss of communication both on-site and to off-site centres, staff members wounded or killed, such protection may not be enough. Examples of Extreme Events are air plane crash, site flooding, large earthquake plus possible tsunamis, *etc*. This paper describes what additional procedures, guidelines, hardware and organisational issues are needed to protect a site against such events. It is based on lessons learned from large destructive events in the past, such as the 9/11 attacks in the USA in 2001 and the tsunami at the Fukushima-Daiichi plants in 2011.

Keywords: *Extreme Events, Accident Management, Procedures and Guidelines*

1 INTRODUCTION

Nuclear power plants have been designed to withstand a large series of Postulated Initiating Events (PIEs), up to and including Design Basis Accidents (DBAs), plus a number of events outside this range, so-called Beyond Design Basis Accidents (BDBAs), in the new IAEA terminology called Design Extension Conditions (DECs). Examples of BDBAs/DECs are: Anticipated Transient Without Scram (ATWS)[1], station blackout (SBO) and loss of ultimate heat sink (LUHS). The design offers a demonstrated protection if any of the DBAs or selected BDBAs/DECs happen, by the use of various safety and non-safety systems with application of Emergency Operating Procedures (EOPs), and it offers methods to mitigate the consequences if the EOPs are not effective and fuel damage must be expected. These latter methods make use of Severe Accident Management Guidelines (SAMG).

The set of EOPs and SAMG is designed on the basis of scenarios, often with the help of the Probabilistic Safety Analysis (PSA). They are, however, shaped in such a way that it is not necessary for the operator to recognise the underlying scenario. Actions are taken on the basis of observed parameters to bring the plant in a safe stable state or, if this is not possible, to mitigate possible releases, irrespective of the accident scenario.

Typically for SAMG is that the instructions need not be followed verbatim, as the evolution of the accident may be different from anticipated analysis, also due to deviating instrument behaviour, and to give the accident management team the opportunity to respond to events as they occur with the tools they have available, or can be assumed to be restored to service in due time. Newer designs have features that are specifically designed to mitigate core damage scenarios, such as systems to cool a reactor pressure vessel from outside or, alternatively, a core catcher.

The typical lesson from Fukushima is, however, that events can happen that have not been foreseen and are, therefore, outside the events for which the design offers protection through

[1] These are events which would lead to an automatic reactor scram, but where the scram functions fails.

hardware, procedures/guidelines (EOPs, SAMG) or both. Such events may be called ´Extreme Events` (EEs), or ´Site Disruptive Accidents` (SDAs). They may include a large destruction on-site, multiple failures of protective equipment, simultaneous damage to multiple stations on-site, staff members being wounded or killed, loss of normal command and control (i.e., loss of control room and emergency control room / shutdown room, possibly staff being wounded or killed; see further sec. 2.1), loss of communication systems, etc.

External initiators of Extreme Events can be e.g., air plane crashes, large explosions, big fires, earthquakes, flooding, and destruction caused by third parties. Internal initiators can be reactor vessel failure, widespread fire from internal cause, large scale flooding from internal cause. Not extreme but complicated are multiple steam generator failures.

Unfortunately, we must assume that events which are (far) outside the design basis will continue to happen, as unforeseen events will continue to happen and no technology is perfect. Also operational errors have been made and will continue to be made in future. Therefore, efforts should be taken to mitigate such events to the extent practical with appropriate procedures and guidelines and, where needed, with additional equipment. This document gives an overview of such a set, and indicates where improvements should be made to present-day approaches.

2 MAIN ELEMENTS TO MITIGATE EXTREME EVENTS/SITE DISRUPTIVE ACCIDENTS

For Extreme Events/Site Disruptive Accidents it must be assumed that a precise prediction of actual damage states is not possible. Hence, bounding damage states have little value and a flexible response is desirable, including the use of portable equipment. Additional pre-fixed hardware may have limited value, as it can be damaged by the same event that caused damage to existing hardware. As stated, the damage can include loss of command and control, and it is a first priority to restore a commanding structure, [1].

2.1 Restoring Command and Control

One of the potential consequences of an EE/SDA is the loss of command and control. This can happen through loss of the control room and the emergency control/shutdown room, either by physical damage, intrusion by third parties, loss of control room staff, or a combination of these. Physical damage can be caused by e.g., an air plane crash, a large fire and/or explosion, a big earthquake, site flooding, etc. Intrusion by third parties can include violent actions and may result in inability to access parts of the site. Loss of control room staff may be the consequences of such events, but also internal events may cause this, e.g., if main steam lines run close to the control room and rupture[2]. Hence, the very first action must be to restore command and control, so that a functioning Emergency Response Organisation (ERO) can be (re)created and response can be organised in a structured way.

Restoring command and control should be done in an organised way. For example, it may call for surviving staff to assemble at some pre-defined location and restore a command and control structure with the ´best` people who are available, e.g., a senior reactor operator (SRO) as the preliminary leader of the ERO, until a more qualified person is able to take over command, e.g., a person designated and trained for the function of Emergency Director (which could also be an SRO).

If not a sufficient number of such people are available, then an ERO may be established on the site by the help of a neighbouring NPP or another crisis organisation[3]. If the plant-ERO room is not available, the ERO may assemble in an alternate ERO/TSC facility, possibly off-site, as is often provided for an SDA (TSC = Technical Support Centre).

[2] In some plants, steam lines run over the control room but there is no protection of the control room against a steam line rupture.

[3] In some countries, there is a national crisis organisation for such tasks (e.g., in France)

Note that it may happen that the re-established ERO - at least initially - may not contain qualified reactor operators. Hence, the initial actions to stabilise the plant are separate from the ordinary EOPs and SAMG.

2.2 Actions to stabilise the plant by the ERO and subsequent actions

After re-establishing command and control or directly, if command and control have not been lost, the ERO takes a number of actions, which may - or may not - be in the following sequence (priorities may change due to circumstances):

1. to re-establish communication on- and off-site (if it had been lost),
2. to regain control over the site (if it had been lost),
3. to limit site damage (e.g., fire fighting),
4. to take care of wounded people,
5. to check the condition of the key structures, systems and components, i.e., the reactor, the spent fuel pool, the core and spent fuel pool cooling systems, and the containment, and the conditions in areas where local actions must be executed,
6. to initiate actions to stabilise the plant and to fulfil the main safety functions of shutting down, cooling the fuel (core and spent fuel), providing containment and mitigating any releases,
7. to check the necessary support systems (AC power, DC power, water, pneumatic air, diesel fuel), and necessary staff,
8. to continue the initial actions by initiating the appropriate emergency procedures and guidelines, i.e., Abnormal Operating Procedures (AOPs), EOPs, SAMG, Technical Support Guidelines, as well as their long-term support functions, where applicable,
9. to initiate necessary logistics such as food and lodging for personnel, changes of shift, contact of site personnel with their families, and
10. to execute the site Emergency Plan.

These tasks are further detailed as follows:

Ad 1: To re-establish communication on- and off-site.

For the case normal communication on- and off-site is lost, it may be useful to have dedicated battery-powered portable phones on dedicated locations, which can be assumed to remain unaffected by a site disruptive event. These can be used to re-establish the communications on- and off-site and to alert off-site emergency services, as is described in the plant emergency plan. They include police, medical support, fire brigades, etc., Battery power should last long enough that replacement batteries can be supplied.

Ad 2: To regain control over the site (if it had been lost), and Ad 4: To take care of wounded people.

If the accident involves a violent action by third parties, the ERO must make sure that safe access to all vital areas of the plant will be regained. It is assumed that this task is in the hands of the local security personnel. This action is not discussed here in further detail, as it is in the security domain of the plants. In addition, the ERO must take measures to take care of the wounded and arrange that medical assistance is provided for those in need.

Ad 3: To limit site damage (e.g., fire fighting).

The ERO should estimate the damage at the site and initiate measures to limit such damage. Priority may be successively with the auxiliary/fuel building, control building and turbine building. Measures may include containing and extinguishing fires, evacuating personnel in danger, making sure sufficient water is available. If the threat is from flooding or fire, personnel should be protected, and damage to power sources should be limited / mitigated. Hazardous material should be secured in agreement with applicable plant procedures. Note: if there is a conflict between nuclear safety (core/spent fuel cooling) and fighting a fire in the turbine building, then nuclear safety has the priority.

Ad 5: To check the condition of the key structures, systems and components, i.e., the reactor, the spent fuel pool, core and pool cooling systems, and the containment and

Ad 6: To initiate actions to stabilise the plant and to fulfil the main safety functions of shutting down, cooling the fuel (core and spent fuel), providing containment and mitigating any releases.

The ERO should initiate actions to stabilise the plant. These are shutdown of the reactor and starting decay heat removal functions. Actions may be done locally/manually, according to pre-established procedures. For example, for a PWR this includes starting the Turbine Driven Auxiliary FeedWater pump (TDAFW) locally and manually, for a BWR starting the Reactor Core Isolation Coolant system (RCIC) locally and manually. This may include the need for emergency lightning, dosimeters, protective clothing, ladders, and other equipment. In addition, a number of other actions are needed, as will be described in the next section.

In a number of these guidelines support by portable equipment may be needed. This equipment should be stored on a safe part of the site (i.e., where it can be assumed that it is not made inoperable by the event).

As it is not a priori clear that sufficient licensed operators are available to execute these actions, other staff members and, possibly, staff from off-site organisations (e.g., fire brigade staff, who have been trained in the procedures) are authorised to execute the actions. The procedures and guidelines need therefore be written in a style and format which makes the actions executable by such personnel. Note: the use of portable equipment includes transporting it to the plant and hooking it on to the connection points. These actions may require quite some additional staff, which should be considered in the ERO.

The ERO should monitor and mitigate releases and make sure working areas are habitable. Where needed, doses should be estimated. If needed, sprays can be used to scrub fission products. The ERO should make sure spent fuel is and remains submerged or, if submergence is impossible, to keep it under spray cooling.

The actions mentioned here and under Ad 7 are often called ´Extensive Damage Mitigation Guidelines (EDMG)`, which are further described in sec. 2.3.

Ad 7: To check the necessary support systems (AC power, DC power, water, pneumatic air, diesel fuel), and necessary staff.

The ERO should establish the needed resources for the various actions. This may include AC power, DC power, air (pneumatic devices), and fuel for diesels. Load shedding may be one strategy to extend battery life. Diesels may be started manually and, if no cooling is available, run with intervals. One should fill necessary water tanks e.g., by the fire extinguishing system. Apart from the hardware provisions, the ERO should assure that sufficient staff is available during all these actions and, if necessary, for prolonged time, i.e., until off-site support is available or on-site resources have been restored. Note: these actions presumably go together with Ad 6, and may even be of higher priority.

Ad 8: To continue the initial actions by initiating the appropriate emergency procedures and guidelines, i.e., AOPs, EOPs, SAMG, Technical Support Guidelines, as well as their long-term support functions, where applicable.

Once the initial actions to stabilise the plant have been completed – which may have been executed by non-licensed personnel – the normal set of procedures, i.e., AOPs, EOPs, SAMG – should be executed by licensed personnel, with the help of the Technical Support Centre. This depends on the fact whether the initial conditions for these procedures / guidelines have been reached, as may be determined from their logic diagrams. Note that these initial conditions may be different from the initial actions to stabilise the plant, as described under Ad 5.

A number of these procedures / guidelines may call for portable equipment. This should be available and its transport, connection to the plant and its operation should be covered in the procedures / guidelines. This is further described in sec. 2.4.

Ad 9: To initiate necessary logistics such as food and lodging for personnel, changes of shift, contact of site personnel with their families.

As a site disruptive accident is a long-term event and may include major damage also off-site (e.g., caused by a major earthquake), it is necessary to take actions for long term accommodation of needed staff on-site. In addition, these people should have the possibility to communicate with their families, as there may be severe consequences also for their relatives. Staff not needed for the accident mitigation should be sent away, so as to minimise the burden on the ERO and on these people themselves.

Ad 10: To Execute the Site Emergency Plan.

The ERO should initiate the site Emergency Plan, which includes radiological assessments. Actions may include evacuation of relevant rooms, including the main control room and the TSC room, and then should make alternate places available from where the plant staff can execute its duties. The Emergency Plan includes also initial protective actions for people off-site, such as sheltering, distribution of iodine pills or evacuation, until the off-site emergency organisation can take over. If releases are high, this action may have a high priority.

The entire tasks to be performed are displayed in Figure 1.

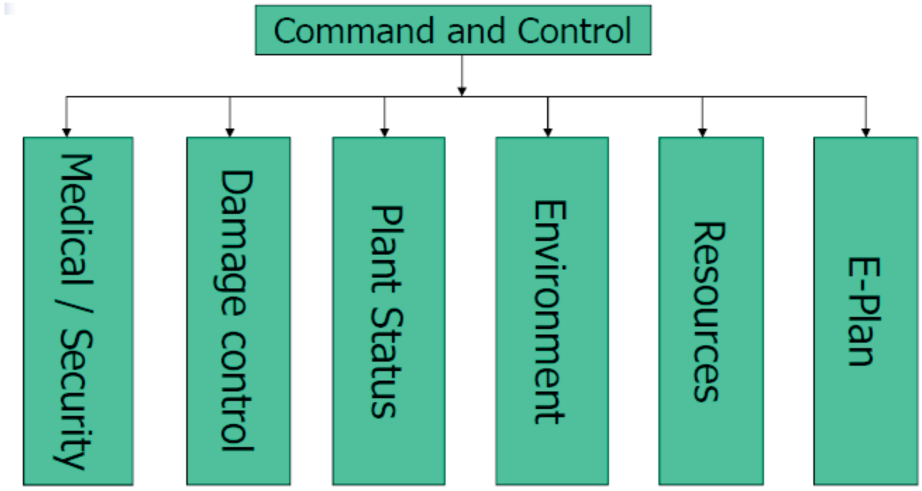

Figure 1: Overview of Tasks in a Site Disruptive Accident (derived from [2])

2.3 Extensive Damage Mitigation Guidelines (EDMG)

The actions mentioned under items 6 and 7 of sec. 2.2 include mainly manual/local actions using, where needed, mobile/portable equipment. They intend to preserve major safety functions and do so by the associated mitigation strategies, see Table 1 (largely derived from [1]).

Whatever system is available to provide cooling water is used. This will also be the fire water system, if still available, or use of a portable pump, stored elsewhere on the site.

Water may come from any source, including the sea. [1] describes various portable/mobile hardware to fulfil these functions. It is usually stored away from the plant, so that there is ample chance it will survive the damaging event. Sec. 2.4 describes such hardware in more detail.

The EDMG which fulfil these functions are typically 10 – 30 guidelines for a plant. In the US, all guidelines are plant specific, as there are no generic guidelines, such as for EOPs and SAMG.

Table 1 Basic Functions and Mitigation Strategies in a Site Disruptive Accident (largely from [1])

Safety Functions:

BWR Safety Functions	PWR Safety Functions
Reactor Pressure Vessel level control	Reactor Coolant System (RCS) inventory control
Reactor Coolant System (RCS) heat removal	RCS heat removal
Containment isolation	Containment isolation
Containment integrity	Containment integrity

Release mitigation	Release mitigation
Spent Fuel Pool (SFP)	Spent Fuel Pool (SFP)

Mitigation Strategies:

BWR Mitigation Strategies	PWR Mitigation Strategies
Manual operation of Reactor Core Isolation Coolant (RCIC) system or Isolation Condenser (IC)	Makeup to Reactor Water Storage Tank (RWST)
DC power supplies to allow depressurisation of RPV & injection with portable pump	Manually depressurize Steam Generators (SGs) to reduce inventory loss
Utilise feed water and condensate	Manual operation of turbine (or diesel-) driven Auxiliary Feed water (AFW) pump
Make up to hot well	Manually depressurise SGs and use portable pump to feed
Make up to Condensate Storage Tank (CST)	Make up to CST or alternate feed source
Procedure to isolate the Reactor Water Clean-Up (RWCU)	
Manually open containment vent lines	
Inject water into the drywell	
Portable sprays	Portable sprays
Internal make-up of SFP	Internal make-up of SFP
External make-up and spray of SFP	External make-up of SFP

2.4 Portable/Mobile Equipment and its use to strengthen Accident Management

Many countries now have decided to have portable equipment available for added capability. As an example in the US, this is called FLEX, which stands for ´flexible and diverse response`. Canada uses a similar term: Emergency Management Equipment (EME). The US industry has set up an approach, documented in NEI 12-06 [3]. It basically consists of three steps:

- Make sure best use is made of existing in-plant equipment, where needed strengthened.
- Have portable equipment on-site, plus ways to transport and connect it to the plant.
- Have portable equipment available off-site, plus a transport means - usually by air - to the stricken site and ways to connect it to the plant; this includes off-site organisational matters.

Other countries have similar or equivalent approaches. But in some countries preference is given to installed and specially protected - often bunkered - hardware such as a diesel generator and/or deep well pumps, despite the risk that this hardware may also be damaged by the event. Portable equipment is then available temporarily, until the new equipment has been installed. In principle, such equipment is available in redundancy and stored separately, so that a single event cannot knock out both stocks of equipment.

The approach assumes that command and control is normally available or has been restored and licensed staff is available to execute emergency procedures / guidelines. Therefore, this portable equipment is meant to add on EOPs with a focus on mitigating Extended Loss of AC Power (ELAP) and Loss of Ultimate Heat Sink (LUHS). Its main function is, hence, preventive, and for selected accidents (ELAP and LUHS). But the portable equipment may also be used for SAMG. Note that it (usually) is not designed to support SAMG. It can of course also contribute to the EDMG.

Figure 2 is an example (from ´FLEX`, [3]) of an overview of the total assembly of procedures and guidelines for plant control. They correspond to the columns ´Plant Status` and ´Resources` in Figure 1. Note: ´broken lines` means that a formal connection between the associated blocks has not yet been established.

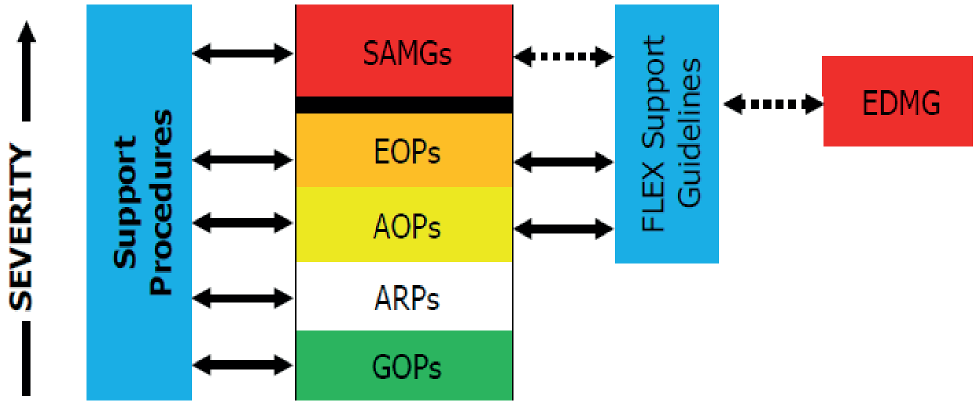

Figure 2: Role and Place of FLEX and EDMG in Assembly of Procedures / Guidelines (from [3]) - GOPs are General Operating Procedures, ARPs are Alarm Response Procedures

Use of portable/ bunkered equipment needs additional procedures/guidelines, such as in the USA the FLEX Support Guidelines (FSG). FLEX equipment or its equivalent usually includes mobile diesel generators, portable pumps, hoses, portable batteries, portable lightning, compressed air bottles, back-up water sources, organised access to alternate water sources (e.g., fire trucks), etc. The equipment should be stored in a place sufficiently protected against external events. Factors to be considered are the following:[4]

- Use of installed equipment as long as is possible, such as the presence of a load shedding program, which sheds loads from all non-essential users.
- Needed portable lighting and communications systems necessary for ingress and egress to plant areas required for deployment of portable equipment.
- Access to areas where local actions must be performed, also under the extended loss of AC, considering also areas locked in absence of AC.
- Effect of loss of ventilation on safety relevant equipment and on access to relevant rooms.
- Operation of containment isolation valves also under the absence of AC/DC.
- Powering igniters for mitigating hydrogen combustion risks.
- Powering relevant instrumentation by portable batteries.
- A sufficient quantity of portable equipment to serve all units on a site simultaneously.
- Possibility to power up local consumers if the plant internal power distribution (LIPD) does not function (requires sufficient number and length of cables and appropriate cable connections).

Portable equipment should be designed appropriately, taking note of needed pump head, head loss in long lines (hoses), back pressure of pressurised volumes (e.g., steam generator), prevention of clogging of inlet lines by debris, ice (winter), etc., taking into account the effect of the external event on the alternate water sources.

The procedures /guidelines to hook on portable equipment should include consideration of difficulty and time involved in its transport to the connection points.

It should be noted that portable equipment may also be needed if the event occurs during an outage. Hence, access routes to transport such equipment and places to hook it on should remain available in an outage. Special attention should be paid to equipment of contractors - of which there are usually many during an outage, plus their equipment - that it does not block routes needed to transport emergency equipment.

Another way to give the structure of plant procedures and guidelines is presented in Figure 3 (derived from [2]).

[4] Largely taken from NEI 12-06, [3]

2.5 Off-site Support

Support from off-site resources requires an appropriate off-site organisation to meet all requirements from the stricken site. This includes provision of equipment and its spare parts and maintenance, consumables, replacement staff, communication means, transport means (trucks, helicopters), personal provisions (food, lodging, medical needs). It should be noted that it will take time before the off-site support is fully available and functional, possibly a whole day. Provisions on-site should be such that this anticipated time lap is appropriately covered:

- The stored equipment must meet appropriate specifications (not necessarily nuclear safety qualifications), have maintenance and regular tests and inspections. Where equipment is out of service for longer times, alternative equipment should be available.
- Transport to the site should not depend on a single transport means or way.
- Plant modifications should be followed and lead to adaptation of equipment or its means of connection to the plant, where needed.
- The off-site organisation should be regularly tested to maintain a high level of alertness and capability to support the stricken plant or site.
- If a multi-unit site is served, it should contain sufficient means to support all units on the site at the same time.

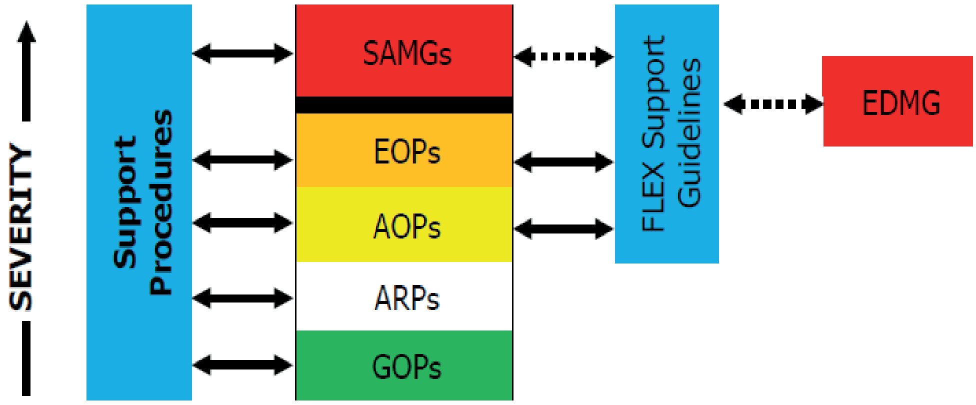

Figure 3: Assembly of Plant Procedures and Guidelines to mitigate Large Scale Events and Accidents; (double arrow: increasing severity; single arrow: information and authority flow).

Teams are available in the US Regional Response Centres to execute these actions. In France there is a Nuclear Rapid Action Force ´FARN` for this purpose.

3 HUMAN ASPECTS OF ACCIDENT MANAGEMENT[5]

Decision making or the preparation of decision making in the Technical Support Centre can be a group decision making process. Groups may act differently from individuals. Group decision making has advantages in that a group may collectively have more knowledge than an individual. As there is deliberation and argumentation, including pros and cons of possible decisions, the final result may be more balanced and better founded. The probability of errors may be smaller as is the chance to overlook important information.

The downside is that group decision making generally, will take more time, and probably is more cautious, due to the long deliberation process. However, groups can also take more risky decisions, because the weight of the decision making is diluted. Due to this dilution, individuals may

[5] This section is drawn from [4]

act differently if placed as an individual for making a decision than as member of a group. The individual contributions may also have different weight, as some people tend to place themselves in the foreground, being ´informal leaders`, whereas others may tend to be more ´followers`. This may also result in a decision process where some are ´winners´ (their arguments have prevailed) and others are ´losers` (their arguments did not prevail).

Consequently, groups can either be more cautious, i.e., trying to avoid risks in decision making, or more courageous, i.e., prepared to take risks in decision making. An example is the need to depressurise the primary system in one of the severe accident guidelines, to avoid high pressure melt ejection. However, the TSC may observe that there is a chance that a lost diesel will be back on line in some hours and so injection into the RCS may be restored, which will prevent vessel failure and all its consequences. Here a cautious TSC will depressurise (the default case), and a courageous TSC will not. The outcome, of course, is only known after the accident evolution is complete.

Unfortunately, there appears to be not much research with respect to the human aspects of decision making in a severe reactor accident, neither application of the extensive work on decision making processes in other disciplines[6]. A recent study in this respect is documented in [6]. Human engineering aspects of nuclear safety are further considered in [7].

4 NEW DEVELOPMENTS IN ACCIDENT MANAGEMENT

Another and important aspect is that the accident management described is of the so-called routine type: ´Routine Accident Management`, i.e., the accident evolves more or less according to scenarios previously analysed. This still can be a large accident, with much damage and the need for large-scale support. But as tasks are well defined and procedures are valid and applicable, accident management can be expected to be effective. An example is a large traffic accident, with big damage and many victims, but the public services (police, fire brigade, medical teams) are trained for such accidents and will know what to do.

However, it can also happen than the accident does not evolve along pre-analysed scenarios or does not result in pre-analysed plant damage conditions, and then many of the procedures and guidelines developed for those scenarios and/or conditions are not effective. In that case, improvisation is needed, with deviations from pre-designed countermeasures. Such accident management is described as `Emergency Accident Management`. It requires a different type of organisation and its success depends also on strong leadership with the decision maker(s). For example, evaluation and decision making are more of cognitive nature than of rule-based nature and, hence, require more knowledge and training. More information is in [5]. It has been argued that the success at Fukushima-Daini was, at least in part, due to this other type of accident management.

Training should, therefore, include this type of training to the extent practical. Good training on ´Routine Accident Management` remains, of course, a basic need.

5 CONCLUSIONS

A Site Disruptive Accident / Extreme Event calls for a large series of actions, which are the subject of an extensive set of procedures and guidelines, and which may require in many cases mobile / portable equipment, stored both on-site and off-site. Still actions may be required beyond the documented set of procedures and guidelines, and these then may require from the ERO an innovative approach. This can only be achieved by a highly skilful ERO which has gone through much in-depth training.

A prime action is to restore command and control, should these have been lost. Then actions must be initiated to stabilise the plant, which include shutting down and removing decay heat as well

[6] See for example http://en.wikipedia.org/wiki/Decision-making

as providing containment. As the event likely has damaged the on-site and off-site power systems, many actions are planned to be locally and manually, where needed with portable batteries, emergency lightning, protective clothing, etc. Apart from the initial actions to stabilise the plant, measures must be taken to provide for needed resources such as cooling water, mobile pumps, mobile power sources, diesel fuel, etc.

The associated assembly of procedures and guidelines are often called ´Extensive Damage Mitigation Guidelines, (EDMG)`. They were first introduced in the US to offer protection against terrorist attacks on nuclear power plants, in the wake of the 9/11 events. In other countries, the approach for site disruptive accidents - if available at all - is not so well documented in the public domain, which prevents proper discussion in a public document as this one is. Together with the execution of the EDMG (or equivalent), the Emergency Response Organisation must take measures to regain control over the site (if lost), limit site damage (fire fighting, etc.), take care of wounded people, provide provisions for lodging and food for plant staff, provide communication for family of staff and initiate the Emergency Plan.

As widespread site damage must be assumed, as well as damage off-site may have occurred, a systematic use of portable /mobile equipment is essential to mitigate the consequences. Counter measures will first try to make best use of surviving plant equipment, but are augmented by equipment stored on-site and off-site. Notably the long term provisions must be provided from off-site sources. The off-site organisation must be well established, its equipment well maintained and regularly inspected and tested. Examples of such methods are the FLEX approach in the US, the use of EME in Canada and the use of the Rapid Response Force (´FARN`) in France.

The total set of procedures and guidelines plus needed equipment and organisational provisions should be regularly inspected and tested.

6 ABBREVIATIONS

AC	Alternate Current (power)
AOP	Abnormal Operating Procedure(s)
ARP	Alarm Response Procedure(s)
ATWS	Anticipated Transient Without Scram
BDBA	Beyond Design Basis Accident
CST	Condensate Storage Tank
DBA	Design Basis Accident
DC	Direct Current (power; usually meaning batteries)
DEC	Design Extension Condition
EDMG	Extensive Damage Mitigation Guideline(s)
EE	Extreme Events
ELAP	Extended Loss of AC Power
EOP	Emergency Operating Procedure
ERO	Emergency Response Organization
FLEX	Flexible and diverse response (includes use of portable equipment)
FSG	FLEX Support Guideline(s)
LUHS	Loss of Ultimate Heat Sink
SAMG	Severe Accident Management Guidance / Guideline(s)
SBO	Station Black Out
SDA	Site Disruptive Accident
SRO	Senior Reactor Operator
TSC	Technical Support Centre

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Equipment Reliability Process in Krško NPP

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ABSTRACT

To ensure long-term safe and reliable plant operation, equipment operability and availability must also be ensured by setting a group of processes to be established within the nuclear power plant. Equipment reliability process represents the integration and coordination of important equipment reliability activities into one process, which enables equipment performance and condition monitoring, preventive maintenance activities development, implementation and optimization, continuous improvement of the processes and long term planning.

The initiative for introducing systematic approach for equipment reliability assuring came from US nuclear industry guided by INPO (Institute of Nuclear Power Operations) and by participation of several US nuclear utilities. As a result of the initiative, first edition of INPO document AP-913, 'Equipment Reliability Process Description' was issued and it became a basic document for implementation of equipment reliability process for the whole nuclear industry.

The scope of equipment reliability process in Krško NPP consists of following programs: equipment criticality classification, preventive maintenance program, corrective action program, system health reports and long-term investment plan. By implementation, supervision and continuous improvement of those programs, guided by more than thirty years of operating experience, Krško NPP will continue to be on a track of safe and reliable operation until the end of prolonged life time.

Keywords: *reliability, safety, criticality, optimization, improvement*

1 INTRODUCTION

Most events on Nuclear Power Plants are caused by errors on plant equipment. Errors can be mechanical, electrical or instrumentation/control caused and they often cause failure of equipment. Dependent on equipment importance or criticality for plant operation and safety, failure of certain equipment may have different consequences for the plant. For example, it can bring to plant shutdown, reduction of power, unplanned entrance into Technical Specification Limiting Condition of Operation (TS LCO), loss of safety functions, etc.

Preventive maintenance program (PM) has important role in ensuring good condition of plant equipment. It is important to perform proper preventive activities within proper time intervals, especially on the most significant plant equipment. Due to the equipment criticality classification and other available criteria, PM Programs are continuously optimized to assure that human and financial recourses for preventive maintenance are properly used to have maximum equipment availability and minimum equipment failures.

Performance monitoring of systems, structures and components (SSC) condition is essential to detect early indicators of SSC degradation and to act in a timely manner to prevent failures. System Health Reports reflect SSC condition by grading the systems due to their performance, recognise insufficiencies and propose measures for improvement (maintenance activities, equipment replacement, design modifications, etc.).

Long term planning of investments on the plant is needed to address all necessary equipment replacements or design modifications and to ensure necessary funds for the investments. Proper prioritization of investments gives the proposed time intervals in which the investments are going to be implemented. Prioritization in made considering various different factors, like current condition of the SSC, unavailability of spare parts, regulatory requirements, radiation dose reduction, influence on plant operation or maintenance, industrial safety, etc. Proper planning and prioritization assures that all plant equipment will be operable/available and will positively affect long term safe and reliable plant operation.

By integration of all important activities/processes which contribute to reliable SSC operation into one process, we have the ability to systematically carry out, control and continuously improve the Equipment Reliability Process.

2 EQUIPMENT RELIABILITY PROCESS SCOPE

Equipment Reliability Process, due to INPO AP-913 [1] consists of five basic elements:

- Scoping and Identification of Critical Components
- Performance Monitoring
- PM Implementation
- Corrective Action
- Continuing Equipment Reliability Improvement
- Life Cycle Management

Graphical illustration of the process scope is shown in Figure 1.

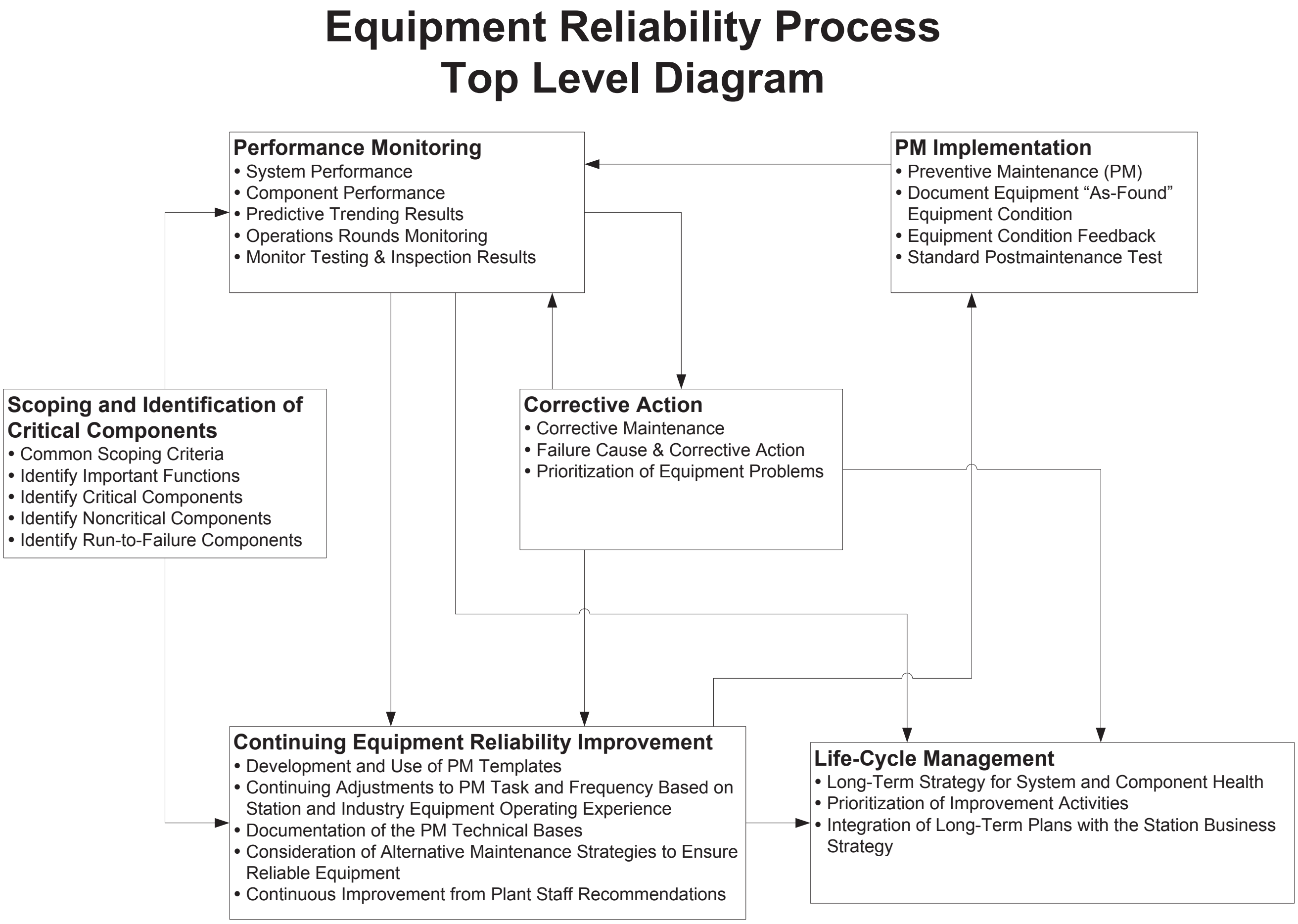

Figure 1: Equipment Reliability Process Top Level Diagram [1]

In Krško NPP we follow the recommendations from the INPO AP-913. Equipment Reliability Process that we implement consists of following elements:

- Equipment Criticality (ECR)
- Preventive Maintenance Program (PM)
- Corrective Action Program (CAP)
- System Health Report (SHR)
- Long-Term Investment Plan

2.1 Equipment Criticality (ECR)

Equipment Criticality is classification of plant equipment into categories based on equipment significance for plant reliable operation and consequences that the failure of equipment is going to have on plant operation. The goal of equipment criticality classification process is to put equipment into different criticality categories and to enable optimization of PM activities performed on equipment based on their criticality. That means, if failure of certain equipment has no major direct influence on plant operation, resources for PM activities on that equipment should be reduced or even abolished. On the other hand, if failure of certain equipment has major direct influence on plant operation, there should be more PM activities on that equipment, they should be executed in shorter time intervals, therefore more resources should be used to maintain that kind of plant equipment. To conclude, equipment criticality enables the plant to optimize the resources for PM on equipment in order to assure reliable plant operation.

There are different methodologies and approaches for criticality classification. The methodology that we use in Krško NPP is to answer the questions considering consequences of equipment failure to plant operation and dependent on the first question answered YES, criticality is determined. We divided equipment into for criticality categories:

1. CC1 – Critical equipment category 1
2. CC2 – Critical equipment category 2
3. NC – Non - Critical equipment
4. RTF – Run to fail

Besides the equipment category, we also mark subcategory that in fact is indicating question that is answered YES. For example, if criticality is CC1-4, it means that equipment is critical category 1 and 4th question that defines this category is answered YES.

Single point vulnerability (SPV) equipment is subcategory of Critical equipment category 1 and consists of CC1-1, CC1-2 and CC1-3 subcategories, which means that one of the three first questions is answered YES. Definition of SPV that we have in Krško NPP is that the failure of equipment that is classified as SPV will cause one of the three consequences:

1. Automatic reactor or turbine trip (shutdown)
2. Power reduction for more than 5%
3. Demand for manual reactor or turbine trip (shutdown)

There are total of 18 questions that define equipment criticality. First five questions define Critical equipment category 1, next five questions define Critical equipment category 2 and the last eight questions define Non - Critical equipment. If all questions are answered NO, equipment is classified as run-to-fail (RTF). RTF means that failure of equipment will not have consequences on plant operation and it is more economical to replace or repair equipment when it fails then to carry out PM activities on equipment. In Krško NPP we are careful with abandoning PM activities based on criticality classification and we also consider other factors, like our own experience on equipment (PM results, test results, equipment history back log), industry experience and other factors available.

Methodology for criticality classification determination is shown in table 1.

CC1 – Critical equipment category 1

1. Is failure of the component going to cause automatic reactor/turbine trip?
2. Is failure of the component going to cause power reduction more than 5%? SPV
3. Is failure of the component going to cause demand for manual reactor trip?
4. Is failure of the component going to cause an entry into Technical Specification Limiting Condition of Operation (TS LCO) that demands plant shutdown within 7 days or less?
5. Is failure of the component going to cause loss of risk significant Maintenance rule function?

CC2 – Critical equipment category 2

1. Is failure of the component going to cause loss of less significant Maintenance rule function?
2. Is failure of the component going to cause power reduction less than 5%?
3. Is failure of the component going to cause partial actuation of protection systems?
4. Is failure of the component going to cause loss of function, which is 100% redundant but can cause a transient that increases risk of loss of production?
5. Is failure of the component going to cause an entry into any Technical Specification Limiting Condition of Operation (TS LCO)?

NC - Non - Critical equipment

1. Is the component needed to for handling beyond design basis accidents?
2. Can the failure of the component lead to outage prolongation for more than 8 hours?
3. Is failure of the component going to cause increase of risk in the area of industrial, radiological or environmental safety?
4. Is failure of the component going to cause violation of regulatory requests?
5. Is failure of the component going to cause difficulties in plant supervision or maintenance or is it going to cause failure of other critical or non-critical components?
6. Is the operability of the component needed for maintenance of other critical equipment?
7. Are there long purchasing time intervals for spare parts that could cause difficulties in repairing the equipment in timely manner?
8. Is it more economical to carry out PM activities on the component than to do corrective maintenance?

RTF – Run to Fail

All questions above are answered NO

Table 1: Methodology for criticality classification determination [2]

Criticality has been classified for approximately 100.000 components. The results of classification are shown in table 2.

Criticality category	Percentage of classified components
CC1 – Critical equipment category 1	**12%**
CC2 - Critical equipment category 2	**15%**
NC – Non - Critical equipment	**36%**
RTF – Run to fail	**35%**
SPV – Single Point Vulnerability	**2%**

Table 2: Results of criticality classification

Results of criticality classification can vary from plant to plant, dependent on methodology that was used and also on the level of details in equipment database.

For equipment criticality determination ECR software application was used. It is an application developed by the plant within the existing business software. The results on criticality classification are automatically transferred to other plant applications, like MECL (Master Equipment Component List), CAP (Corrective Action Program) and Work order application. That way the information about component criticality are visible to all users and can help with prioritizing work activities as well as proper preparation for them.

2.2 Preventive Maintenance Program (PM)

Preventive Maintenance (PM) includes predictive, periodic and planned maintenance activities in order to keep the equipment under its design basis, ensure that equipment fulfil all of its expected functions and to prolong the equipment's life time [3].

Predictive maintenance is continuous or periodic surveillance of components to predict when the component is going to fail and to be able to prevent the failure. Results of predictive maintenance activities show the current condition of equipment and the ability to perform its functions. They are also used for equipment performance monitoring and trending and to enable for needed maintenance activities to be executed before the failure of equipment actually occurs. Examples of predictive maintenance activities: vibration monitoring, infrared thermography, lubricating oil analysis, bearing temperature analysis, isolation resistance, etc.

Periodic maintenance is performed within certain time intervals or based on time the component has been in operation (work hours). Based on available factors, time intervals can be changed, if needed.

Planned maintenance is performed based on results from predictive or periodic maintenance, manufacturer's recommendations or operating experience. It if performed before the project functions of the component are jeopardized. Planned maintenance can be performed under plant shutdown or under operation, if there is a redundant or spare component that can cover the function fulfilment while the component is out of service.

PM program is continuously evaluated and optimized. One of the criteria for PM optimization is equipment criticality classification. Based on equipment criticality, some PM activities may be introduced, cancelled, scope of activity can be changed, time periods for performing PM activities can be reduced or prolonged. Criticality is not the only criteria for PM changes. Other criteria considerable are:

1. Industry experience shows there are more efficient methods and strategies for maintenance.
2. There are new, better predictive technologies available
3. Equipment degradation is occurring in a shorter or longer time intervals than it was expected

4. Equipment has been replaced with new one
5. Information from Maintenance department are indicating there is a need for change
6. Equipment failure occurs between PM time intervals

Changes within the PM program have to be made in controlled manner. That means there is a process which defines responsibilities and jurisdictions within the process. Changes are usually proposed by component engineers within the maintenance department. The approval process goes through department superintendent, system engineer and the final approval is made by maintenance manager. The graphical illustration of the PM review approval workflow is shown in figure 2.

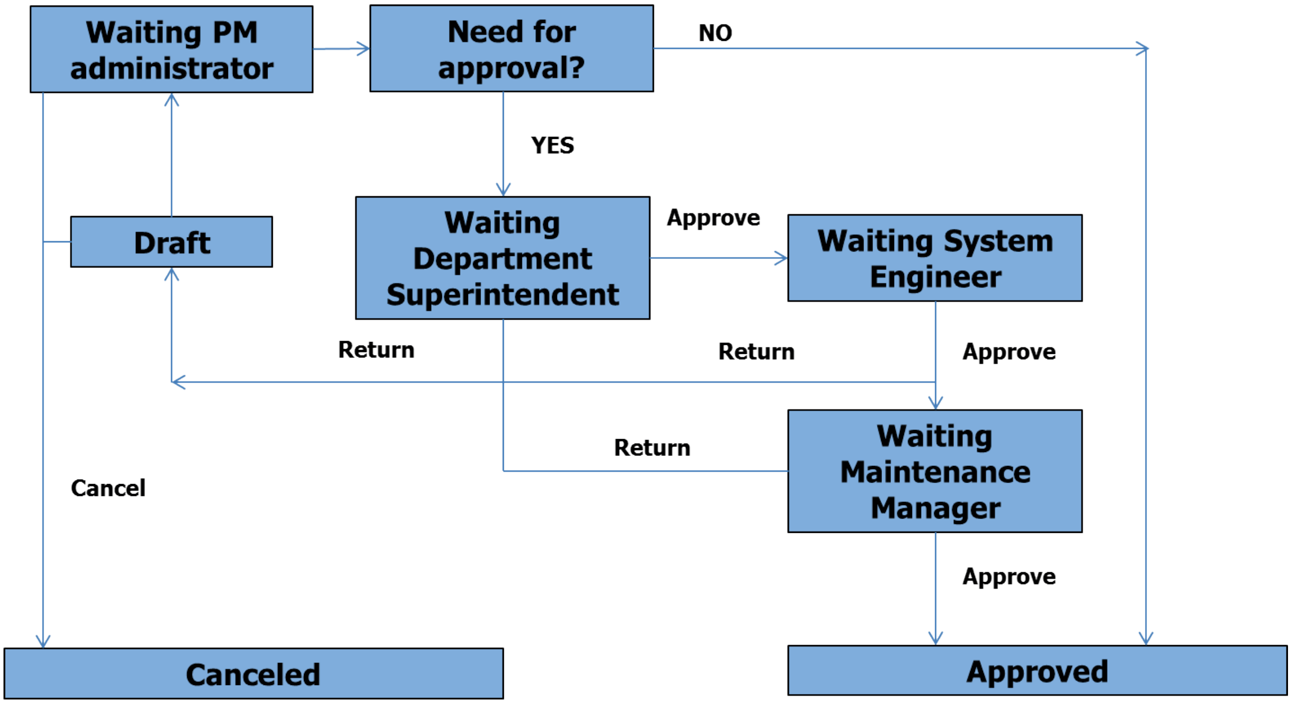

Figure 2: PM review approval workflow

The goal of the PM optimization is to perform proper PM activities on the right equipment in optimum time intervals and to allocate human and financial recourses to achieve that.

2.3 Corrective Action Program (CAP)

The purpose of the Corrective Action Program (CAP) is to enable documenting all equipment deviations, degradations or failures, nonconformance issues or suggestions for improvement [4]. It is also used to keep track on solving equipment issues, to perform analysis for more complex issues and to give action plans for resolving them. Action plans should also prevent deviations from repeating and propose actions for improvement of equipment condition or plant processes.

The CAP process begins by issuing a Corrective Action Request (CR). Any employee on the plant can issue it. All CRs are approved by author's department supervisor or shift supervisor. CRs are processed by CAP screening team, which defines the way to solve CR, given time for solving and assign a department that is going to be responsible for solving the CR. Possible ways to solve a CR are:

1. Direct action to solve a problem - assigned when the CR can be solved directly through work order or some other direct action
2. Analyses – problem is more complex or deeper investigation is needed to propose proper actions

3. Non-conformance analyses – to prescribe actions to resolve issues when component, part or material is not meeting all original specifications (original supplier does not exist on the market any more, original part is no longer manufactured, etc.)

Process is graphically shown in figure 3.

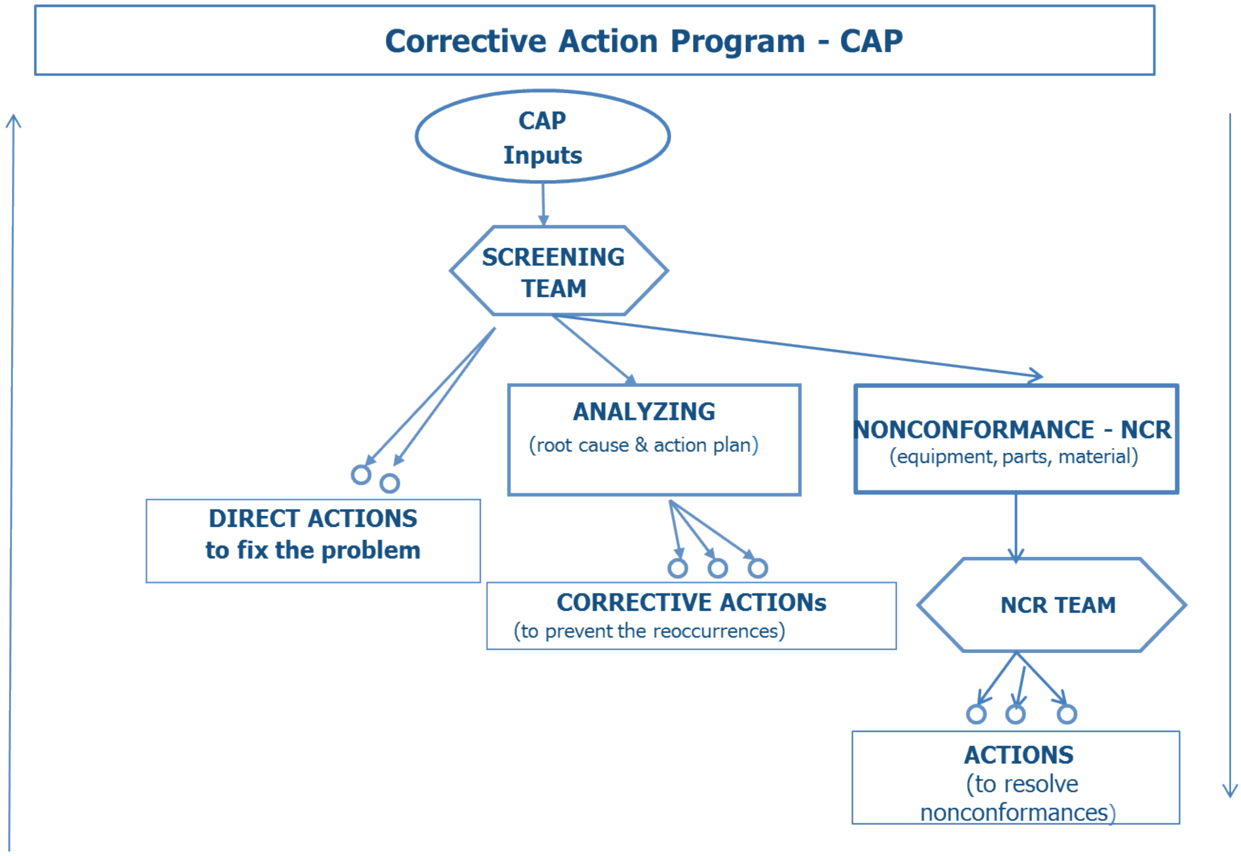

Figure 3: Corrective Action Program process

There are approximately 4000 CRs issued per year. Whole process is conducted through a software application, developed by Krško NPP. Application is connected with other major plant applications, like Equipment Database, Work Order, Equipment Criticality, Documentation Database and Maintenance Rule Application. That enables information to be shared between applications and gives a possibility to collect a lot of information considering certain equipment, its history backlog of deviations, failures, corrective work, replacements and modifications. It is a very broad and useful tool for assuring equipment reliability and collecting information to monitor the performance of equipment.

2.4 System Health Report (SHR)

System Health Report (SHR) is a document that describes condition (health) of the plant systems and proposes actions that has to be performed to improve the health. Intent of the SHR is to assure the information about system condition and necessary improvement activities for plant management and to be basis for composing long-term investment plan. The final goal is to ensure reliable plant operation based on frequent evaluation of plant systems condition.

SHR is written quarterly. Responsibility for writing SHR is on System Engineering department. Information for SHR is collected from available plant sources, like CAP, work order backlog, various reports, information from other departments, etc. Systems are graded based on their current condition. We use for colour grades [5]:

- Green (Satisfactory)
- Yellow (Acceptable)
- Orange (Improvement Needed)
- Red (Potential Danger)

For every grade, except Green, reasons for grade are stated as also proposed activities that have to be performed, to enable transition to higher grade. For every graded system, following aspects are evaluated [5]:

- Reasons for system grade
- Activities to be done on system to put it in a higher grade
- Function failures on the system in the last quarter
- New issues on system equipment in the last quarter
- Improvement plan for planned activities (on-line; outage)
- Prioritization recommendations for un-planned activities
- Aging and obsolescence issues on system equipment
- Major performed activities on system in the last quarter

Based on the system grade, activities are prioritized. The lower the grade is, activities will get greater importance and thus the advantage within the prioritization process. That will also effect due date in the long term investment plan.

SHRs are presented and approved on the Plant Health Committee (PHC) meetings. PHC is an expert panel comprised of representatives/managers from various plant departments: production/operations, maintenance (mechanical, electro, I&C), system engineering, licencing, independent safety evaluation group, and design modification. It's PHC responsibility to monitor and direct the Equipment Reliability Process, to recognize negative trends within the process, propose action for improvement and assure proper implementation of those actions.

2.5 Long-Term Investment Plan

The purpose of the Long-Term Investment Plan is to sort the equipment issues, in fact the solutions to the issues dependent on the importance or prioritization of the issue. Plan contains equipment replacements, technological upgrades and maintenance activities. It is a part of Equipment Reliability Process and it ensures long term safe and reliable plant operation. [6]

Long-Term Investment Plan has a role to establish clear view on activities, which demand larger financial or human resources, involvement of multidisciplinary teams and to assure timely preparation and implementation of those activities as well as optimum use of resources. It is a basis for preparation of other projects, like on-line or outage planning. [6]

Long-Term Investment Plan includes planning of activities for at least five years ahead. To include proposed activity in a Long-Term Investment Plan, analyses within CAP has to be performed and the result of the analyses are actions, which demand design modification process. Those kinds of actions are prioritized within the analyses due to following attributes:

- System grade in SHR
- Influence on operational effectiveness
- Influence on maintenance effectiveness
- CAP classification
- Regulatory requests
- Probabilistic Safety Analysis: influence on core damage frequency (CDF) and unwanted containment releases
- Influence on Industrial Safety
- ALARA

For every attribute certain number of points is assigned, dependent on how the proposed action affects each of the attributes. All points are then added together and the higher the overall point number is the higher priority the action will get. Off course, this prioritization process is not the only exclusive condition for action prioritization within the Long-Term Investment Plan, but it helps in composing the plan, when it is revised. The revision of the plan is carried out at least once a year, first to keep it in a time frame of at least five years ahead and second because new issues are addressed and new prioritization of the activities within the plan can occur.

3 CONCLUSION

Krško NPP implements Equipment Reliability Process in accordance with industry standards. The process is comprised of several elements: Equipment Criticality (ECR), Preventive Maintenance Program (PM), Corrective Action Program (CAP), System Health Report (SHR), Long-Term Investment Plan. Every element is a process for itself ant each contribute to assurance of reliable equipment operation. To be able to effectively monitor and guide the Equipment Reliability Process, coordination and integration of those individual processes is necessary.

It is important to exchange information concerning equipment reliability with other industry representatives to gather best industry practices. For this purpose, working groups for equipment reliability are established (Equipment Reliability Working Group/International Equipment Reliability Working Group). Krško NPP also participates in the activities of International Equipment Reliability Working Group.

Only with systematic approach and continuous improvements we can ensure long term safe and reliable operation of Nuclear Power Plants.

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Lessons Learned from Missing Flooding Barriers Operating Experience

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ABSTRACT

Flooding hazard is highly significant for nuclear power plant safety because of its potential for common cause impact on safety related systems, and because operating experience reviews regularly identify flooding as a cause of concern. Source of the flooding could be external (location) or internal (plant design). The amount of flooding water could vary but even small amount might suffice to affect redundant trains of safety related systems for power supply and cooling. The protection from the flooding is related to the design-basis flood level (DBFL) and it consists of three elements: structural, organizational and accessibility. Determination of the DBFL is critical, as Fukushima Daiichi accident terribly proved. However, as the topic of flooding is very broad, the scope of this paper is focused only on the issues related to the missing flood barriers.

Structural measures are physically preventing flooding water to reach or damage safety related system, and they could be permanent or temporary. For temporary measures it is important to have necessary material, equipment and organizational capacity for the timely implementation. Maintenance is important for permanent protection and periodical review is important for assuring readiness and feasibility of temporary flooding protection. Final flooding protection element is assured accessibility to safety related systems during the flooding.

Appropriate flooding protection is based on the right implementation of design requirements, proper maintenance and periodic reviews. Operating experience is constantly proving how numerous water sources and systems interactions make flooding protection challenging. This paper is presenting recent related operating experience feedback involving equipment, procedures and analysis. Most frequent deficiencies are: inadequate, degraded or missing seals that would allow floodwaters into safety related spaces. Procedures are inadequate typically because they underestimate necessary time or they do not provide sufficient instructions. Most of the events are related to deficiencies discovered during walk-down, review, maintenance and sometimes to incidents. Perhaps these lessons learned from recent events could help filling the missing gap to have most complete flooding protection.

This paper presents results from the most recent activity related to the operational experience feedback for the nuclear power plant safety in the EC JRC Clearinghouse.

Keywords: *Flooding protection, missing flooding barriers, operating experience, lessons learned*

1 INTRODUCTION

Flooding hazard is substantial for nuclear power plant safety because it is usually happening and it has high potential for common cause failure of safety related systems. Source of the flooding could be external (e.g., river, lake, sea or precipitation) or internal (e.g., maintenance and failure of the service, cooling or fire water systems). The volume of flooding water may vary but even a limited amount might suffice to affect redundant trains of safety related systems for power supply or cooling. The protection from the flooding is related to the design-basis flood level and it consists of three elements: structural, organizational measures and accessibility (e.g. [1.] lists several regulatory and guidance documents). Determination of the design-basis water level is critical, as Fukushima Daiichi accident terribly proves (Figure 1). However the scope of this paper is limited to the issues related to the missing flood barriers.

Structural measures physically prevent flooding water to reach or damage safety related systems, and they can be permanent (i.e., water tight doors and barriers, sealed penetrations, drains and pumps) or temporary (i.e., pumps and barriers). For temporary measures it is important to have necessary material, equipment and organizational capacity for the timely implementation. Maintenance is important for permanent protection and periodical review is important for assuring readiness and feasibility of temporary flooding protection. Final flooding protection element is assured accessibility to safety related systems during the design base flooding level (DBFL).

Appropriate flooding protection is built on the right implementation of design requirements and proper maintenance. Nevertheless, operating experience is constantly proving how numerous water sources and systems interactions make flooding protection challenging. That is the reason why this paper is centered on most recent related operating experience feedback (OEF). Perhaps the specific lessons learned could help to fill the missing gap to have most complete flooding protection. The following section describes considered sources with short description for the most relevant events. Final section describes results from events analysis in a form of list of recommended actions with related purpose and example of potentially avoided consequences.

Figure 1 Example of Flood in Electric Equipment Room of Unit 6, pictured on 03.17.2011.
(Source: TEPCO photo.tepco.co.jp/en/date/2011/201108-e/110810-01e.html)

2 SELECTED RELEVANT OPERATING EXPERIENCE

Operating experience is indispensable in finding the problems which went unnoticed during design, construction or maintenance. These events can also serve as motivation for additional checking and inspections. Hence this presentation contains selected most recent illustrative operating events and findings from related walk-downs and inspections. This experience is related to the external and internal flood protection failed to prevent or mitigate the effects because of deficiencies with analyses, procedures and equipment.

Two most comprehensive sources are consulted for operating experience: IAEA OECD International Reporting System (IRS) and US Nuclear Regulatory Commission (NRC). The IRS is closed event database system where all countries voluntary report important events. NRC provides open access to all US events. For this paper main source was NRC IN (information notice) 2015-01 ([4.]a) with recent relevant flooding issues. Selected list of older relevant INs ([4.]b-[4.]i.) illustrates different issue, e.g.: water leakages through conduits into buildings; unsealed concrete floor cracks and equipment hatch floor plugs; backflow through equipment and floor drain system.

Flooding protection related events from the rest of the world are described in several topical studies (TS) and could be also studied from the event reports in the IAEA OECD International Reporting System (IRS). Three relevant reports are consulted for this presentation: IRS Topical Study (TS) OEF External Flooding, [3.]a 2010; Joint Research Centre (JRC) TS on External events, [3.]b 2012; and Nuclear Energy Agency (NEA) Working Group on Operating Experience (WGOE) report on Fukushima Daiichi NPP Precursor Events, [3.]c 2014. Since all these reports are dealing with events mostly relevant from the external events perspective simple IRS search (i.e., free text "flood" in full report from 2010) was performed in order to find more relevant recent events. From total of twelve events, the four most relevant were selected and briefly presented here.

For this presentation priority was given to the most recent events because Fukushima Daiichi accident has motivated new consideration of the flooding protection issue in the last five years. All but two US events (i.e., E07 and E08) are taken from the NRC IN 2015-01. Four non-US events are described based on the respected IRS report. All these events are related to reduced ability to mitigate flooding. These are examples of flooding prevention and mitigation deficiencies with equipment, procedures, and analysis. They are frequently related to design and maintenance of flood barriers. Fifteen presented events are grouped in two groups: 1) Inadequate External and Internal Flood Protection Because of Missing or Degraded Flood Barriers; and 2) Inability to Demonstrate the Capability to implement Site External Flood Mitigation Procedures in Time. Short description for selected most relevant events follows (only title is provided for the other events).

2.1 Inadequate External and Internal Flood Protection Because of Missing, Inappropriate or Degraded Flood Barriers

E01 IRS#8311, Fuel oil tank room flooding caused EDG unavailability. On 15.05.2010, fuel oil pumps for emergency diesel generator (EDG) A were flooded with groundwater from malfunction of the wastewater pumps in the plant sewer system. This caused EDG A unavailability for about 30 h. EDG B and gas turbine were available as redundant power source. Even this is finding before Fukushima accident it is still included because it illustrates event outside US. The cause of the wastewater pumps failure was loss of the electrical cabinet. Operators failed to monitor and anticipate flooding. Finally leak tightness of the EDG room was insufficient (lack of civil engineering compliance). Corrective actions included rework of the leak tightness of buildings, alarms modification (in the EGD room and at the pumping station), and operation experience feedback distribution. This event demonstrates that several independent causes could result in safety system train loss and emphasize that redundancy has to be maintained.

E02 IRS#8316, Rainwater ingresses into the RB and TB due to heavy rainfalls[1]

[1] This event with some others is considered in the analysis but for brevity not described in the paper.

Z. Šimić, M. P. Veira, Lessons Learned from Missing Flooding Barriers Operating Experience, Journal of Energy, vol. 65 Number 3–4 (2016) Special Issue, p. 41-50

E03 IRS#8293, Flood in the containment of the Unit 2 (at power) from Unit 1 (in refueling) leading to controlled Unit 2 shutdown. The drainage piping of the spent fuel, refueling, and storage pools interconnects both units of the NPP. Reactor building of the Unit 2 was flooded and forced to controlled shutdown because Unit 1 refueling pools drainage failure. Flooding occurred on 06.09.2011 because of faulty check valve and open isolating valve for the Unit 2 storage and other pools. Even flooding was noticed early the source was uncovered 5 hours later because of the lack of communication between units. Faulty check valve was probably not properly assembled. It total 11 other causes contributing to the incident are identified, e.g.: poor post-assembly control, operational inspection miss after similar event, change of operating instruction which set isolation valves open for operating unit (with unclear reason and later questioned), refueling unit personnel did not verify amount of drained water into the auxiliary building tank, operators of flooded unit could not identify the source, slow sampling analysis, poor response from the plant shift engineer and the technical support group. The major lessons learned from this event are: importance of improved equipment configuration management for flooding protection; systematic review of similar check valves and other identified failures; improved event response procedures.

E04 IRS#8407, Potable water leak leading to controlled shutdown[1]

E05 St. Lucie Unit 1, Internal RAB heavy rain flooding due to degraded conduits lacking internal flooding barrier (IFB). On 09.01.2014, after heavy rainfall (below DBF) and storm drain capacity degradation water backup within Emergency Core Cooling System (ECCS) pipe tunnel and flooded Reactor Auxiliary Building (RAB) through two degraded conduits that lacked IFB, [5.]a. The extent of condition review (ECR) identified four additional conduits that lacked IFB. This event is important because Fukushima related walk-downs in 2012 (per NEI 12-07 guideline [2.]) were successful in finding similar issues (i.e., *Degraded manhole conduit seals bypassed external flood protection*, [6.]c) but clearly not all of them. Initial estimate of the 2012 finding was non-conservative with respect to site water hold up volumes. The problem is related to the permanent change modifications in 1978 (*Primary water degassifier and transfer pump*) and in 1982 (*Waste monitor tank addition*) when six power supply conduits were added without IFB in the ECCS pipe tunnel that penetrated the Unit 1 RAB (located below the DBF elevation). The design packages failed to perform an in-depth evaluation of the changes because items being installed were not safety-related. Further, failure to identify these during two walk downs (2009/10 when corrosion was identified and in 2012 related to Fukushima) highlights the importance of design control (to translate flood protection DB into specification, drawings, procedures and instructions) and need for related periodic inspections implementation. The St. Lucie Unit 2 also has had missing or degraded flood seals. This event and related findings are documented in several reports, e.g. [6.]d.

E06 Arkansas Nuclear One, Turbine generator stator collapse causing leakage and leading to identification of numerous MFB related deficiencies. On 31.03.2013, collapse of the temporary lifting rig carrying main turbine generator stator caused water leakage from the fire system to safety related Decay Heat Removal room through a room drain pipe, [5.]b. The valve was open due to incorrect safety classification and maintenance. Related licensee ECR identified numerous pathways in the auxiliary and emergency diesel fuel storage buildings not effectively sealed, [6.]b. Fukushima related flooding walk-downs earlier identified many other deficiencies. Walk-down was not more effective because of incomplete documentation and inadequate contractors' oversight. Finally, related NRC inspections identified even more deficiencies. Degradation of flood barriers was caused by inadequate design, improper construction, and insufficient maintenance. Findings in total have identified over 100 unsealed pipe penetrations, degraded penetrations seals, un-isolable floor drains, open ventilation penetrations and ductwork, non-water tight door and hatch, and abandoned pipes openings. Further NRC inspection with preliminary yellow finding provides additional details, [6.]a. The problems are related to configuration control, aging and rollout of seals. Corrective actions included: implementing compensatory measures, adding instructions and procedures, sealing penetrations, repeating essential flood protection reviews and completing the missed portions of the walk-downs. This is another example of findings after flooding occurred.

E07 Sequoyah NP, Unanalyzed condition affecting ERCW system due to external flooding[1]

E08 Vermont Yankee NPS, Potential to flood switchgear room due to missing conduit seal[1]

E09 Brunswick Steam Electric Plant Units 1&2, Inspection and walk-downs findings of degraded and nonconforming flooding protection. On 20.04.2011, NRC inspection has found EDGs fuel oil tanks chamber containing openings which would impact flooding mitigation in the case of PMF. This has led to the licensee walk-downs and finding of numerous examples of degraded or nonconforming flood protection features (mainly penetration seals). Further on during the Fukushima walk-downs additional degradations were identified related to the service water, reactor and EDG buildings (e.g., degraded flood penetration seals, conduit seals and gap in the weather stripping along the bottom of the Unit 2 RB railroad door). They also identified EDG rollup door seal deficiency, unsealed shims, leaking flood penetration seals and unsealed conduit (SWB). Together all findings resulted in over 450 work requests/orders and condition reports for degraded or nonconforming flood protection features. According to NRC IR [6.]g licensee failed to correct identified problems. Cause is historical lack of a flood protection program without established preventative maintenance program. These problems existed from 1995 till 2012. The Licensee also failed to develop engineering program to mitigate consequence of external events after NRC finding in 2011. Corrective actions were: repairing degraded seals, developing engineering program for mitigating external flooding, and developing topical design basis for internal and external flooding. This example is more interesting because of timeline and number of findings than because of safety relevance (white[2] NRC finding).

E10 Three Mile Island Station, Inspection and walk-downs findings of flood barriers deficiencies. On 02.08.2012, after Fukushima flooding walk-down observation NRC inspection noted degraded conduits couplings in the Air Intake Tunnel (AIT is the source for safety-related ventilation and it contains safety related electrical conduits). Inspectors also identified 13 unsealed penetrations through the Intake Screen and Pump House and multiple deficiencies to maintain the integrity of the flood barrier. This is an example of cross-cutting aspects: in the area of human performance, decision making, problem identification and resolution, and corrective action program (CAP). Licensee later, in the CAP and ECR, identified a total of 43 deficient external flood barriers (e.g., seals on the Crouse-Hinds Couplings). The plant failed to notice them during comprehensive review in 2010. These deficiencies could allow flood water to impact decay heat removal function. Prompt compensation was: sandbags and earth moving equipment. Permanent corrective actions included sealing the conduits by injecting watertight sealant. This is an example of licensee failure to identify external flood barrier deficiency during Fukushima walk-downs (e.g., they relied on design information instead on as built, etc.) as well as failure during previous review after increasing PMF level in 2011 after river discharge revision. Background and scope of the problem in this example is much more important than safety relevance. More is available in the related NRC IR, [6.]h.

E11 R.E. Ginna NPP, Fukushima flooding walk-down finding MFBs when draining insufficient[1]

2.2 Inability to Demonstrate the Capability to Implement Site External Flood Mitigation Procedures in Time

E12 Watts Bar NP Unit 1, Licensee identified inability to implement in time external flood mitigation procedures[1]

E13 Monticello NGP, NRC identified plant's failure to demonstrate implementation of flood protection against PMF during Fukushima flooding walk-downs[1]

[2] White US NRC finding represents low to moderate safety significance

E14 Fort Calhoun Station, NRC identified inability to protect the intake structure and auxiliary building during external flooding. Demonstration in Sep 2009, revealed that sandbags could not retain 1.5 m static head of water. Licensee failed to implement appropriate corrective actions based on new flooding information. Licensee's ECR revealed unsealed penetrations below the licensing basis flood elevation that could cause intake structure vulnerability during external flood. Corrective actions were: redesign and installation of flood protection features without need for sandbags; sealing affected penetrations; revising procedures. This is yellow[3] finding from NRC IR, [6.]e. Even this is finding before Fukushima accident it is still included because it demonstrates appropriate preventive inspection regarding later licensee's findings and later demonstrated capability to withstand major external flood on June 6, 2011, Figure 2. Licensee's findings include unsealed penetrations through outside wall of the auxiliary and chemistry and radiation protection buildings (below the DBF elevation); unsealed conduits, 2 leading to the auxiliary feedwater pumps and 1 leading to safety related electrical switchgear; drain path from chemistry and radiation protection room to ECCS pumps rom; weak flood protection strategy for raw water pumps. Root causes include: weak procedure revision process; insufficient oversight of work activities; ineffective identification, evaluation and resolution of external flooding protection performance deficiencies; and related "safe as is" mind-set, [6.]f.

E15 Point Beach NP Units 1 & 2, NRC identified failure to establish procedural requirements per FSAR during licensee Fukushima flooding walk-downs. In March 2013, NRC inspection has identified licensee's failure to establish procedural requirements to implement wave run-up protection per Final Safety Analysis Report (FSAR). This was observed during licensee Fukushima flooding walk-downs. There was not enough concrete jersey barriers to protect turbine building and pump house and erection time was not properly considered. They were also missing sandbags to fill barriers gaps. Corrective actions were to modify the jersey barriers to eliminate gaps, stock additional sandbags and jersey barriers and to revise related procedure. This was preliminary yellow with final white finding per NRC Inspection Report, [6.]i.

Figure 2 Fort Calhoun plant on 16.06.2011 during the Missouri River Floods; vital buildings were protected using water-filled perimeter "flood berms" (Source: OPPD Public pres., 04.04.2012)

[3] Yellow US NRC finding represents substantial safety significance.

3 LESSONS LEARNED AND CONCLUSIONS

Properly implemented and maintained flooding protection has potential to significantly increase safety at nuclear power plant because flooding can affect safety systems with common cause potential. The majority of findings presented here are the result of post-Fukushima walk-downs or related inspections. In the US, after Fukushima flooding walk-downs, about 90% of licensees entered an issue into its CAP, [9.]. Furthermore some examples clearly illustrate that even re-evaluation with this walk-downs (inspired by actual accident, supported with industry guidelines approved by regulator [2.] and therefore well planned) were not able to identify all deficiencies. Also all these findings are more reminding about known issues than discovering new insights from the industry wide OE feedback perspective. Evidently they are illustrating how challenging is to implement some of these insights and completely assure flooding protection. This all shows how pervasive flooding protection problem is and also how important are periodic maintenance and review.

Flooding protection related lessons learned are derived from comprehensive assessment of all presented events. The result is a list of actions with a potential to prevent flooding problems resulting in particular consequences. After aggregation, recommended actions are grouped based on potential purposes they could serve. There are four groups of lessons learned related to:

1) Configuration management and monitoring;
2) Non-safety systems interaction;
3) Timely identification of flooding protection problems; and
4) Flooding protection procedures validation.

Table 1 lists all four groups of lessons learned with respective actions (A1-12), purposes (P1-4) and consequences. In total twelve recommended actions are identified with four related purposes and twelve observed consequences (C1-12). Grouping is visible from the table in relations between actions, purposes and consequences/events. By judging from the number of related events and consequences following three recommended actions are standing out:

A4. Treat non-safety system in respect to potential safety impact (events: E02, E04 and E05).

A5. Systematically review a) identified failures for potential similar failures (events E03 and E08) and b) follow up on relevant operating experience (events E04 and E05).

A8. Assure more effective a) maintenance, b) review (events E08 and E10), and c) testing; (for a), b) and c): events E06 and E11).

Here proposed actions have potential to address all identified problems and help prevent specific consequences. However, flooding protection presents continuous challenge because even after all legacy issues are resolved there is need to continuously take care of aging, wear out and modifications related problems. In search for improvements EPRI has recently initiated a project to develop maintenance recommendations associated with flood seals with plan to publish them in the flood protection systems guide, [10.]. Therefore periodic maintenance and review are essential. Figure 3 and Figure 4 are illustrating some measures taken after Fukushima inspired review.

One way to emphasize the most important aspects of flooding protection challenges is perhaps this renaissance observation: "*…in the beginning of the malady it is easy to cure but difficult to detect, but in the course of time, not having been either detected or treated in the beginning, it becomes easy to detect but difficult to cure*." N. Macchiavelli, "The Prince", Chapter III

Table 1 Lessons learned groups, proposed actions, purpose and potentially prevented consequences

1. Lesson learned related to configuration management and monitoring

A1. **Install water level alarms for flooding monitoring (E01).**
P1.a Enable early identification (E01) and reduction (E03) of flooding.
C1. Water accumulation without early identification (E01, E03).

A2. **Assure safety buildings leak tightness water stops (E01).**
P1.b Enable prevention (E01) of flooding.
C2. Safety system flooded (E01, E02).

A3. **Improve flooding protection configuration management (E03).**
P1. Enable prevention (E01), early identification (E01) and reduction (E03) of flooding.
C3. Flooding through open/failed valve (E03).

2. Lesson learned related to non-safety systems interaction

A4. **Treat non-safety system in respect to potential safety impact (E02, E04, E05).**
P2. Better design of the non-safety system to prevent flooding: rainwater drainage system (E02), potable water system (E04), modifications (E05).
C2. Safety system flooded (E01, E02).

3. Lesson learned related to timely identification of flooding protection problems

A5. **Systematically review a) identified failures for potential similar failures (E03, E08) and b) follow up on relevant operating experience (E04, E05).**
P3.bcde Timely identification of potential internal flooding problems (E09, E10), i.e.: known check valve failure (E03), identified MFB (E05), applying learned from similar issues (E08), effective maintenance and review (E06, E07, E08).
C1. Water accumulation without early identification (E01, E03).
C5. Available similar relevant events not used (E04, E05).
C8. Numerous deficiencies discovered only after water leak (E08) or inspection (E09).

A6. **Improve event response (E03).**
P3.b Timely identification of potential internal flooding problems, i.e.: known check valve failure (E03).
C1. Water accumulation without early identification (E01, E03).

A7. **Improve documentation and configuration control (E06, E07).**
P3.efTimely identification of potential internal flooding problems (E09, E10), i.e.: effective maintenance and review (E06, E07, E08), improved contractors' oversight (E06).
C6. Numerous flooding barriers deficiencies not identified after maintenance, review or walk downs (E06).
C7. Inadequate conduit seals installed (E07).

A8. **Assure more effective a) maintenance, b) review (E08, E10), and c) testing; (for all: E06, E11).**
P3.ade Timely identification of potential internal flooding problems (E09, E10), i.e.: sealing for insufficient drainage (E11), applying learned from similar issues (E08), effective maintenance and review (E06, E07, E08).
C7. Inadequate conduit seals installed (E07).
C8. Numerous deficiencies discovered only after water leak (E08) or inspection (E09).
C10. Drain adequacy disproved after inspection and testing (E11).
C11. Licensee identified problem but failed adequate response (E12).

A9. **Establish flood protection engineering program with maintenance (E09).**
P3. Timely identification of potential internal flooding problems (E09, E10).
C8. Numerous deficiencies discovered only after water leak (E08) or inspection (E09).

A10. **Perform thorough follow up on flooding related changes (E10).**
P3. Timely identification of potential internal flooding problems (E09, E10).
C9. Inadequate review after PMF increase with numerous deficiencies (E10).

4. Lesson learned related to flooding protection requirements validation

A11. **Improve analysis of ability to timely implement flooding mitigation procedure (E12, E13, E15).**
P4.a Fulfil flood procedure requirements: for PMF (E12, E13), and required by FSAR (E15).
C11. Licensee identified problem but failed adequate response (E12).
C12.a NRC identified problems during Fukushima flooding walk downs observation (E13, E15)

A12. **Improve flooding procedure revision process, work oversight and protection review (E14).**
P4.b Fulfil flood procedure requirements for new flooding information (E14).
C12.b NRC identified problems during inspection (E14).

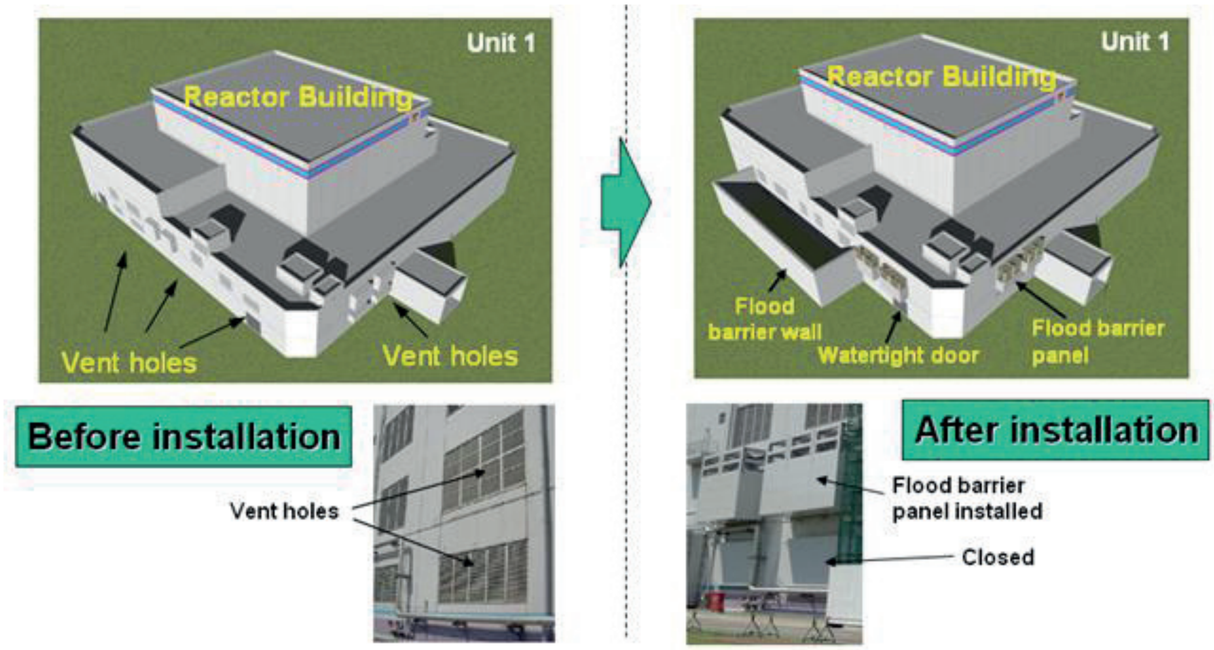

Figure 3 Example of measures to prevent flooding of the building: Kashiwazaki-Kariwa NPS. (Source: TEPCO www.tepco.co.jp/en/nu/kk-np/safety/flood-e.html)

Figure 4 Example of measures to prevent flooding of the critical equipment room and water tight cable conduits penetrations: Kashiwazaki-Kariwa NPS. (Source: TEPCO www.tepco.co.jp/en/nu/kk-np/safety/flood-e.html, tepco.webcdn.stream.ne.jp/…/201404-e.wmv)

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RELAP5/MOD3.3 Analyses of Core Heatup Prevention Strategy during Extended Station Blackout in PWR

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ABSTRACT

The accident at the Fukushima Dai-ichi nuclear power plant demonstrated the vulnerability of the plants on the loss of electrical power for several days, so called extended station blackout (SBO). A set of measures have been proposed and implemented in response of the accident at the Fukushima Dai-ichi nuclear power plant. The purpose of the study was to investigate the application of the deterministic safety analysis for core heatup prevention strategy of the extended SBO in pressurized water reactor, lasting 72 h. The prevention strategy selected was water injection into steam generators using turbine driven auxiliary feedwater pump (TD-AFW) or portable water injection pump.

Method for assessment of the necessary pump injection flowrate is developed and presented. The necessary injection flowrate to the steam generators is determined from the calculated cumulative water mass injected by the turbine driven auxiliary feedwater pump in the analysed scenarios, when desired normal level is maintained automatically. The developed method allows assessment of the necessary injection flowrates of pump, TD-AFW or portable, for different plant configurations and number of flowrate changes.

The RELAP5/MOD3.3 Patch04 computer code and input model of a two-loop pressurized water reactor is used for analyses, assuming different injection start times, flowrates and reactor coolant system losses. Three different reactor coolant system (RCS) coolant loss pathways, with corresponding leakage rate, can be expected in the pressurized water reactor (PWR) during the extended SBO: normal system leakage, reactor coolant pump seal leakage, and RCS coolant loss through letdown relief valve unless automatically isolated or until isolation is procedurally directed. Depressurization of RCS was also considered. In total, six types of RCS coolant loss scenarios were considered. Two cases were defined regarding the operation of the emergency diesel generators. Different delays of the pump injection start following the station blackout were assumed and analysed. For each scenario, two kinds of SBO calculations for two-loop PWR were performed, base and verification. Base calculations were needed to determine necessary minimum flowrate for steam generators feeding in such a way that they are not overfilled or emptied. Namely, it was assumed that instrumentation is not available during extended SBO. The verification calculations have been then performed to verify if the determined minimum flowrates are sufficient to prevent the core heatup.

The calculated results show effectiveness of the proposed extended SBO prevention strategy provided that the water injection is available in the first two hours after SBO occurring at full power. If diesel generator is running after loss of offsite power for some time, e.g. one hour, the available time for steam generator water injection is significantly longer. The obtained results demonstrate the need for assessment of the pump injection flowrates before the utilization of the pump for mitigation of the event. The applicability of the developed method for assessment of the required pump injection flowrate has been validated on PWR.

1 INTRODUCTION

The events at the Fukushima Dai-ichi nuclear power plant [1] and stress tests [2] showed that the loss of electrical power (LOOP) followed by station blackout event (SBO) and loss of the ultimate heat sink (UHS) can have large impact on the safety of the nuclear power plant (NPP). The SBO event when power from all emergency power sources, including batteries, is lost is named extended SBO and leads ultimately to core heatup and core damage [3]. After Fukushima Dai-ichi accident the strategies were proposed for coping with such events. They include utilization of portable equipment, permanent equipment or combinations of portable and permanent equipment. Severe accident management guidelines (SAMGs) have been also updated [5]. In the United States of America (USA) the Diverse and Flexible Coping Strategies (FLEX) [4] have been developed which are focused on maintaining or restoring key plant safety functions. Further, in FLEX guidelines [4] it is stated that while FLEX strategies are focused on the prevention of fuel damage, these strategies would be available to support accident mitigation efforts following fuel damage. However, coordination of the FLEX equipment with Severe Accident Management Guidelines (SAMGs) is not within the scope of the guideline.

For the LOOP, SBO and the loss of the UHS scenarios, cooling of the core can be established by means of water injection to reactor coolant system (RCS) and/or steam generators (SGs). For RCS injection the source of borated water is needed. On the other hand, for SG makep sustained source of water is needed. Different systems for performing this function have been identified in the stress test reports, including electric power-independent turbine driven pumps, arrangements for gravity feed, dedicated diesel driven pumps and pre-installed connections for external feed, such as from the on-site fire trucks [6].

The FLEX strategies also suggest development of thermal hydraulic analyses to support plant specific decision-making. Therefore in this paper utilization of the pump, either turbine driven auxiliary feedwater or portable pump, for mitigation of the extended SBO event and prevention of core damage in pressurized water reactor (PWR) is investigated. It presents a follow-up study to stress tests [7]. Methodology is developed for assessment of the necessary fix/portable pump flowrates within analysed time period. It should be noted that the FLEX strategy to inject water into steam generator [4] suggests that in certain circumstances, auxiliary feedwater (AFW) system operation may be extended by throttling flow to a constant rate, rather than by stroking valves in open-shut cycles. Further, the assessed flowrates should also prevent the core damage without overfilling the steam generators in the analysed scenarios.

The paper is organised as follows. The description of the deterministic safety analysis input model, the methodology used for the assessment of the necessary injection flow of the pump to SGs and developed case scenarios is given in Section 2, while in Section 3 the obtained results from the deterministic safety analyses are given.

2 METHODS DESCRIPTION

For calculations the RELAP/MOD3.3 Patch 04 thermal-hydraulic computer has been used [8]. The RELAP5 input model of NPP with two-loop PWR, described in details in refs. [7] and [9] is briefly described first. The method for calculation of necessary pump flowrate during extended SBO event is described. Finally, the scenarios are described.

2.1 RELAP5 Input Model

The noding scheme of PWR, used in the study, is shown in Figure 1. It was created using Symbolic Nuclear Analysis Package (SNAP) [10]. The base plant model consists of 469 control volumes, 497 junctions and 378 heat structures with 2107 radial mesh points. However, since in SNAP the pipes are represented as one component (not by volumes) and since heat structures connected to pipe volumes are represented as one heat structure, the number of SNAP hydraulic components is 304 and the number of heat structures is 108. The connection point for the portable pump is the same as for turbine driven auxiliary feedwater pump.

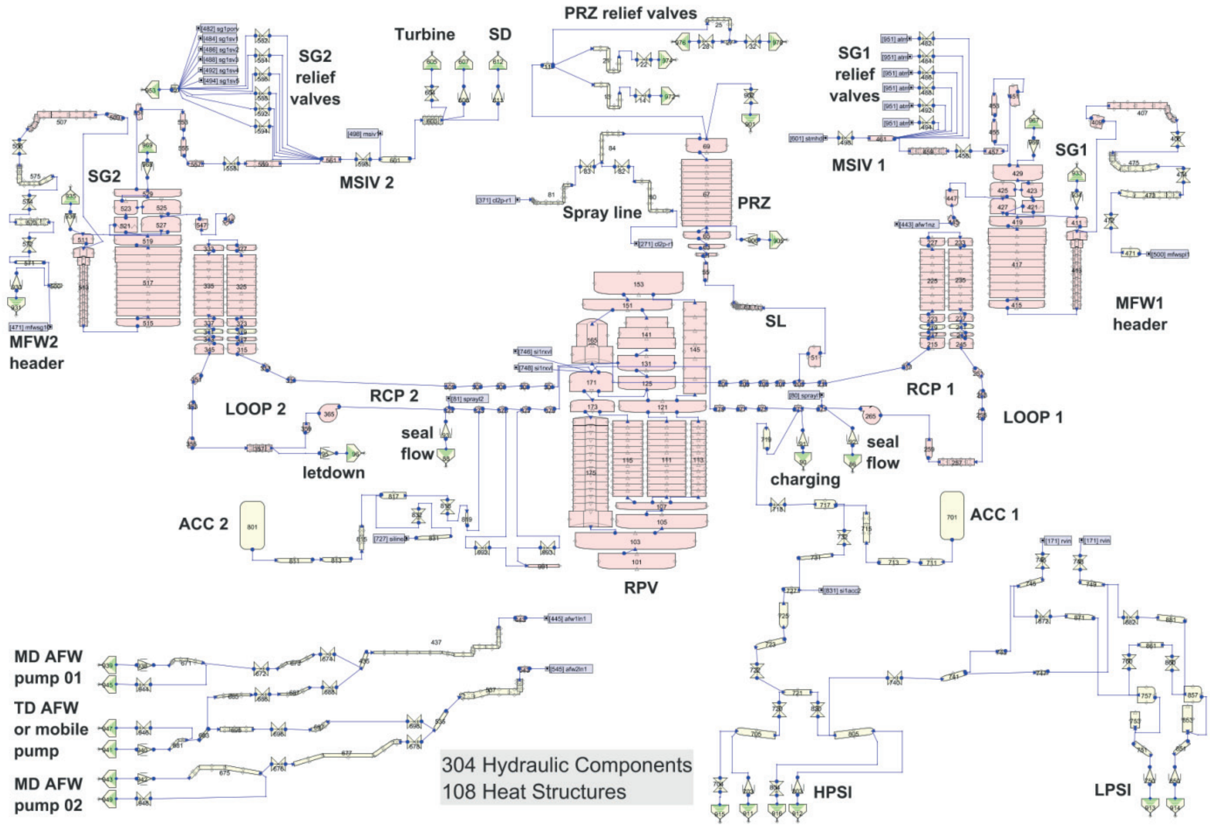

Figure 1: Noding scheme of two-loop PWR represented by SNAP

2.2 Method for Necessary Flowrate Determination

Base calculations of extended SBO with assuming available TD-AFW pump and steam generator level control are needed to determine the necessary minimum flowrate for steam generators feeding in such a way that they are not overfilled or emptied. The later verification calculations are performed to verify if the determined minimum flowrates are sufficient to prevent the core heatup. The method for necessary flowrate consists of the following steps:

- Step 1: From the base calculation for given scenario with given boundary conditions the integral of necessary water mass to be injected into steam generators is obtained to restore and maintain the desired water level. The analysis and evaluation of the cumulative water mass is done for time a window equal to the operational time of the pump during extended SBO. The scenarios that in the analysed time window result in the core heatup and damage need not to be considered in verification calculations.
- Step 2: From cumulative water mass the necessary mass flowrate of the pump for steam generators makeup has to be set so, that the integral water mass injected in the steam generators at a given time ideally corresponds to the mass, determined in step 1, but in any case remains in the operable range of the steam generators (between 8% and 96%). In the simplest case the necessary mass flowrate is set constant for a longer time period. Care should be taken that the mass flowrate is high enough because of the initial period, in which the residual power is higher than in later times. It may turn out that for longer time periods this may not be achieved with a constant mass flowrate (i.e. the constant flow to

be too small in initial period) and that more adjustments of the mass flowrate are required as we progress with time.

- Step 3: The verification calculations are performed with the determined necessary flowrate in the Step 2. These calculations verify if injected water into the SGs, for given scenario, prevents core heat up and SGs overfill.

2.3 Scenarios description

Different scenario types have been investigated to demonstrate proposed method for necessary flow, including plant design improvements which may be done to better cope with extended SBO. During extended SBO three different RCS coolant loss pathways, with corresponding leakage rate, can be expected in the PWR: (a) normal system leakage, (b) RCPs seal leakage, and (c) RCS coolant loss through letdown relief valve unless automatically isolated or until isolation is procedurally directed. Normal system leakage is assumed to be equal to plant technical specifications identified leakage (0.63 l/s at nominal RCS conditions) for selected PWR. The seal leak rate of 1.32 l/s per RCP (at nominal RCS conditions) due to loss of seal cooling and RCP pump stop is assumed as representative for the plants using a high temperature O-ring RCP seal package [11]. The seal leakage can be practically prevented (negligible loss of the order of 0.06 l/s) with the installation of special passive RCP thermal shutdown seals [12]. At nominal RCS pressure the letdown loss of 5.68 l/s is expected and this value was assumed in the study.

To limit the RCS loss and enable passive accumulator injection the RCS depressurization (through the depressurization of the secondary side) scenarios have been also investigated.

The main assumptions considered in the model for verification calculations are the following:

- A1) Loss of all electric power in the plant, including batteries, what results in loss of active safety systems depending on AC power and loss of all instrumentation and control in the plant.

- A2) The pump, TD-AFW or portable (through available connection point), is available for the whole analysed period for injection of water in the steam generators. The portable pump flow measurement and regulation is also assumed to be available in the study. The TD-AFW pump speed is manually controlled. Hand wheels are provided for local manual operation of TD-AFW control valves. The TD-AFW pump flowrate local indicators, not relying on electric power, are also available locally.

- A3) Availability of water for operation of the water pump is assumed in the model.

- A4) Pressurizer and steam generator safety valves are assumed available.

- A5) The nitrogen gas required for the operation of the steam generator power operated relief valves (SG PORVs) is assumed available in the RCS depressurization scenarios. The alternative compressed air supply from the portable diesel compressor is providing required gas.

- A6) The criterion used as indication for the steam generator overfill is 96% of wide range (WR) steam generator level, while for the loss of heat sink is 8%.

- A7) The criterion used for core heatup is significant core uncovery causing core heatup start.

In base calculations it was in addition assumed the operable TD-AFW with all instrumentation and control. The regulation of the TD-AFW was set to restore and maintain narrow range level at 69% (plant normal level, at normal power condition this means 77% wide range level). Using such assumptions the integrated mass injected to steam generators is obtained, which satisfy A6 criterion that SGs are not overfilled.

Six types of RCS coolant loss scenarios, given in Table 1, have been developed and analysed: the NO_LOSS (no RCS loss), N_LOSS with normal system leakage, the S_LOSS with RCP seal loss starting one hour after the start of extended SBO, the SL_LOSS with RCP seal loss and loss of coolant through the letdown relief valve when RCS pressure is greater than 4.24 MPa, SLD_LOSS with RCP seal and letdown loss (if RCS pressure is greater than 4.24 MPa) and depressurization of the primary side through the secondary side to 1.57 MPa, started 30 minutes after SBO occurrence, and NSLD_LOSS (SLD_LOSS case with additionally assumed normal system leakage).

Table 1 Types of RCS Coolant Loss Scenarios

Scenario type	**N**ormal system leakage	**S**eal loss	**L**etdown loss	**D**epressurization
NO_LOSS	no	no	no	no
N_LOSS	yes	no	no	no
S_LOSS	no	yes	no	no
SL_LOSS	no	yes	yes	no
SLD_LOSS	no	yes	yes	yes
NSLD_LOSS	yes	yes	yes	yes

For each type of loss scenarios given in Table 1 two cases were simulated:

(1) In the Case 1 (C1) it is assumed that the emergency diesel generators (EDG) started and normally operated for one hour after the loss of offsite power. Availability of all safety systems is assumed during that hour. After one hour the extended SBO is assumed. It was similar in Fukushima Dai-ichi event when the EDGs were running for 45 minutes until tsunami arrived.

(2) In Case 2 (C2) it is assumed that the extended SBO occurrence is concurrent with LOOP, resulting in large decay heat in the initial period. This is more severe as larger time delays of SBO start (i.e. delays of EDG loss) would mean smaller decay heat levels at the time of SBO occurrence (i.e. less heat to be removed from the core).

Different time delays between the extended SBO start and start of the pump injections to steam generators have been considered and analysed in base calculations. In the Case 1 delays 0, 0.5 h, 1 h, 2 h, 3 h, 4 h and 5 h (42 scenarios in total) and in the Case 2 delays 0, 0.5 h, 1 h, 2 h and 3 h were considered (30 scenarios in total). The scenario name is compouned from scenario type name and the delay time (e.g. NO_LOSS type scenario with 4 hour delay of steam generator feeding is labelled as NO_LOSS_4). The no delay scenarios have been analysed for base calculations to verify if for selected scenario type the core heatup could be prevented by proposed strategy with SGs makeup.

The calculations have been performed for 72 h, consisting of two time intervals for injections. The first time interval is 24 h reduced for pump start delay. The second time interval lasts from 24 h to 72 h. In the simplest case, the flowrate in the interval is constant.

3 RESULTS

3.1 Base Calculations

The obtained necessary flowrates obtained from base calculations using method described in Section 2.2 are shown in Table 2. The first row in Table 2 specifies the RCS coolant loss scenario type defined in Section **Error! Reference source not found.**. The first column contains the delay time, in hours, of the pump start following the extended SBO. The pump constant flows in the first time interval (from pump start until 24 h), in kg/s, are given in the remaining columns for Case 1 and Case 2 scenarios. The largest minimum necessary flows are obtained for SLD_LOSS type scenarios and the smallest for S_LOSS type scenarios. For the second time interval starting 24 h after SBO start and lasting 48 h the constant mass flow of 3 kg/s was assumed. Finally, Table 2 shows that core heatup and damage is calculated for some of the analysed scenarios.

Table 2: Minimum Necessary Constant Flowrates in First Time Interval (from Pump Start until 24 h after SBO Start)

Scenario type	NO_LOSS	N_LOSS	S_LOSS	SL_LOSS	SLD_LOSS	NSLD_LOSS
Case 1, EDG running 1 h						
Delay (h)	Flow (kg/s)*	Flow (kg/s)*	Flow (kg/s)*	Flow (kg/s)*	Flow (kg/s)*	Flow (kg/s)*
0	4.79	4.77	4.68	N.A.	5.51	5.50
0.5	4.90	4.87	4.78	N.A.	5.63	5.62
1	4.99	4.98	4.88	N.A.	5.75	5.74
2	5.22	5.20	5.11	N.A.	6.00	5.99
3	5.47	5.45	5.34	N.A.	6.25	6.25
4	5.63	5.62	5.51	N.A.	6.48	6.47
5	N.A.	N.A	N.A.	N.A.	N.A.	N.A.
Case 2, EDG not running						
Delay (h)	Flow (kg/s)*	Flow (kg/s)*	Flow (kg/s)*	Flow (kg/s)*	Flow (kg/s)*	Flow (kg/s)*
0	5.50	5.46	5.38	N.A.	6.21	6.19
0.5	5.60	5.58	5.49	N.A.	6.34	6.33
1	5.72	5.70	5.61	N.A.	6.46	6.45
2	5.86	5.83	5.72	N.A.	6.54	6.57
3	N.A.	N.A.	N.A.	N.A.	N.A.	N.A.

* - N.A. because the core heatup occurs due to the core uncovery before 24 h

For all SL_LOSS scenarios, as shown on Figure 2(a1) and Figure 2(a2), core uncover (results until calculation was aborted) before 24 h. Obtained results are expected considering large letdown loss of RCS coolant, causing core uncovery as shown on Figure 2(a1) and Figure 2(a2). When steam generator empties (see on Figure 2(b1) and Figure 2(b2)), the RCS starts to heat what resulted in the pressure increase and additional RCS inventory loss due to pressurizer safety valves opening. Even if SG injection started before the steam generators are emptied as shown on Figure 2(b1) and Figure 2(b2), continuous RCS mass loss due to RCP seal leakages and unisolated letdown line loss resulted in the core uncovery in approximately 12 h (in case of significant accumulator injection this time may be prolonged for some hour - see scenario SL_LOSS_3 (C1)).

The core heatup, as shown in Table 2, is obtained also for other type scenarios when pump operation is delayed for extended period. Figure 3 shows that the core uncovery is not prevented for C1 scenarios with delay 5 hours and for C2 scenarios with delay 3 hours. With no heat sink for the core decay heat due to steam generator boil-off, the RCS starts to heat up and the primary pressure increases until pressurizer safety valves open. This resulted in RCS mass discharge and core uncovery with heatup. As can be seen from Figure 3, the EDG operating one hour delays core uncovery for additional two hours. This demonstrated that operation of safety systems in the initial hour besides cooling the core gives additional time to operators and requires smaller injection flows (smaller demand for pump flow capacity). In the worst case scenario, with SBO occurring at full power, the core starts to uncover after 2 h in case no injection to steam generators is available. This means little available time to the operators, especially to use the portable pump.

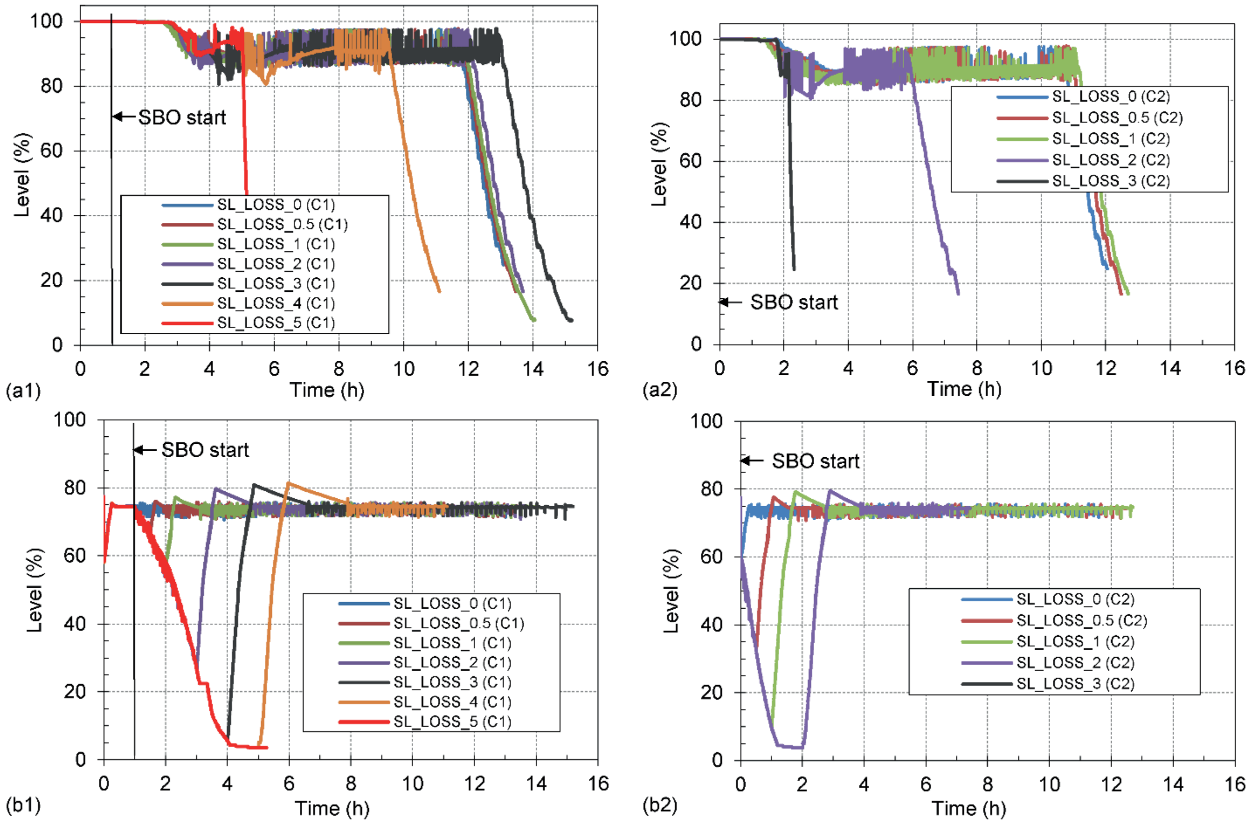

Figure 2 Comparison of C1 and C2 Cases for SL_LOSS Type Base Calculations with Different SG Injection Delays: (a) Core collapsed liquid level, (b) SG. no. 2 WR level.

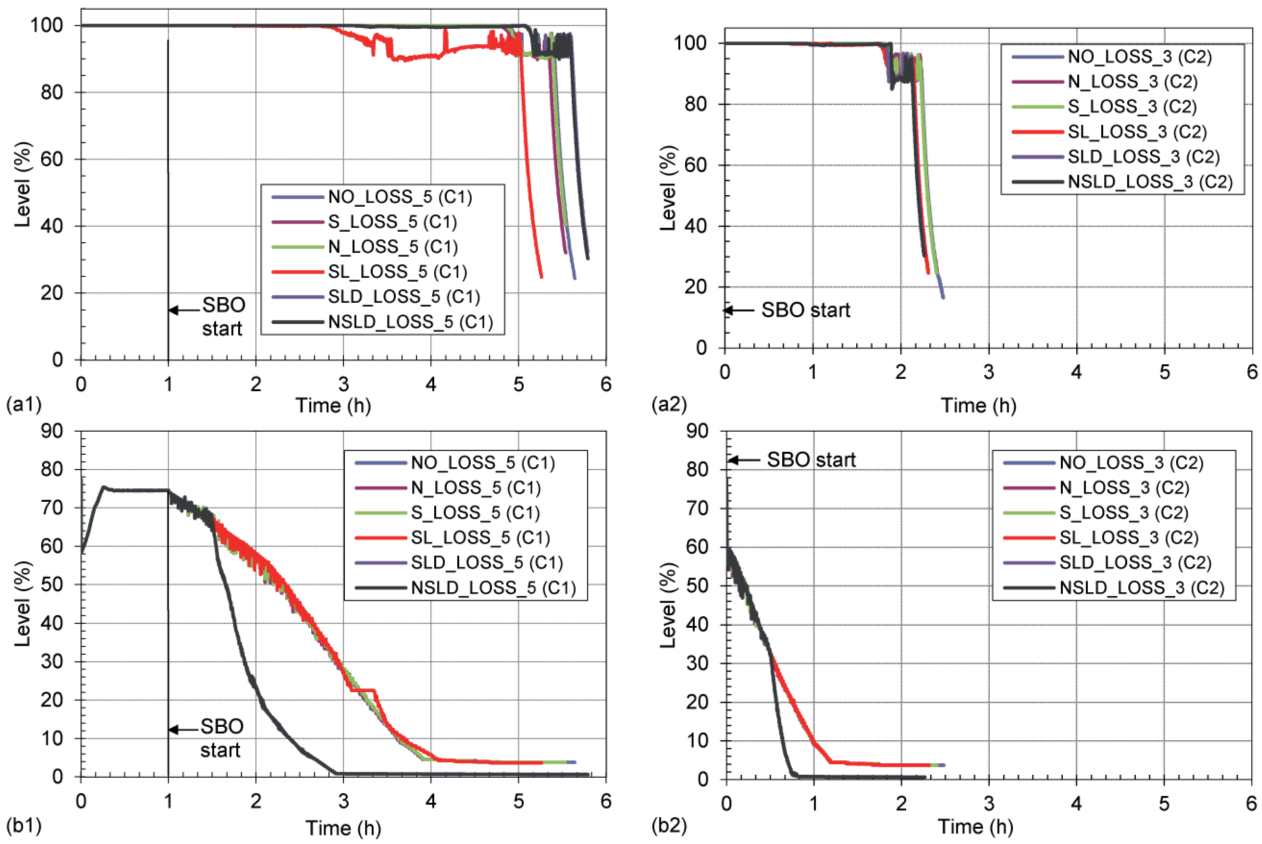

Figure 3: Comparison of Base Calculations for C1 (5 h Injection Delay) and C2 (3 h Injection Delay): (a) Core collapsed liquid level, (b) SG. no. 2 WR level.

3.2 Verification Calculations

The third step in the methodology presented in Section 2.2 is to verify the calculated minimum necessary pump flowrates given in Table 2. The verification calculations are performed for all scenarios listed in Table 2 in which core heat up is prevented. The results are shown in Figures 4 through 7 for RCS pressure, core collapsed liquid level, SG no. 1 and SG no. 2 wide range level, respectively. The results are shown for the 72 h except for RCS pressure the 24 h interval is used as the initial hours are the most important from the point of RCS discharge through safety valves. Later (after one day) the RCS pressure start to follow the secondary pressure or it is on this trend. Each of Figures 4 through 7 show five scenario types with different assumed injection delays (all types except SL_LOSS) for both cases (C1 and C2).

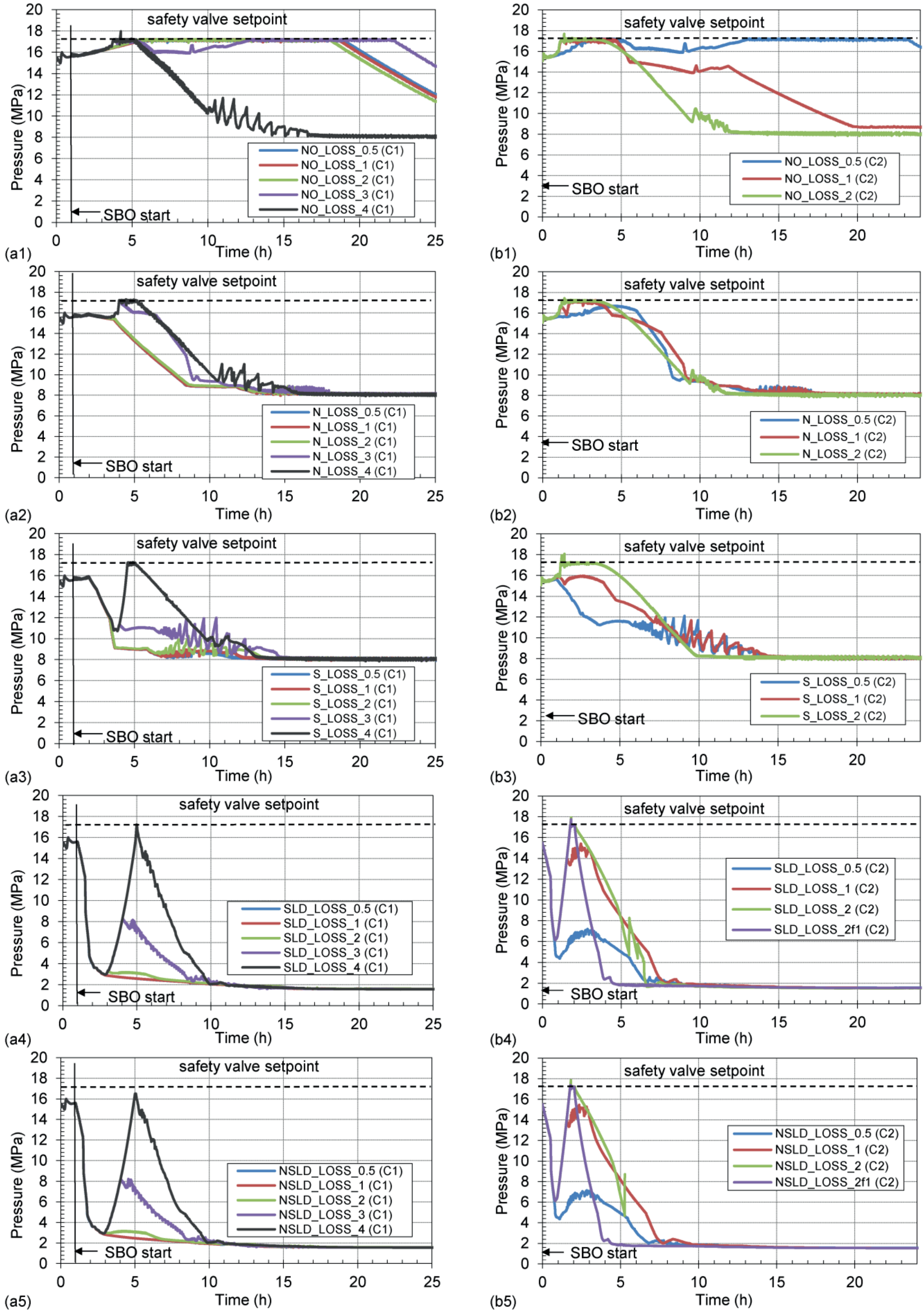

Figure 4 shows that it takes the longest time for NO_LOSS type scenarios that the RCS pressure drops below the safety valve setpoint (see Figures 4(a1) and (a2)). In all other type scenarios the one or more RCS losses are present (see Section 2.3) resulting in faster initial RCS pressure decrease. If secondary side cooling is not available, the RCS may repressurize until the injection into SGs starts. As normal RCS loss is smaller than RCS loss through RCPs, the pressure decrease is slower in N_LOSS type scenarios than in the S_LOSS type scenarios. However, if depressurization is performed, there is no qualitative difference between SLD_LOSS and NSLD_LOSS type scenarios.

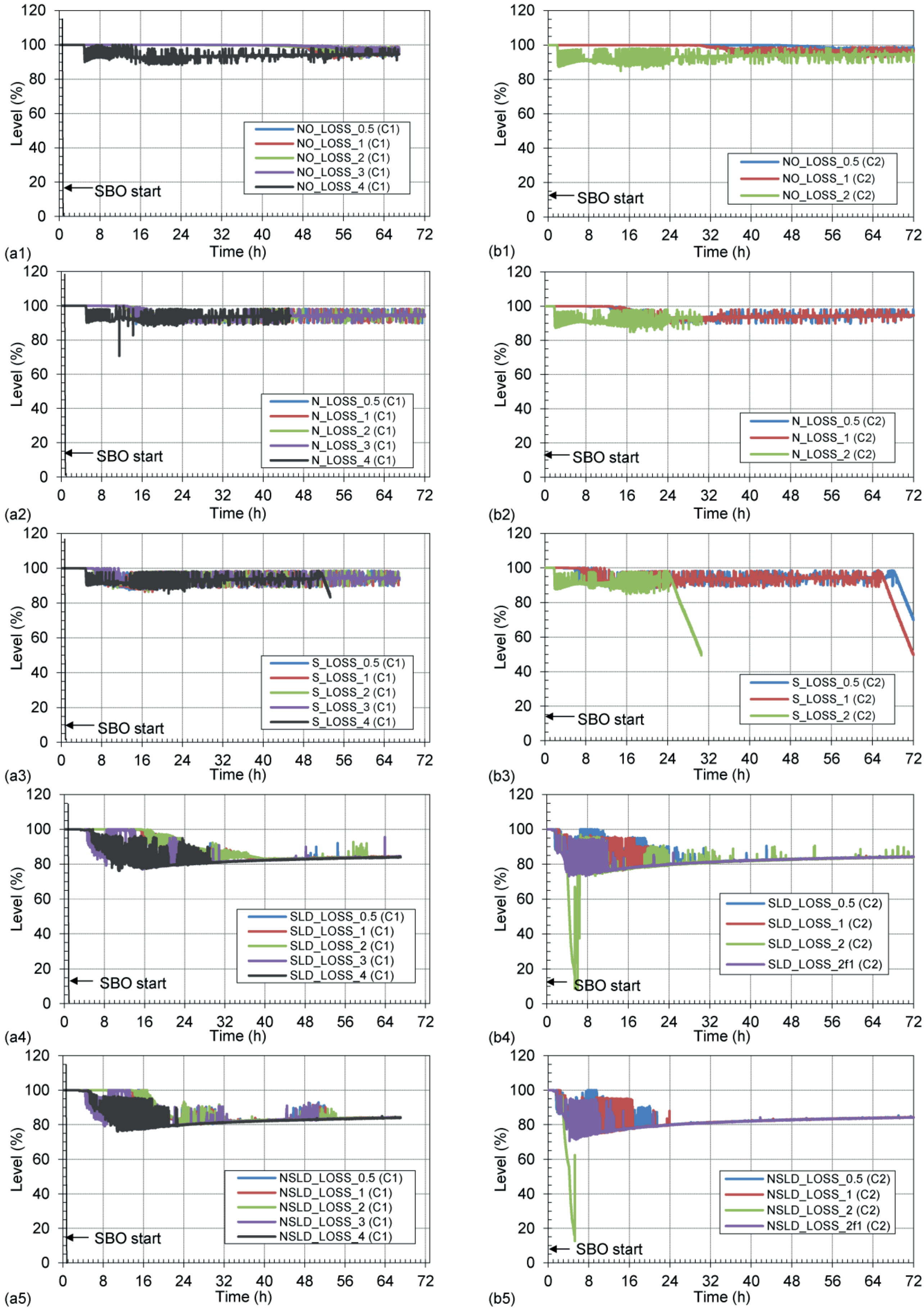

Figure 5 shows that the core in the first 24 h is not significantly uncovered except for SLD_LOSS_2 and NSLD_LOSS_2 scenarios. The reason is too small flowrate in the beginning of transient comparing to base calculations in which according to assumption the SG level was automatically maintained. When the first time interval was divided in two sub-intervals, i.e. assuming the constant flow of 10 kg/s (verification calculations SLD_LOSS_2f1 (C2) and NSLD_LOSS-2f1 (C2)) in the first four hours and 5.77 kg/s in the remaining 18 hours instead of 6.54 kg/s until 24 h, core uncovery was prevented as shown in Figures 5(b4) and 5(b5). However, when larger RCS loss is present (S_LOCA type scenarios, Case 2), after one day the core may start to uncover, even if there is sufficient water inventory in the steam generators. This time may be prolonged, if injection to SGs is started very early (in the first hour after SBO occurrence). On the other hand, if EDG is running one hour after LOOP with reactor trip (Case 1), the decay heat level is lower than after reactor trip and therefore the calculations showed that the core remains uncovered if injection into SGs starts in 3 h. Nevertheless, when the reactor is depressurized the core uncovery is prevented for assumed RCP seal loss and maximum normal RCS loss in the first 72 h with proposed flow injection to the steam generators.

Finally, Figures 6 and 7 show steam generator wide range levels for steam generator no. 1 and 2, respectively. It is clearly demonstrated the efficiency of the method for necessary flow determination and at the same time not to overfill or empty steam generators. It can be seen that steam generator levels in all shown scenarios are well maintained. As one pump is feeding both steam generators, in certain scenarios (e.g. SLD_LOSS_3 (C1)) the SG level oscillates. It should be noted, when level in one steam generator increases, in the other decreases, and vice versa. The small oscillations present in the SG level trends are due to SG relief valves discharge at the discrete periods. Nevertheless, later the filling of both steam generators is smooth and similar for both steam generators.

The assessed flowrates from the TD-AFW cumulative injected mass in the base calculations of analysed case scenarios minimize the required number of flow changes and potential operator's errors during required manipulations. The assessed flow for steam generator makeup assures effective core cooling without overfilling the steam generators.

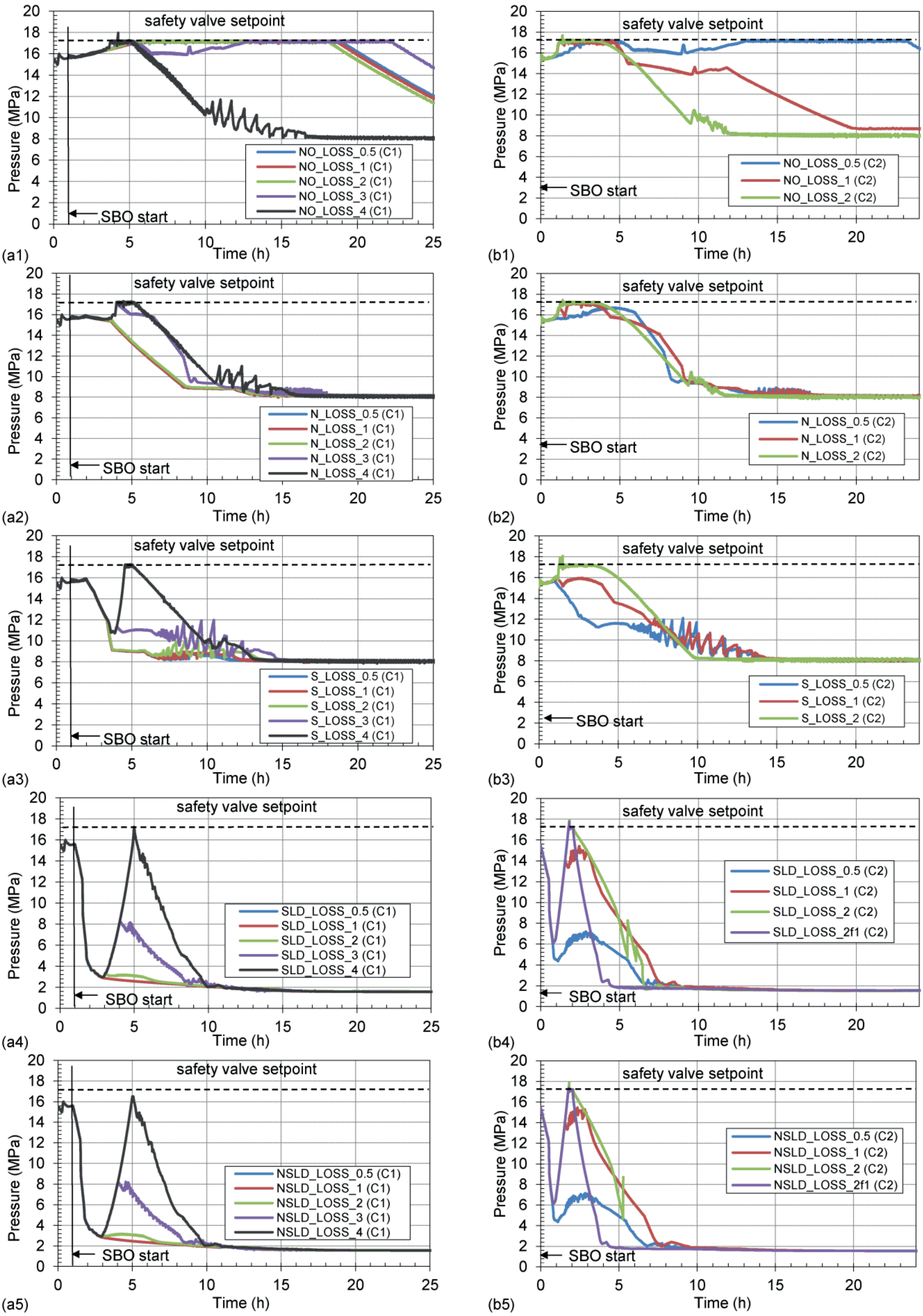

Figure 4: Dependence of RCS Pressure on Injection Time Delays during Different Scenario Types for: (a) C1 – SBO Start 1 h after LOOP, (b) C2 – SBO Concurrent with LOOP.

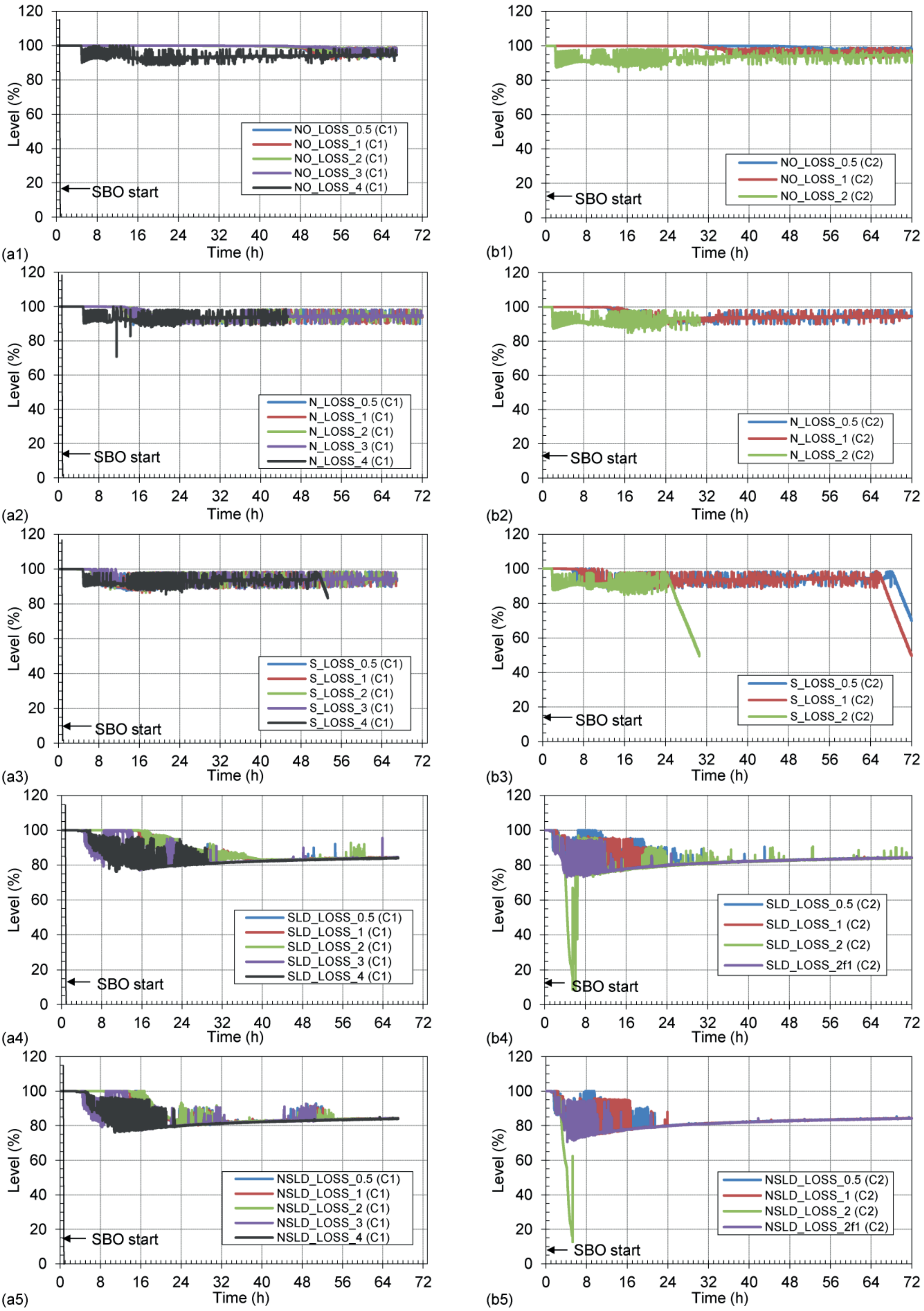

Figure 5: Dependence of Core Collapsed Liquid Level on Injection Time Delays during Different Scenario Types: (a) C1 – SBO Start 1 h after LOOP, (b) C2 – SBO Concurrent with LOOP.

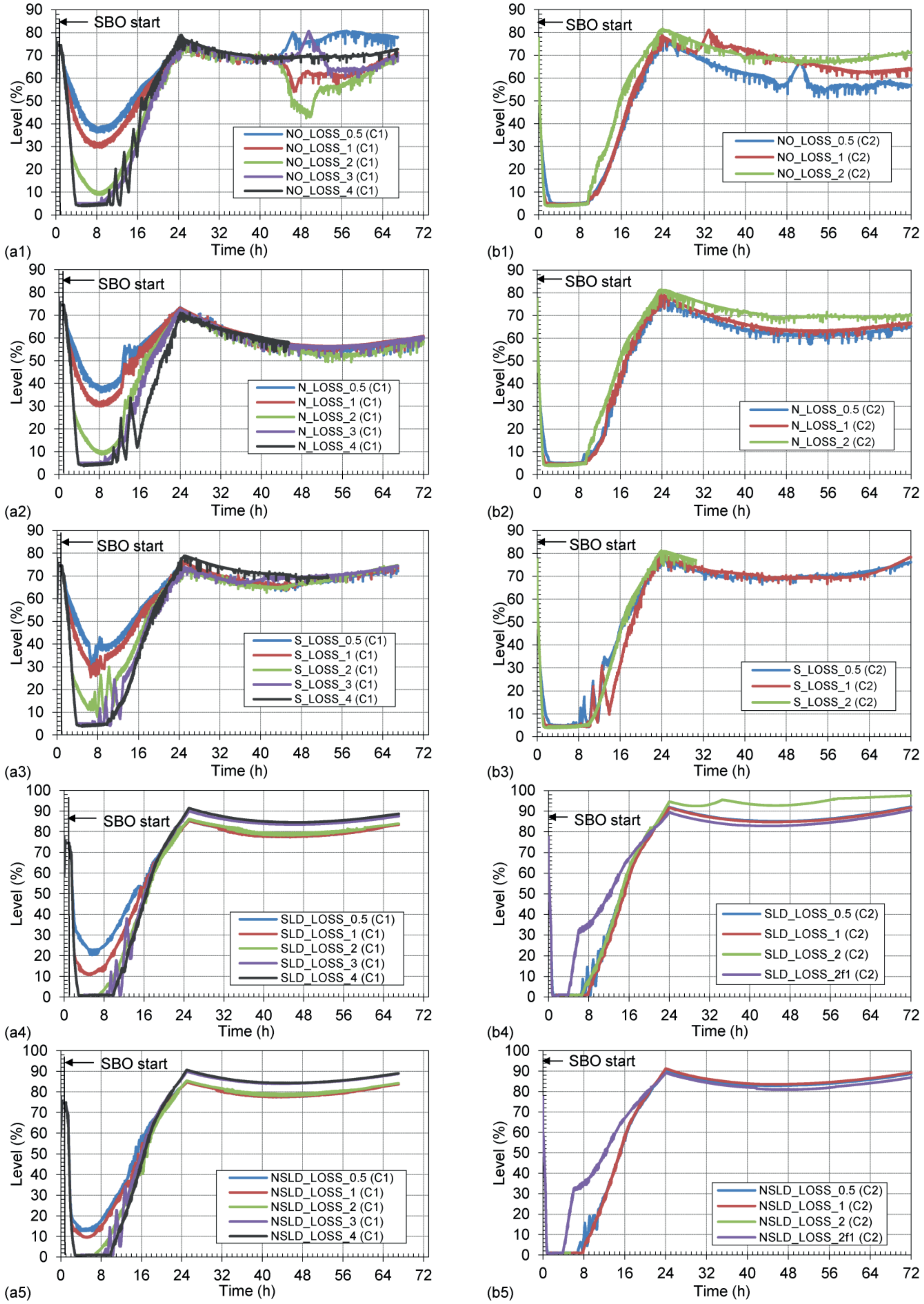

Figure 6: Dependence of SG No. 1 WR Level on Injection Time Delays during Different Scenario Types: (a) C1 – SBO Start 1 h after LOOP, (b) C2 – SBO Concurrent with LOOP.

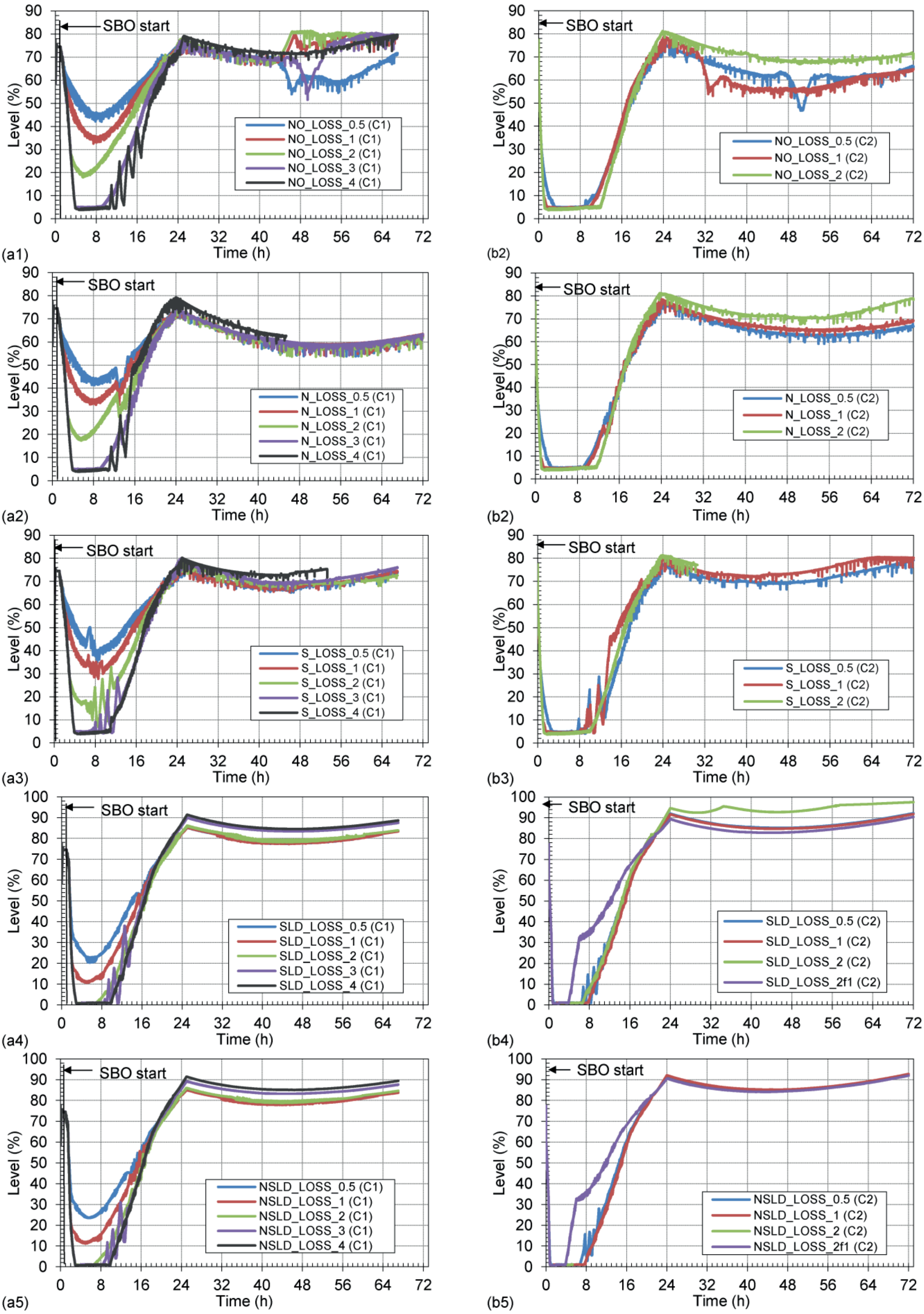

Figure 7: Dependence of SG No. 2 WR Level on Injection Time Delays during Different Scenario Types: (a) C1 – SBO Start 1 h after LOOP, (b) C2 – SBO Concurrent with LOOP.

4 CONCLUSION

To support the FLEX implementation to prevent damage to the fuel in the reactor the RELAP5/MOD3.3 Patch04 thermal-hydraulic system code has been used to study the utilization of pump for mitigation of the extended blackout condition. The method for the assessment of the necessary injection flowrate for steam generators makeup for mitigation of extended blackout event has been proposed.

Six different scenarios of reactor coolant loss have been developed and analysed. Two cases have been defined considering the operation of the emergency diesel generators. Different delays of the pump injection (into steam generators) start following the station blackout have been assumed and analysed.

Obtained results show that typical pressurized water reactor with the leak tight reactor coolant pump seals with small seal loss or depressurization of the reactor coolant system can cope the first 72 hours of extended SBO if pump is available to start inject into secondary side within two hours of the SBO start. Operation of the emergency diesel generator for one hour extends the available time for the start of pump on four hours. Failure to isolate the letdown results in the core damage before 24 h in all analysed scenarios.

The effective mitigation strategy for the extended blackout condition can be developed with the utilization of the presented method. The method can be modified for different type of pumps and their characteristics. One of the main conclusions in the study is that availability of equipment is prerequisite but not guarantee of successful mitigation. Namely, the verification analyses suggest that the time available requires quick response of operators in case of using portable equipment. On the other hand, base calculations show that there may be some delay in TD-AFW pump start and that it is urgent to isolate letdown line in order to prevent core heatup, when no RCS injection is available.

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Application of Best Estimate Plus Uncertainty (BEPU) Methodology in a Final Safety Analysis Report (FSAR) of a Generic Plant.

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ABSTRACT

The licensing process of a nuclear power plant is motivated by the need to protect humans and the environment from ionizing radiation and, at the same time, sets out the basis for the design and determining the acceptability of the plant. An important part of the licensing process is the realization of accident analysis related to the design basis, which should be documented in the Final Safety Analysis Report (FSAR). There are different options on accidents calculation area by combining the use of computer codes and data entry for licensing purposes. One is the Best Estimate Plus Uncertainty (BEPU), which considers realistic input data and associated uncertainties. Applications of BEPU approaches in licensing procedures were initiated in the 2000s, first to analysis of Loss of Coolant Accident (LOCA), and then to the accident analysis as a whole, documented in Chapter 15 of the FSAR. This work has as main objective the implementation of BEPU methodology in all analyses contained in FSAR, through the homogenization of the analytical techniques and identification of key disciplines and key topics in the licensing process.

Keywords: *Licensing, Nuclear Power Plants, Safety Analysis, Final Safety Analysis Report, BEPU*

1 INTRODUCTION

Nuclear Reactor Safety Technology (NRST) is the set of materials, components, structures, procedures and numerical tools used to minimize the risk of contamination of humans and environment by radioactive material. NRST has been established for several decades, since the discovery of nuclear fission and since that time, any installation involving the use of radioactive material has been designed according to safety requirements [1].

Nuclear safety has become a technology following extraordinary industrial investments since the 50's. A step impulse to the technology came when powerful computers were available at the beginning of the 80's [1]. Events in the last decades occurring in the Three Mile Island Unit-2, Chernobyl Unit-4 and Fukushima Units1-3 have challenged the sustainability of nuclear technology and undermined the trust of the public, of the decision makers and even of the scientific community toward nuclear safety [2].

The NRST consists of two components – the Fundamentals and the Application – as demonstrated in Figure 1. The first component includes the key safety objective, the related safety principles, and safety requirements developed by the International Atomic Energy Agency (IAEA). The Application refers to the application of those principles and requirements for the design, licensing, construction, operation and decommissioning of any nuclear installation [2].

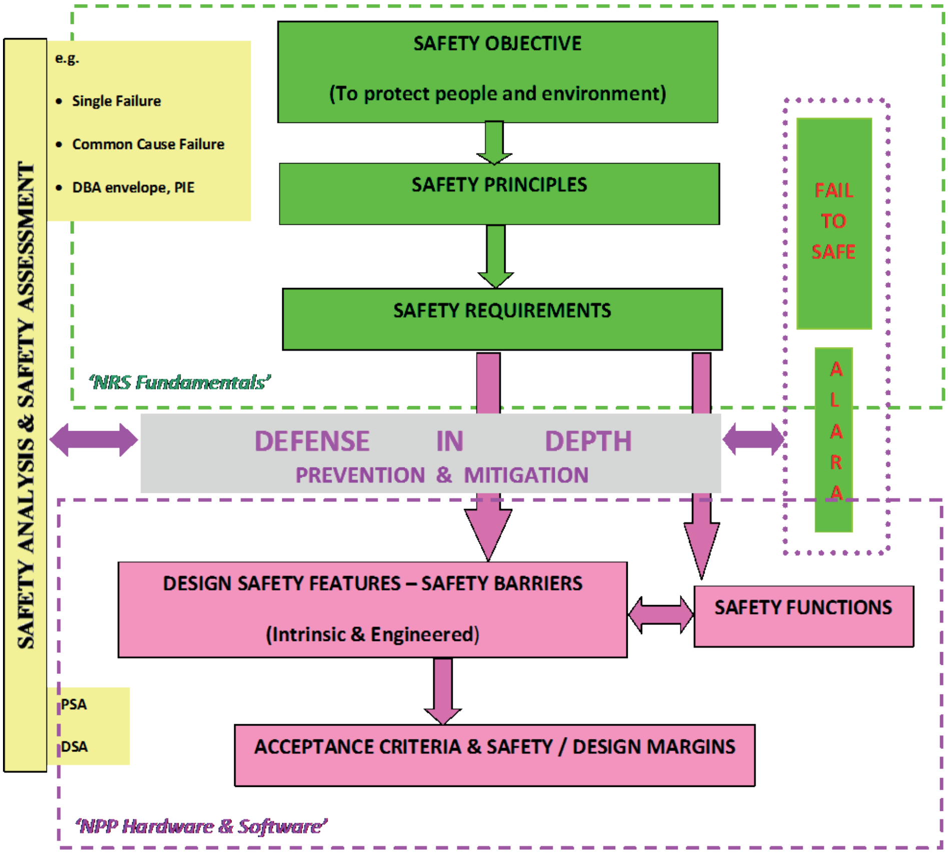

Figure 1: Simplified sketch for Nuclear Reactor Safety Technology.

The accomplishment of safety fundamentals in the Nuclear Power Plant (NPP) design is achievable by suitable safety analysis and assessment. The safety evaluation of the NPP is based on the fulfilment of a set of design acceptance criteria such as maximum peak cladding temperature, maximum pressure in the primary system, among others, to be met under a wide range of plant operating conditions to confirm the preservation of physical barriers [3].

The national regulator normally defines the acceptance criteria, and a comprehensive Safety Analysis Report (SAR) for individual NPP provides the demonstration that the plant is safe and, noticeably, that acceptable safety margins exists [2]. The SAR shall be seen as the survey of information concerning the safety of the specific NPP and includes the demonstration of acceptability of the NPP against the rules and related criteria established for the Country. The Safety Analysis is part of the licensing process and is documented in the Final Safety Analysis Report (FSAR) [3].

In all countries using nuclear energy for power production, safety analysis has to be performed and documented in the FSAR, as well as all the important characteristics of the plant, which is reviewed and/or approved by the national regulator. The FSAR should have a predefined structure and content and approved procedures and methodologies, brought out by the regulator by requirements in the form of guides, rules and recommendations [3].

The accident analysis is an important part of a NRST and should be performed and documented on Chapter 15 – Transient and Accident Analysis, of a FSAR. The Chapter 15 includes the analysis of the following event categories [1]:

1. Increase in heat removal by the secondary side;
2. Decrease in heat removal by the secondary side;
3. Decrease in flow rate in the reactor coolant system;
4. Increase in flow rate in the reactor coolant system;
5. Anomalies in distributions of reactivity and power;
6. Increase in reactor coolant inventory;
7. Decrease in reactor coolant inventory;
8. Radioactive release from a subsystem or component.

Each category of events is typically subdivided into several events that are more specific. Events which are expected to occur during the plant lifetime are called anticipated operational occurrences (anticipated transients). They are also analyzed under the assumption of a complete failure of the fast reactor shutdown system, or Anticipated Transient Without Scram (ATWS).

There is variety of codes that allows predicting the response of the NPP during accident conditions. In the last decades, several complex system codes have been developed to simulate the main thermal-hydraulic phenomena that occurs during transient conditions. Originally, system thermal-hydraulic codes were used to support the design of safety systems, but since the publication of the 10 CFR 50.46, in 1978, they started to be applied widely in the licensing process. In parallel, especially after the TMI-2 accident, several "realistic" or so-called "Best-Estimate" (BE) codes started being developed in order to switch from the previously-used conservative assumptions to more realistic description of the processes. Since then, BE system codes are used to perform safety analysis of the NPP during accident scenarios, uncertainty quantification, Probabilistic Safety Assessment (PSA), reactor design, etc. Some examples of BE codes are RELAP5, TRAC, TRACE, CATHARE, ATHLET, and others [4].

There are different options on accidents analysis area by combining the use of computer codes and input data for licensing purposes. Four options can be identified [5]:

1. Very conservative approach, shown in Appendix K of 10 Code of Federal Regulations (CFR) 50.46 (USNRC, 1974), for examination in case of Loss Of Coolant Accident (LOCA);

2. Realistic conservative approach, which is similar to the first, except for the fact that best estimate computer codes instead of conservative codes are applied;

3. Initial and boundary conditions taken as realistic considering its uncertainties. In some countries like USA this option would be to Best Estimate Plus Uncertainty (BEPU); and

4. Realistic approach considering the actual installation conditions of the operation and the use of best estimate codes.

These options are represented in the Table 1.

Table 1: Options for combination of a computer code and input data.

Option	Computer code	Availability of systems	Initial and boundary Conditions
1. Conservative	Conservative	Conservative assumptions	Conservative input data
2. Combined	Best estimate	Conservative assumptions	Conservative input data
3.Best Estimate (Best Estimate Plus Uncertainties BEPU)	Best estimate	Conservative assumptions	Realistic plus uncertainty; partly most unfavourable conditions
4. Risk informed (Extended BEPU)	Best estimate	Derived from probabilistic safety analysis	Realistic input data with uncertainties

In the last years, were performed several calculations making use of a BEPU methodology for the LOCA analysis, and most recently, for the others transients present on Chapter 15 of FSAR. However, the FSAR of a generic plant includes more eighteen chapters, totalizing nineteen. Each one relates to the others, addressing different important characteristics of the plant to insure the safety, as the location, training of the employees and meteorological aspects, for example.

Due to historical reasons, the accident analysis part of FSAR received considerable attention in the nuclear reactor safety discipline. However, a set of accidents can happen in peripheral areas or as a consequence of precursory events which can bring the NPP in conditions outside those previously considered for accident analysis. This can be easily observed by the root-causes of the major nuclear accidents, as Fukushima. Therefore, the homogenization of the FSAR topics is required, through the systematic identification of topics and their consideration for the analysis [1].

The objective of the present paper is to discuss one entire FSAR based on the BEPU methodology, through the homogenization of the analytical techniques and identification of key disciplines and key topics in the licensing process.

2 BEPU METHODOLOGY

BEPU approach is characterized by applying the best estimate code with BE initial and boundary conditions to simulate the considered event. When performing the licensing calculations, it is expected that the availability of safety and control components and systems be defined in a conservative way, including the assumption of the single failure and loss of off-site power. However, uncertainty of the best estimate calculation has to be quantified and considered when comparing the calculated results with the applicable acceptance criteria [3].

The BEPU approach has been adopted as the methodology for accident analyses covering the established spectrum of Postulated Initial Events (PIE). Procedures have been applied to identify the list of PIE and applicable acceptance criteria. Finally, the application of computational tools including nodalizations required suitable boundary and initial conditions and produced results related to the Atucha II transient scenarios originated by the PIE. The proposed BEPU approach follows current practices on deterministic accident analyses, but includes some key features to address particular needs of the application. The BEPU-flow diagram is represented in the Figure 2, where CA means Component Analysis, SA means System Analysis and RA, Radiological Consequences Analysis [6].

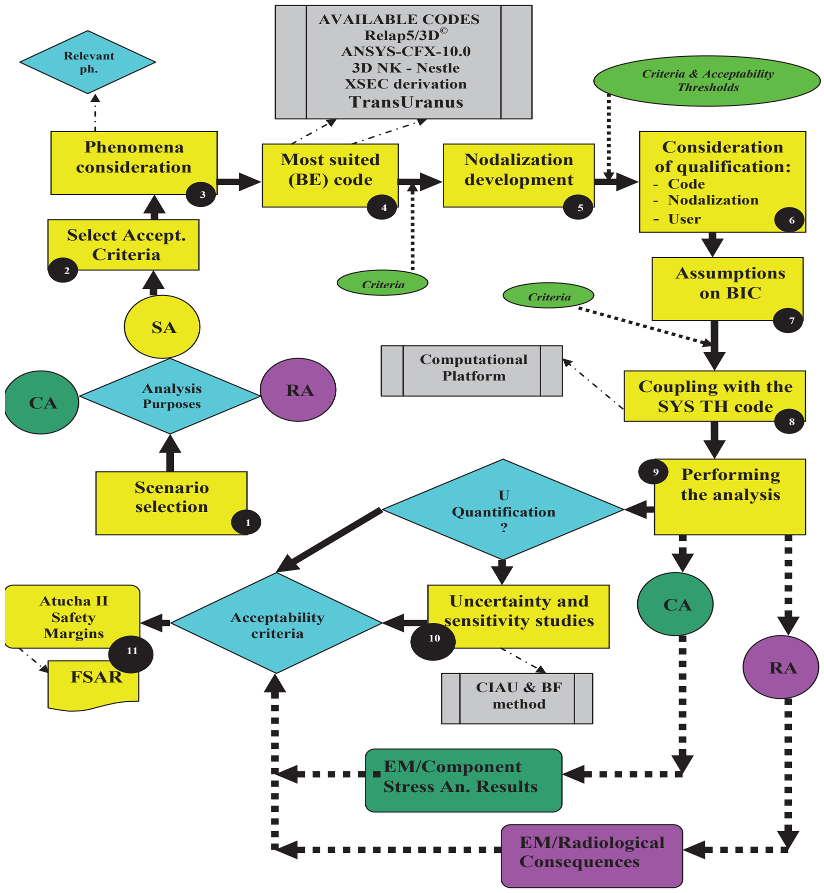

Figure 2: BEPU flow-diagram

The approach takes credit of the concept of evaluation models (EMs), and comprises three separate possible modules depending on the application purposes [6]:

• For the performance of safety system countermeasures (EM/CSA);

• For the evaluation of radiological consequences (EM/RCA);

• For the review of components structural design loadings (EM/CBA), where the acronyms CSA, RCA and CBA stand for 'Core Safety Analysis', 'Radiological Consequence Analysis' and 'Component behaviour Analysis'.

There are several methods for the BEPU application and all of them have the identification and characterization of the relevant uncertainty parameters in common as well as the quantification of the global influence of the combination of these uncertainties on calculated results [3].

BE analysis with evaluation of uncertainties is the only way to quantify the existing safety margins. Uncertainty quantification has been used mainly in two different areas, generally aiming at investigation of the effect of various input uncertainties on the results calculated with the complex thermal-hydraulic codes, and of performing uncertainty analyses for licensing purposes [7].

2.1 BEPU and Licensing

Licensing is motivated by the need to protect humans and the environment from ionizing radiation and, at the same time, sets out the basis for the design and determining the acceptability of nuclear installations, guiding the life of the NPP from the conceptual design to decommissioning. The licensing objective is to demonstrate the capability of safety systems to maintain fundamental safety functions and it is supported by the IAEA General Nuclear Safety Objective, which is "to protect individuals, society and the environment from harm by establishing and maintaining in nuclear installations effective defenses against radiological hazards" [8].

Nowadays, in most countries the national regulators allow the use of best-estimate codes to be applied in the licensing process. Some examples of such countries are United States (US), France, Brazil and Argentina. Initially BEPU methods were applied mainly to Large Break Loss–of-Coolant Accident (LB-LOCA). However, later these methods start also to be used for analysis of Small Break LOCA (SB-LOCA), as well as for operational transients [9].

The US Westinghouse developed and licensed a best-estimate LB-LOCA methodology for three- and four-loop designs in 1996 and, later, extended the methodology to two-loop upper plenum injection plants [10].

In France, an accident analysis method was developed based on the use of realistic computer codes called Deterministic Realistic Method (DRM), found on qualification of the calculation uncertainty, which is taken into account deterministically when the results are compared to the acceptance criteria. The DRM was first applied in 1997 to LB-LOCA for a French three-loop pressurized water reactor [11].

In Brazil, the uncertainty analysis of SB-LOCA scenario in Angra-1 NPP was an exercise for the application of an uncertainty methodology. For Angra-2, a LB-LOCA analysis was performed and the treatment of uncertainties was carried out separately in three basic categories: code uncertainty (statistical quantification of the difference between calculated and measured parameters); plant parameters uncertainties (statistical variations); and fuel uncertainty parameters (statistical variations) [12] [13].

For the licensing process of the Atucha-II NPP in Argentina, the BEPU approach was selected and applied to the Chapter 15 of FSAR "Transient and Accident Analysis" in 2008 [6]. Thus, the BEPU methodology has been adopted covering the established spectrum of PIE, wherein procedures have been applied to identify the list of PIE and applicable acceptance criteria, and the application of computational tools produced results related to the Atucha II transient scenarios originated by the PIE [6].

Considering all the successful applications of the BEPU methodology for licensing purposes, it is therefore proposed therefore to extend its range of use to each area of FSAR, principally the chapters and the topics where the analytical techniques are needed.

3 BEPU-FSAR

BEPU approach includes the use of the most recent analytical techniques, the existence of validated computational tools, and the characterization of expected errors or the evaluation of uncertainty affecting the results of application.

To perform an entire FSAR bases on BEPU, or so-called "BEPU-FSAR", a homogenization of the analyses is proposed, including calculation processes, that are not limited to accident analysis but cover selected topics that are connected with the design and the operation of the NPP.

Key disciplines and key topics have been defined by areas of knowledge based on the FSAR chapters, the Regulatory Guide divisions, and the IAEA Safety Standard Series. The list of key disciplines and related key topics that were derived from the FSAR content is provided in Table 2.

Table 2: Key disciplines and Key topics in the licensing process of a NPP.

Key Disciplines	Key Topics
Legal Licensing Structure	FSAR writing and assessment Knowledge of, IAEA, US NRC, ASME, ANS, IEEE frameworks of requirements Defense in Depth application
Siting & Environmental	Climatology Seismology Earthquake and Tsunami Geology including stability of slopes Hydrology and Floods Meteorology Catastrophic (including natural and man-originated) events Atmospheric diffusion Loadings Population Distribution
Mechanical Engineering: Design of Structures, Systems and Components	Structural Mechanics Thermodynamic Machinery Control Rod mechanisms
Nuclear Fuel	Nuclear Fuel performance Fuel movement
Materials	Corrosion Mechanical resistance Radiation damage Creep Analysis Fatigue Analysis Erosion
Neutron Physics	Cross Section Derivation Monte Carlo

Chemical Engineering	Chemistry of nuclear fluids Chemistry of water Metal Steam production Zircaloy reactions Boron control
Electronic Engineering	Instrumentation and Control (l & C) Nuclear Instrumentation (in-core) Ex-core instrumentation Digital systems Analogical systems
Electrical Engineering	Transformers Alternators
Civil Engineering	Containment Foundation
Deterministic Safety Analysis	Accident Analysis Computational tools Uncertainty Analysis Severe Accident Consequences
Probabilistic Safety Analysis	Reliability Cost-Benefit Analysis Severe Accident Probability Probability of Meteorite
Human Factors Engineering	Man-Machine interface Simulator Human failure
Occupational Health and Radioprotection	Radiological Protection Accessibility to remote Radioactive Zones Shielding
Physical Security	Fire protection Hazards
Plant Operation and Procedures	Emergency Preparedness Emergency Operating Procedures Plant procedures for normal operation In-service Inspection Administrative Procedures Inspections, Tests, Analyses and Acceptance Criteria
Quality Assurance[1]	Management Procedures Standards
Computational Science[1]	Information Technology Software

[1] Cross Cutting Disciplines, which are presented throughout the FSAR.

4 CONCLUSION

The description of BEPU methodology in nuclear reactor safety and licensing process involves a wide variety of concepts and technological areas. Notwithstanding the considerable growth of BEPU applications over last decades, there is still a margin for further improvements.

The application of BEPU methods were carried out in several countries; however, the framework to introduce the BE analysis, as well as BEPU methodology, into the licensing process is still an open issue. Notwithstanding, over the years, more and more applications have proven to be satisfactory, since the BE analysis with the evaluation of uncertainties is the only way to quantify existing safety margins, even uncertainty evaluations being considered as a need to improve practicability of methods.

Some problems can be associated and addressed within the historical licensing process as high cost, reluctance to innovation and lack of homogeneity. Nowadays, the licensing process is based on a non-homogeneous interpretation of licensing requirements, engaging different groups of experts without coordination, resulting in a lack of homogeneity. Assembling the top level competence in relation to each of the listed topics and disciplines, on the one hand, there is an obligation and importance to demonstrate the safety of any nuclear installation and, on the other hand, there is difficulty to address the safety in a holistic way. Therefore, the idea of a BEPU-FSAR proposal is to fill this lack by providing the homogenization of analytical techniques and thus to increase the safety of the plant.

The idea of a BEPU-FSAR is connected with the use of BEPU for qualified computational tools and methods as well as for the analytical techniques that are presented in FSAR. The qualified analytical techniques shall be adopted together with the latest qualified findings from technology research, thus homogenizing what is in the concern to the safety of nuclear power plants: the analysis including calculation process, but not only limited to accident analysis, but all the analysis that encompass any FSAR topic. For this purpose, it is necessary to create a connection between safety analysis and the hardware of the NPP, starting from the connections between the chapters and the disciplines.

In the table with the key topics and disciplines that are dedicated to the licensing process, one can recognize areas which need specific expertise knowledge (Climatology and Instrumentation and Control, e.g.). The future steps of this work will concentrate on propagation of this expertise into the remaining areas, thus building a BEPU-FSAR in the most gradual and integrated manner, adding new knowledge and improving plant safety.

One can conclude from the finalized BEPU applications that this methodology is feasible, which encourage to extended the use for other areas and demonstrate the industrial worth and interest. Another point that should be emphasized is the main obstacle in the spread of BEPU, which consists, basically, in the needed of deep expertise, numbers of wide databases and sophistication of computational tools. A lack of expertise in many areas of a FSAR and consequently the nuclear reactor safety technology, results in a simplification of how the safety analysis is conducted nowadays.

The future steps of this work will mainly be focused on the propagation of this expertise into the remaining technical areas of FSAR, adding new knowledge and therefore creating coherent and rigorous background of the BEPU-FSAR methodology.

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Possible accident scenarios related to the Spent Fuel Pool operating events

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ABSTRACT

After the accident in Fukushima Daiichi NPP, the Nuclear Energy Agency Committee on the Safety of Nuclear Installations (OECD/NEA CSNI) initiated activities to address some technical issues. As a one of results, the Status report on Spent Fuel Pools (SFPs) under Loss of Cooling Accident Conditions was created. To give a valuable contribution to the post-Fukushima accident decision making process, brief summaries were produced on:

- The status of SFP accident and mitigation strategies;
- Assessment of current experimental and analytical knowledge about loss of cooling accidents in SFPs and their associated mitigation strategies;
- The strengths and weaknesses of analytical methods used in codes to predict SFP accident evolution and assess the efficiency of different cooling mechanisms for mitigation of such accidents;
- Identification of additional research activities required to address gaps in the understanding of relevant phenomenological processes, where analytical tool deficiencies exist, and to reduce the uncertainties in this understanding.

The final report was approved by OECD/NEA CSNI in December 2014 with the reference NEA/CSNI/R(2015)2 and it is available for download on the OECD website http://www.oecd-nea.org/nsd/docs/2015/csni-r2015-2.pdf.

Joint Research Centre of European Commission took the leading role in creation of the chapter about possible accident scenarios, past accidents and precursor events. Evaluations of past events where SFP cooling has been lost show that malfunctions of the SFP cooling system are in most cases caused by inoperable cooling pumps. The other important causes are inadvertent diversion of coolant flow and Loss of ultimate heat sink.

This paper is providing short general report overview and more details about JRC contribution.

Keywords: *Nuclear Safety, Spent Fuel Pool, Accident scenarios*

1. INTRODUCTION

Spent fuel pools (SFPs) are large accident-hardened structures that are used to temporarily store irradiated nuclear fuel. Due to the robustness of the structures, SFP severe accidents have long been regarded as highly improbable events, where there would be more than adequate time for corrective operator action. The Fukushima Daiichi nuclear accident that followed after the Great East Japan Earthquake on 11 March, 2011, has renewed international interest in the safety of spent nuclear fuel stored in SFPs under prolonged loss of cooling conditions.

Following the 2011 accident at the Fukushima Daiichi NPP, the Nuclear Energy Agency Committee on the Safety of Nuclear Installations (NEA CSNI) decided to launch several high-priority activities to address certain technical issues. Among other things, it was decided to prepare a Status Report on Spent Fuel Pools (SFPs) under loss of cooling accident conditions [1]. This activity was proposed jointly by the CSNI Working Group on Analysis and Management of Accidents (WGAMA) and the Working Group on Fuel Safety (WGFS). The main objectives, as defined by these working groups, were to:

- Produce a brief summary of the status of SFP accident and mitigation strategies, to better contribute to the post-Fukushima accident decision making process;
- Provide a brief assessment of current experimental and analytical knowledge about loss of cooling accidents in SFPs and their associated mitigation strategies;
- Briefly describe the strengths and weaknesses of analytical methods used in codes to predict SFP accident evolution and assess the efficiency of different cooling mechanisms for mitigation of such accidents;
- Identify and list additional research activities required to address gaps in the understanding of relevant phenomenological processes, to identify where analytical tool deficiencies exist, and to reduce the uncertainties in this understanding.

The proposed activity was agreed and approved by CSNI in December 2012, and the first of four meetings of the appointed writing group was held in March 2013. The writing group consisted of members of the WGAMA and the WGFS, representing the European Commission and the following countries: Belgium, Canada, Czech Republic, France, Germany, Hungary, Italy, Japan, Korea, Spain, Sweden, Switzerland and the USA. Status Report on SFP mostly covers the information provided by these countries.

The final report was approved by OECD/NEA CSNI in December 2014 with the reference NEA/CSNI/R(2015)2 and it is available for download on the OECD website http://www.oecd-nea.org/nsd/docs/2015/csni-r2015-2.pdf.

Joint Research Centre of European Commission took the leading role in creation of the chapter about possible accident scenarios, past accidents and precursor events. Evaluations of past events where SFP cooling has been lost show that malfunctions of the SFP cooling system are in most cases caused by inoperable cooling pumps. The other important causes are inadvertent diversion of coolant flow and Loss of ultimate heat sink.

Special attention is given to three fuel events that happen outside the reactor. Events from Paks NPP and Bruce-A NPP resulted in fuel damage were also outside SFP at the time of event, but those events are interesting because of similarity with conditions in SFP. Third event is about the most serious SFP scenario from Fukushima Daiichi nuclear accident.

2. STATUS REPORT ON SFP UNDER LOSS OF COOLING AND LOSS OF COOLANT ACCIDENT CONDITIONS

Status Report on SFP is intended to summarize current understanding of the behaviour of SFPs in loss of cooling and loss of coolant accident conditions. Past accidents and precursor events are reviewed, in particular the behaviour of the spent fuel facilities during the Fukushima Daiichi accident. Important aspects of the accidents and involved phenomena are addressed, such as the thermal-hydraulic behaviour of the pool, the issue of criticality, the accident progression under partial or complete loss of coolant, the hydrogen risk, the fission product release, etc. The report provides a brief assessment of current experimental knowledge about loss of cooling and loss of coolant accidents. It also presents state-of-the-art computer codes used for analyses of SFP accidents, and discusses strengths and weaknesses of models and methods used in these codes. The probability of SFP accidents and assessments of off-site health effects and contamination consequences of SFP accidents are, however, beyond the scope of this document.

The fuel residing in At-Reactor (AR) SFPs is usually characterized by higher decay power than fuel stored in Away-From-Reactor (AFR) pools. Since the progression rate and severity of a loss of cooling/coolant accident correlates with the power of the stored fuel assemblies (FAs), the most challenging accident scenarios are expected in AR storage pools. For this reason, the report focuses on AR SFPs in light-water reactor (LWR) and Canada Deuterium Uranium (CANDU) reactor nuclear power plants.

The report is also aimed to identify areas that need additional knowledge and to identify potential improvements in computational models and tools for better predictions of SFP accident progression and time margins to significant radiological releases. Better understanding of the SFP accidents phenomenology and coolability mechanisms is needed for reliable estimation of accident progression and radiological consequence.

3. SPENT FUEL POOLS DESIGN

A typical design of the SFP in pressurized water reactors (PWRs) is shown in Figure 1, whereas Figure 2 shows a typical pool design for boiling water reactors (BWRs). The pools are constructed of reinforced concrete with a stainless steel liner to prevent leakage and maintain water quality. The pools are envisaged to withstand design-basis seismic events.

Each plant has a source of high purity water to fill the SFP, referred to in nuclear power plants as make-up. The preferred sources are usually the refuelling water storage tank for PWRs and the condensate storage tank for BWRs. The normal make-up is through a connection from the water source to the suction of the SFP cooling system pumps, and local valve operations are needed to initiate SFP make-up. The make-up rates among plants have a wide range. Plants also have alternate methods to provide make-up, if normal make-up is unavailable, and may include the service water system and the fire water system.

For BWRs, the SFP is generally located within the reactor building, but outside of the primary containment. Also for PWRs, the SFP is usually located outside the containment, but adjacent to it in a separate fuel handling building or within the auxiliary building. Exceptions are the Russian VVER-1000 design, the German Kraftwerk Union (KWU) KONVOI or pre-KONVOI PWR design, and the AREVA EPR design, where the SFP is located inside the containment.

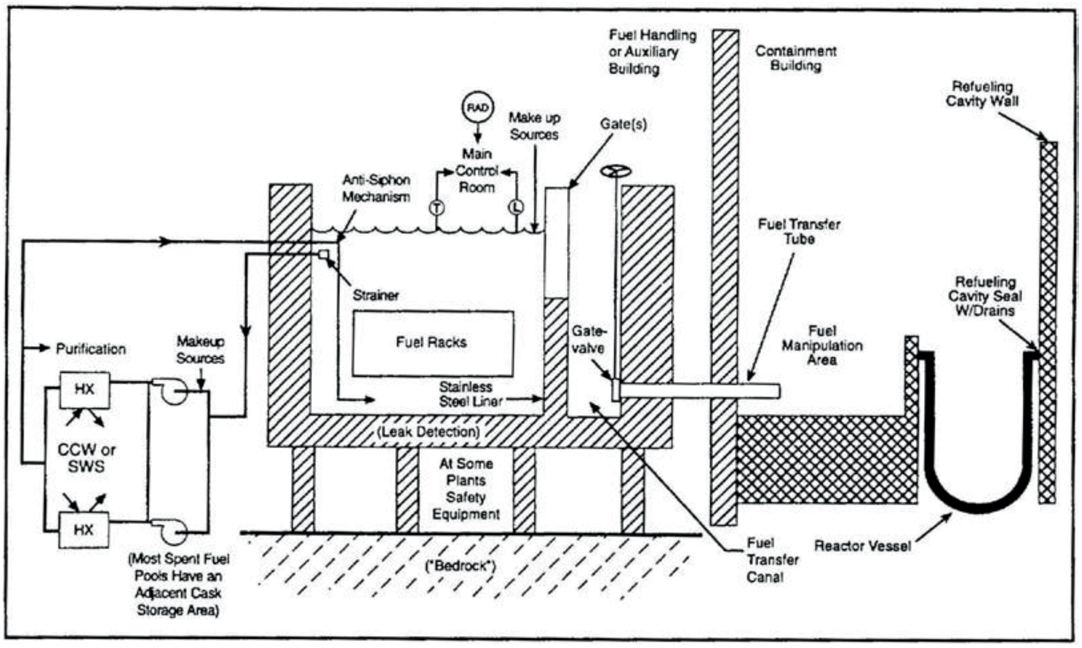

Figure 1: Generic SFP design for PWRs

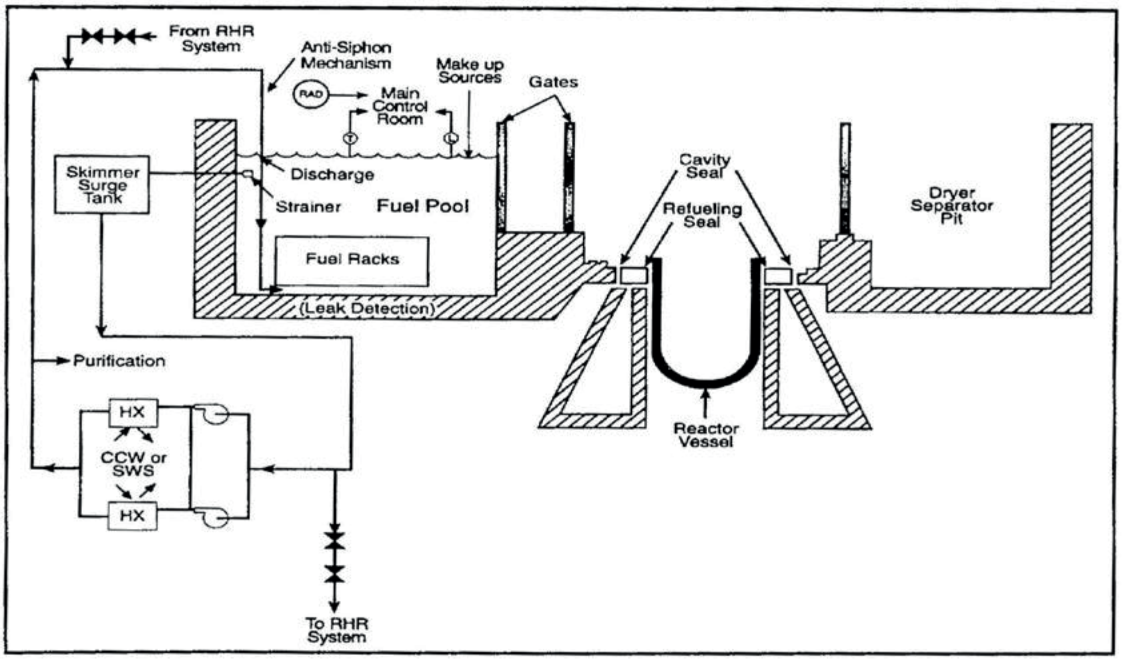

Figure 2: Generic SFP design for BWRs

Typically, SFPs in light water reactors are about 12 m deep and vary in width and length. The fuel is stored in stainless steel (SS) racks that are submerged with approximately 7 m of water above the top of the stored fuel. The water in the SFP of a BWR is demineralized water, whereas PWRs and VVERs use borated water. In addition to cooling, the SFP water inventory provides radiological shielding for personnel in the fuel pool area and adjacent areas. Shielding is also provided by the thick concrete walls of the SFP. Each plant generally has technical specification requirements for water temperature and level, and for the margin to criticality for the fuel stored in the SFP.

4. SPENT FUEL POOL OPERATING EVENTS

Status Report on SFP describes selected events where the cooling of pools or part of their water inventory is lost are described. We can say that no major events occurred, i.e., events with significant consequences related to SFP loss-of-cooling have not happened. However, the described events illustrate different scenarios in which SFP cooling or water level was affected, and in escalated scenarios, it could result in fuel damage. Therefore, the described events are selected such that difference in their nature is emphasized rather than level of significance.

4.1. Fuel related events

In 2009, the European Clearinghouse on NPP Operational Experience Feedback performed a study on events reported in the IAEA International Reporting System for Operating Experience [2] related to nuclear fuel. The SFP related events causes relate to human errors and, to a lesser extent, design deficiencies. Some of these deficiencies originated from the initial design of the SFP, but in some occasions, the design deficiencies derived from a change in the characteristics of the stored fuel (e.g. higher enrichment and burn-up, re-racking).

In the study [2], 28 events have been identified to be related to fuel integrity in storage facilities, mainly in the SFP. According to their nature, they have been classified in events related to Loss of cooling; Loss of margin to criticality; Fuel integrity, and Radiological impact.

Events with loss of cooling (including SFP water level drops) (Table 1) are relatively the most frequent ones, and are caused or related to configuration control of the SFP cooling system (interconnections and manual operation), leakages in the liner, and the lack of monitoring systems, so detection of the problem could be delayed. This group of events includes all the events that have originated from any kind of loss of cooling capacity, either cooling system equipment malfunction or loss of coolant (significant drop of the fuel pool water level). It should be remarked that these events are slow, and there is usually enough time to restore the cooling function.

Table 1: Some past events related to loss of SFP cooling

NPP, type	Event	Cause	Consequences
Kori 1, Korea, PWR	Loss of shutdown cooling due to station blackout during refuelling outage	Loss of off-site power resulted in loss of SFP cooling.	SFP temperature increased slightly.
Catawba 1, USA PWR	Dual unit loss of off-site power resulting from inadequate relay modification	Loss of off-site power resulted in loss of SFP cooling.	Short loss of SFP cooling capability.
Forsmark 3, Sweden, BWR	Emergency diesel generators failed to start after undetected loss of two phases on 400 kV incoming off-site supply	Loss of two phases on 400 kV off-site power resulted in loss of SFP cooling.	Loss of SFP cooling capability with no increase in SFP temperature.
Almaraz-2, Spain, PWR	Irradiated fuel, both in the vessel and the SFP, without forced cooling during refuelling outage	Loss of component cooling water system capability.	SFP cooling was lost for 7 hours and temperature increased by 12 K.
Belleville 2, France, PWR	Disruption of the SFP cooling	Fire in the pump room of one of the two SFP cooling system trains, while the other was out of order.	Simultaneous failure of SFP cooling trains, for 6 hours and then for 15 hours.
SONGS 2, USA,	Inoperable SFP cooling pumps results in loss of safety function	One SFP cooling pump out of service and the other	SFP temperature increased slightly.

NPP, type	Event	Cause	Consequences
PWR		tripped on overcurrent signal.	
SONGS 2, USA, PWR	Both trains of SFP cooling inoperable results in a loss of safety function	Salt water cooling system low flow affected component cooling water system.	SFP temperature increased slightly.
Khmelnitski 1, Ukraine, VVER-1000	Drop in the level of the wet refuelling pool and flooding of reactor building areas	Inappropriate valve maintenance activities caused leak through open joint on one SFP valve.	Significant drop in the level of the wet refuelling pool (1.9 m).
Borssele, The Netherlands, PWR	Insufficient testing of functionality of siphon breaker valve at spent fuel basin	Lack in inspection program.	None*.
Cattenom 2 and 3, France, PWR	"Siphon breaker" missing on SFP cooling systems pipes	Injection pipe could extract the water from the pool through a siphon effect.	None*

* Design/Inspections changes implemented later to prevent potential uncontrolled drainage of SFP.

Events with radiological impact are also relatively frequent, due to the numerous activities and circumstances that could lead to radiological exposure of the workers. It should be noted that radiological exposure in this context does not mean fuel overheating and radioactive release from damaged fuel. The main cause of these events is activities in the SFP building, not carefully planned and analysed to identify all possible sources of radiological exposure (risk analysis).

Events with a reduction in the margin to criticality and events with fuel integrity concerns are rare, and their specific causes are erroneous concentrations of boron, incidents with inadequate neutron absorbers in structures, errors in the calculation of the margin to criticality, or failure in the control and monitoring of the SFP water chemistry. Restoring the boron concentration, repairing the neutron absorber in the structures, and correcting the calculations are sufficient to recover the criticality margin.

4.2. Fukushima Daiichi accident

The Great East Japan Earthquake took place on 11 March 2011, and the resulting tsunami caused loss of emergency diesel powered AC generators and produced conditions known as station blackout (SBO) at the Fukushima Daiichi Nuclear Power Station (NPS). As results of the SBO, the emergency cooling systems and water supply systems failed and the three Units 1 to 3 subsequently suffered severe core damage.

Following the loss of all AC power at Units 1 to 5, the SFP cooling flow was lost in the Unit 1 to Unit 4 SFPs, while the Unit 6 air-cooled emergency diesel generator survived the tsunami and was used to maintain cooling and water supply for the Unit 5 and Unit 6 SFPs [3]. With no pool cooling to remove decay heat at the Unit 1 to Unit 4 SFPs, emergency water injection was conducted by using a helicopter, a concrete pump truck, a fire truck, and the make-up water condensate or SFP cooling and cleanup system line. Eventually, pool water cooling by the alternative cooling system was started at Unit 1 to Unit 6 SFPs, and the water temperature has then been maintained below 40 °C, which is a typical SFP temperature. Video inspections reveal that the fuel racks appear to be intact, and water analyses indicate that fuel damage in the pools is unlikely. Hence, there is no evidence that the fuel in the pools was damaged.

The most serious scenario happened in unit 4 SFP. Around 15:35 on 11 March, the SFP 4 lost all AC power when EDGs stopped functioning as the seawater pumps and power panels were flooded by the tsunami. The cooling and water supply for the SFP likewise failed. Around 6:00 on 15 March, the reactor building was damaged by a hydrogen explosion, and a large amount of debris dropped into the pool. On 16 March, a helicopter flew close to the operating floor of Unit 4, at which time the water surface of the pool could be seen, but no exposed fuel was observed.

According to the original schedule, the drain of the reactor vessel and reactor well should have been completed by 7 March 2011, but the operations were delayed and the reactor well was still filled with water on March 11 [4] [5]. This situation played an important role during the course of the accident and significantly slowed the decrease of the SFP 4 water level: the water-tightness of the pool gate was lost due to the pressure from the reactor well side as the SFP water level became low. It induced a water inflow from the reactor well to the pool.

At first, the water inflow from the reactor well was not considered and it was estimated that the fuel would be uncovered by late March. Therefore, from 20 March, water was added via helicopter, fire truck, and concrete pump truck. Eventually, the water level reduced to 1.5 m above the top of the fuel racks as the amount of evaporation was larger than the water injection, including water inflow from the reactor well, until around 20 April, as shown in Figure 3[3].

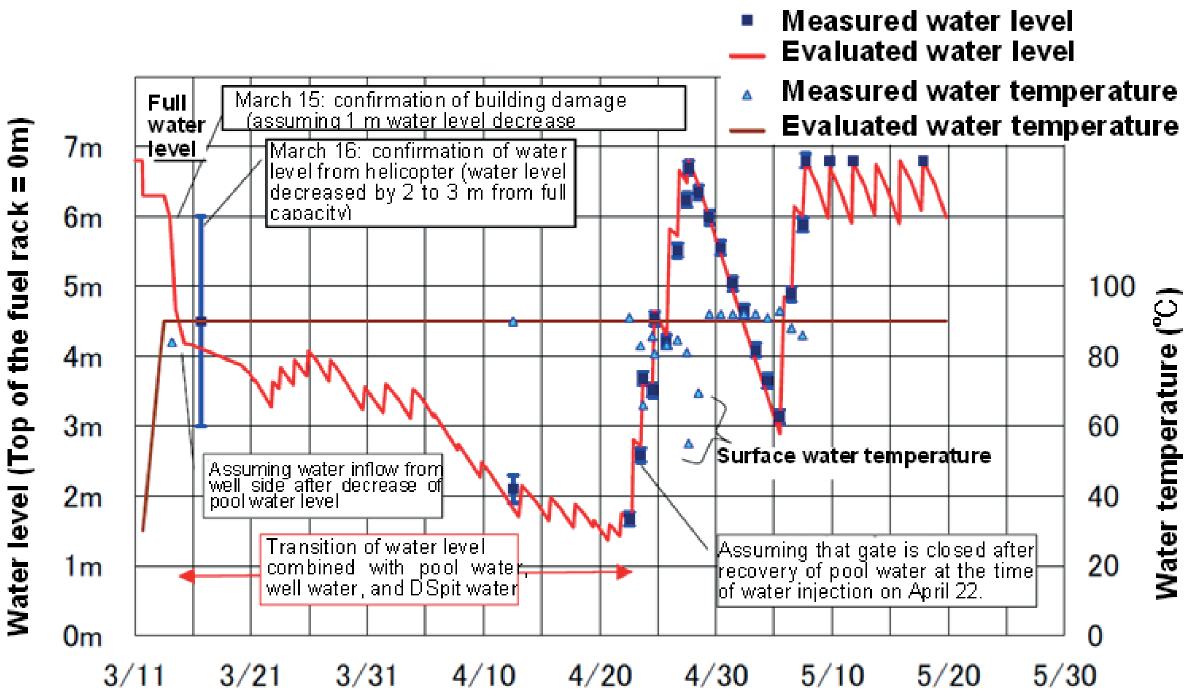

Figure 3. Evaluated water level and water temperature in the SFP 4 by TEPCO

The assumptions made in the estimation are as follows:

- The water level is assumed to have been reduced by 1.5 m as a result of sloshing by the earthquake and the explosion.

- Inflow from the reactor well occurred before 22 April. The water level was calculated by considering the water in the pool and the reactor well and dryer and separator pit collectively. After 22 April, the pool gate was closed, and no inflow from the reactor well was considered.

Later, intensive water injection conducted between 22 and 27 April succeeded in recovering the water level to the full capacity. The water injection was then suspended until 5 May to study the trend of the reducing water level. Subsequently, the water level recovered again to full capacity by intensive water injection, and then the water level is considered to be maintained at near full capacity by repeated reduction and recover due to evaporation and water injection.

On 31 July, pool water cooling by the alternative cooling system was started. The pool water temperature was around 75 °C when the cooling was started and reached a steady condition on 3 August when the water temperature stabilized at about 40 °C.

4.3. The Bruce-A Unit 4 fuel transfer incident

On 1983 November, a CANDU 37-element fuel bundle was overheated in steam-air environment in the fuel transfer mechanism of Bruce-A Unit 4 reactor, Canada [6]. This event differs from the case of loss of coolant in CANDU SFP because there were no neighbouring fuel bundles, and the decay heat was higher than expected for a bundle in a CANDU SFP. Also, the temperatures at the end-plates were lower than the temperatures in the rest of the bundle.

Bundle G70551W was a standard 37-element Bruce Nuclear Generating Station bundle irradiated to an average burnup of 5.9 MWd/kgU at an average outer element power of 41 kW/m. It was discharged from the reactor as part of normal scheduled fuelling. The bundle was kept in the fuelling machine under heavy water cooling for about 2 hours before discharge to the fuel transfer chamber. Because of problems with other equipment, the bundle was left on the cradle in the flooded fuel transfer chamber for many hours. The vent valve was left closed, so injection of air from the instrumentation bubbler formed an air space in the top of the chamber, which slowly uncovered the bundle. After the bundle was uncovered, local boiling would also have occurred, which could have displaced further water from the chamber. The bundle was probably partly uncovered for a total of 5 hours; the vent valve was opened once during the incident, which would have re-flooded the bundle. The incident was terminated when the fuel port seal was opened, submerging the bundle and releasing airborne activity from the chamber.

Sheath oxidation was more rapid at the bearing pad braze heat affected zones. At the central bearing pad plane of the bundle, the sheaths of 22 of the 37 elements (in the upper and central parts of the bundle) were severely oxidized, and in many cases were ballooned and distorted. Hydriding was found in the unoxidized part of some sheaths. The inner elements had degraded into rubble. None of the sheath oxides had a columnar structure, so the bundle temperatures probably did not exceed 1050°C. Melting of a few of the beryllium brazes indicated temperatures above 970°C. Oxidation of the sheath inner walls and of the UO_2 was minimal, so the air was probably rapidly deoxygenated by the Zircaloy oxidation. Also, fuel-sheath interaction was not observed.

4.4. The Paks cleaning tank incident

On April 10, 2003, during refuelling outage in the Paks unit 2, Hungary, 30 spent fuel assemblies were being cleaned in a special container in the fuel manipulation pit of the SFP. After completing the cleaning process, the fuel was left in the container with reduced cooling, which resulted later in severe cladding oxidation and fuel damage [7][8].

The cleaning system consisted of a container installed in a pit for fuel manipulations connected via a lock to the SFP, interconnecting lines, heat exchangers and filter equipment (Figure 4). This technical system formed an internal closed circuit almost completely submerged into water, except for the heat exchangers and filters that were located on the reactor desk or beside it. The container received 30 assemblies for cleaning at a time, and the cleaning process was performed by circulation for about 35-40 hours. During the annual outage of Paks unit 2, altogether 210 FAs, i.e. assemblies for 7 containers, were scheduled to be cleaned.

The cleaning programme for the sixth batch of FAs loaded into the cleaning tank was completed by 16:55 on April 10. The fuel was not removed from the cleaning tank immediately, since the crane was busy with other tasks. The coolant was circulated by a submergible pump with much lower mass flow rate than used in the cleaning process (Figure 4). Contracted specialists continuously maintained the cooling of the cleaning tank at 37 °C. At 21:53, activity was detected by the krypton measurement system installed in the cleaning circuit, and at the same time, the 'alarm' level was reached by the noble gas activity concentration monitors in the reactor hall, and then the operational dosimetry systems installed in the ventilation stack indicated abrupt increase of noble gas activity (max. 0.2×10^{13} Bq/10 min). The plant supervisor ordered to terminate the work carried out in the reactor building and to leave the area. An extraordinary maintenance committee was called, in order to evaluate the event and to take necessary actions. As highest priority, it was decided to open the cleaning tank, to carry out visual inspection, and if possible, to separate the nonhermetic FA and also to analyse the water quality.

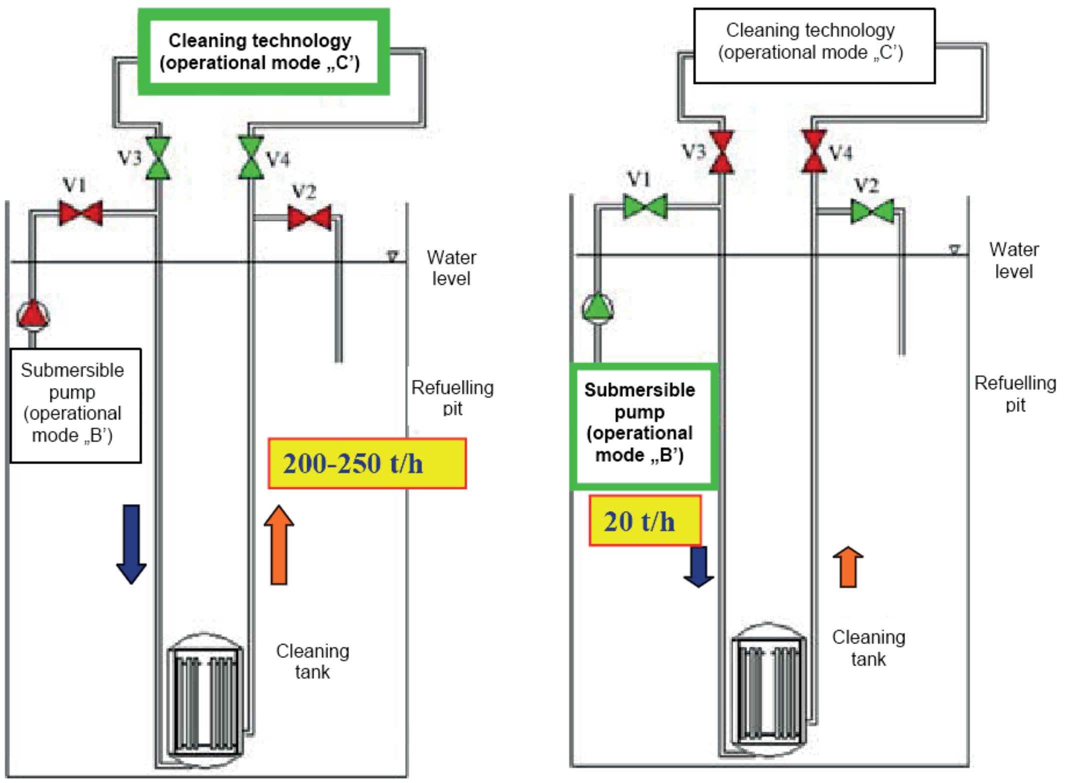

Figure 4: Arrangement of the cleaning tank cooling during cleaning (left) and post-cleaning (right) [7].

The cleaning tank head was unlocked by a contractor at 02:15, April 11. Immediately after this, an activity increase was observed in the dosimetry control and monitoring system) and, at the same time, the water level in the SFP lowered by approximately 7 cm. The first results of water chemistry analysis identified the fission products ^{134}Cs, ^{137}Cs, ^{131}I, ^{132}I, ^{133}Xe and ^{85}Kr in the samples at activity level of 10^4-10^7 Bq/kg. Activity of less volatile species was also detected in the water samples [9]. During the cleaning tank head removal operation, one of the three cables of the lifting tackle broke, thus the head removal failed. The head was lifted on April 16, 2003, and the inspection was performed with video camera. The damage of the fuel assemblies was seen to be more severe than assumed before.

Later inspection has revealed that, due to the special design of the cleaning tank and the characteristics of the fuel assemblies, the cooling by the submersible pump of lower mass flow was insufficient. The low flow rate pump was not capable of removing the decay heat (241 kW) due to a by-pass effect. The temperature stratification blocked the flow, and therefore, the coolant temperature reached saturation temperature in the upper part of the cleaning tank. Then, the steam-formation pushed the main volume of the coolant out of the cleaning tank vessel. This way, the FAs were left without proper cooling for hours and heated up to above 1000 °C, which resulted in severe damage and oxidation of the FAs. Oxidation in high temperature steam and hydrogen uptake resulted in embrittlement of the fuel assembly shrouds and the fuel rod cladding. When opening the cleaning tank, the injection of cold coolant caused the breaking up of the embrittled shrouds and fuel rod cladding.

Video examination indicated that most of the fuel assemblies suffered damage. Brittle failure and fragmentation of FAs were observed. Above the upper plate, several assembly heads were broken, standing in inclined position. One assembly header was found far from its original place. Many FAs were broken and fragmented also below the upper plate, and some assemblies were fractured in their entirety. Fuel rod fragments and shroud pieces accumulated on the lower plate between the FAs. Many fuel rod pieces and fragments of assembly shroud were dispersed within the tank. Some fuel pellets fell out of fuel rods, their form remained mainly intact. Heavy oxidation of the zirconium components was identified. Less oxidation was found in the periphery than in the centre, and the bottom part of the fuel remain intact. The radioactive noble gases that escaped from the damaged FAs were released into the environment through the reactor hall stack with negligible impact on the environment. Most of the non-gaseous fission products were trapped by the large water mass of the pool, and removed by the water purification system [9].

The OECD-IAEA Paks Fuel Project was performed to support the understanding of fuel behaviour in accident conditions on the basis of analyses of the Paks incident. Computer simulations of the most relevant aspects of the event and comparisons of the calculated results with the available information were carried out between 2006 and 2007. The numerical analyses improved the understanding of the Paks incident and helped to make more precise some parameters of the incident, such as:

- the by-pass flow at low flow rate amounted to 75-90 % of the inlet flow rate, which led to the formation of a steam volume;

- the maximum temperature in the tank was between 1200 and 1400 ºC;

- the degree of zirconium oxidation reached 4-12 %;

- the mass of produced hydrogen was between 3 and 13 kg.

The OECD-IAEA Paks Fuel Project improved the current knowledge on fuel behaviour under accident conditions, and led to recommendations for some further actions for research in this area.

5. CONCLUSIONS

Adequate cooling of the spent fuel in the SFP can principally be lost either by malfunction of the pool cooling system or by loss of the pool water inventory. In the Status report on SFPs under Loss of Cooling Accident Conditions examples are given for both types of events. The chosen events are not the most significant ones, but they present different possible scenarios that lead or could lead to loss of spent fuel cooling.

Losing the cooling of the SFP in a nuclear power plant is an event mostly connected with inability of SFP cooling pumps to operate. This is often caused by loss of electrical supply to pumps. Usually, if the loss of off-site power occurs, manual action is needed to connect back-up electrical supply. If back-up electrical supply is unavailable, the pumps cannot operate and the temperature of the pool will consequentially start to increase. In evaluated recent events, electrical supply became available and cooling continued well before the SFP water reached the maximum allowed temperature.

The most serious SFP scenario from Fukushima Daiichi nuclear accident studies showed several latent weaknesses that could result in fuel uncovery in SFP 4 and damage. This was the main reason why interest in the safety of spent nuclear fuel stored in SFPs under prolonged loss of cooling conditions increased.

Although Paks and Bruce incidents cannot be considered as a typical SFP accident, they gave insights that can be useful for understanding phenomena related to SFP loss of cooling/coolant accidents. They also prompted research about such accidents.

An improved understanding of the phenomenology of SFP accidents and coolability mechanisms, along with a consensual view of the extent of remaining uncertainties, are indispensable for reliable estimation of accident progression and radiological consequence.

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Analysis of Manual Reactor Trip of NEK NPP in APROS Computer Code

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ABSTRACT

The Slovenian Krško Nuclear Power Plant (NEK) model was built in using APROS - Advanced PROcess Simulation environment. The basis for the this model was the RELAP5/MOD3.3 Engineering Handbook, the model was updated to the 26th cycle and also includes the upflow conversion modification.

A detailed model nodalisation was created for each system and every system was separately validated. The current model covers the primary circuit with the core kinetics model, the secondary circuit and their control systems. The steady state of the APROS NEK model already having been validated, the plan now is to validate the model for some transients and design basis accidents. In this article the plant behaviour after the manual reactor trip is analysed in detail. Two scenarios of the manual reactor trip transient are performed, where either the Main Steam Isolation Valve (MSIV) closes after 60s – case A, or remains open – case B.

After the manual reactor trip from the 100% power, the control system signal actuations and their times were followed and then the responses of different affected systems were being observed. All those recorded values were then compared with the identical transient performed on the similar NEK model with the RELAP5/MOD3.3 system code. This procedure allowed to bring the current APROS NEK model one step forward towards being assured to have accurate calculations.

Keywords: *APROS 6, NEK, reactor trip, point kinetics, steam dump, PORV*

1 INTRODUCTION

The Krško Nuclear Power Plant – NEK in Slovenia has a two loop Westinghouse PWR nuclear steam supply system with 1994 MW thermal output power. A model of the primary circuit with a reactor core and the secondary circuit was built using the Advanced PROcess Simulation environment – APROS [1].

Up to the present time the APROS model of primary and secondary circuit have been verified in the steady state. Beforehand also singular systems have been validated, as separate plant system tests have been performed comparing the APROS model response to the set of plan surveillance tests – for example steam dump (SD) control actuation, Reactor Coolant Pump (RCP) coastdown curve, pressurizer (PRZ) pressure and level control, accumulator full flow test, high/low pressure safety injection (SI) recirculation and full flow test, etc.

The aim of work presented in this paper is to perform the manual reactor trip from the 100% power, analyze the results and evaluate them by comparison to the similar transient made in the RELAP5/MOD3.3. As the long-term goal is to validate the APROS NEK model, comparing its simulation results with already existing NEK analyses performed in the past with RELAP5 code and reanalysis of on-site transients that occurred at NEK in the past.

APROS, developed by the Research Centre VTT and Fortum Engineering in Finland, is a program package that allows making the dynamical simulations for engineering purposes. The tool is suitable for modelling and simulation of the dynamics of a process plant during all phases of its life span from predesign to training and model supported operation and control, for small simple models and full scope simulators. The data used for the model specification is inserted through the graphical diagrams. With the help of a graphical interface and use of a selection of process components i.e. basic building blocks, the system can be built. These building blocks have different graphical symbols representing the tank, valve, heat structure, etc. and can then be spatially discretized into several volumes for simulation purposes [2].

The RELAP5 data of the trip were obtained from the RELAP5/MOD3.3, a transient analysis code for complex thermal-hydraulic system and it was also used for the analysis of the manual reactor trip [3]. The NEK RELAP5 model represents the updated model to the state of the 26th cycle. It is updated in accordance to Resistance Temperature Detector Bypass Elimination (RTDBE) project and therefore are also updated the systems that are affected by RTDBE: the NR temperature measurement, OTΔT (over temperature Δ temperature) and OPΔT (over power Δ temperature) protection functions, compensated low steam line pressure lead-lag time constants, and Steam Dump System, were updated and changed in accordance with RTDBE project [3].

2 APROS MODEL DESCRIPTION

The current APROS NEK model built in APROS 6.04 version currently consists of 52 diagrams describing the processes of the primary and secondary coolant and the point kinetics model of the reactor core. With the changes and updates done in NEK also the already validated APROS NEK model of the 23rd cycle was brought up-to-date to reflect those changes. Therefore the RDTBE project required changes were entered into the model according to the NEK RELAP5\MOD3.3 Post-RTDBE Nodalization Notebook [4]. Additionally the upflow conversion of the Reactor Pressure Vessel was added, the modification that was introduced to NEK in the 28th cycle. The steady state of this new updated model was used as the initial condition. And for the case of this simulation the part of the secondary system was isolated and the turbine was set as a boundary of the model.

2.1 Boundary conditions

With the intention to minimize the CPU calculation times, part of the secondary system was isolated and the boundary of the simulated system was set to be the turbine. The turbine boundary condition in APROS was simply represented by two points (Figure 1), where the boundary pressure (p = 60.5 bar) and temperature (T=275.9°C) were set.

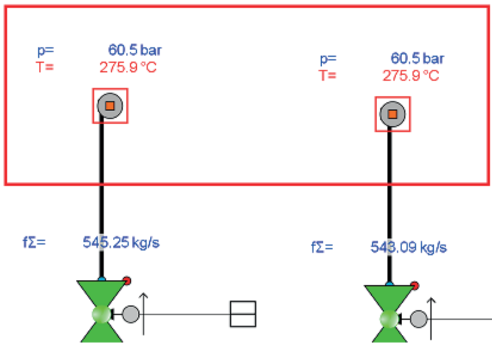


Figure 1: The boundary condition of the Reactor Trip model is set via 2 points in APROS representing the turbine.

Other main boundary conditions for the simulation of the transient were identical as in the RELAP5 model and are listed in Table 1 below.

Table 1: Main boundary conditions for the analysis [3].

Boundary condition	RELAP5 Value	APROS value	Comment
Letdown flow	3.7 kg/s	3. 7 kg/s (valve initially at 80%).	assumed constant flow until isolation
Letdown closing time for isolation	5 s	5 s	
Charging	270°C	270°C	assumed constant temperature
FW	219.35°C	219.4°C	assumed constant temperature
FW isolation valves	5 s	5 s	closing time
AFW	38°C	38°C	assumed constant temperature
SG SV	3 s	3 s	closing time
SG PORV	RELAP5 servo valve model	3 s	closing time
SG MSIV	3 s	3 s	closing time
Turbine valve	0.1 s	0.1 s	closing time
Secondary heat losses	neglected	neglected	

2.2 Initial conditions

The initial conditions of the transient simulation are the steady state values of the updated APROS NEK model of the 26th cycle including the upflow conversion of the RPV. To obtain these values, the model was left to run for several hours. And the values calculated with the APROS model were compared to NEK reference data and RELAP5 calculations, for 26th cycle in both cases. The results of the new steady state and the NEK reference data and RELAP5 found in the NEK RELAP5/MOD3.3 Post-RTDBE Steady State Qualification Report [5] were compared are presented below in Table 2.

Table 2: Comparison of NEK reference data and RELAP5 and APROS calculated values [5]. The below listed APROS values were also used as initial conditions of the transient simulation.

	Unit	NEK cycle 26	RELAP5	APROS
1. Pressure	MPa			
pressurizer		15.513	15.513	15.51
steam generator·		6.281	6.278/6.289	6.39
accumulator		4.93	4.93	4.93
2. Fluid Temperature	°C			
cold leg		285.6	286.37/286.14	286.83
hot leg		324.4	323.63/323.63	324.3

feedwater		219.45	219.55	219
3. Mass Flow	kg/s			
core		8966.9	9040.7	8867.6
cold leg		4694.7	4721.1/4720.1	4691.2/4693.5
main steam line		544.5	540.9/544.7	545.4/542.8
DC-UP bypass (2%)		187.8	184.94	200.67
DC-UH bypass (0.5%)		28.2	29	28.75
buffle-barrel flow (1.25%)		117.4	116.8	109.23
RCCA guide tubes (2%)		187.8	186.4	178.43
4. Liquid level	%			
pressurizer		55.7	55.8	54.09
SG narrow range		69.3	69.3/69.3	69.3/69.3
5. Fluid Mass	t			
primary system		-	131.27	132
SG (secondary)		47	49.1/49	
6. Power	MW	1994	1994	1994
core		1000	996.6/1003.1	997.6/1002.4
steam generator		15.513	15.513	15.51

3 GENERAL DESCRIPTION OF THE REACTOR TRIP TRANSIENT

To better validate our APROS NEK model, the same trip scenario made in RELAP5 model was also performed in APROS NEK model. The main events of the manual reactor trip form the 100% power were the following.

At manual signal for the reactor trip all rods fall into the core, the turbine trip is initiated and steam flow to the turbine stops abruptly. The loss of steam flow results in a rapid rise in secondary system temperature and pressure, therefore the SD is almost immediately initiated, it operates in the T_{avg} mode. The automatic SD system releases the excess steam generation, therefore the reactor coolant temperatures and pressure have no significant increase, while the SD and PRZ pressure control system are functioning. When the LO-T_{avg} temperature is met, it is followed by the isolation of the FW system via the Main Feedwater Isolation Valve closure, afterwards the Auxiliary Feedwater system (AFW) is started ensuring the adequate residual and decay heat removal capability. It operates in cycles keeping the steam generator (SG) level between 60% and 70%. There is no RCP trip [3]. In the later times two different scenarios are evaluated.

In case A there is a MSIV closure after 60 s, done as operator action and thus the SD valves that release the excess steam from the secondary system are cut-off and the task of lowering the main steam line pressure falls to the steam generator pressure operated relief valves (SG PORVs).

In case B the MSIV remains open and the SD valves continue in operation lowering pressure in the main steam line. And in both cases the simulation was left to run for 5000s.

The actuation of the above mentioned events is governed by the list of signals that are listed below in Table 3 and are the same in the RELAP5 and APROS NEK model.

Table 3: Delays of main protection signals and actions [3].

Event	Action	Setpoint	Delay [s]
REACTOR TRIP	manual		0
TURBINE TRIP	reactor trip		0
FW ISOLATION	LO-T_{avg} + R_x trip	295.6°C	0
Closure time for FW FIV			5
SL ISOLATION	manual		60
AF INJECTION	FW isolation signal		0
AF injection added delay			5

4 REACTOR TRIP CALCULATION IN APROS

The process diagrams in the APROS environment simulate the thermal-hydraulic variables of all the elements within one system that have been subdivided into volumes. The flow through the primary and secondary system is calculated using a six-equation flow model based on conservation equations of mass, momentum and energy for two phases. The core is modelled with the point kinetics model and the decay heat calculation was based on the ANS-79 decay curve, as in the RELAP5 model. The time step control had the maximum step size at the beginning of the transient 0.01s, 0.05s from 10 s and 0.1 s after 1000 s.

The APROS NEK model had all the values of its steady state at 100% power saved as the initial conditions, which are described in the chapters before. Then the simulation queue file was included which started the simulation with the manual reactor trip signal actuation and stopped it after 5000s, inside also the changes of the maximum step size during transient were set, and in case A it actuated the MSIV closure.

The reactor trip signal actuated the rod drop signal, the turbine trip signal and one of the FW isolation required signals (Figure 2).

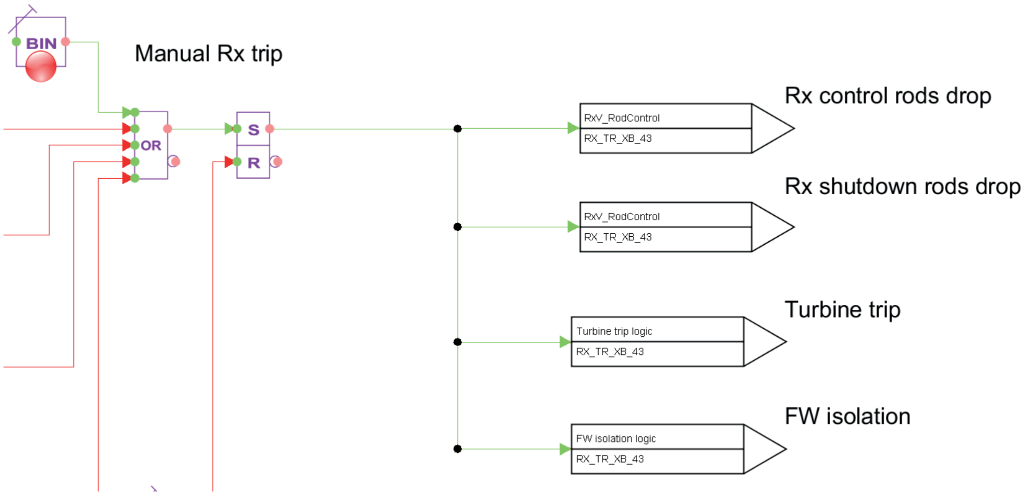


Figure 2: The Manual Reactor Trip in APROS actuation the rod drop signal, the turbine trip signal and one of the two required signals for FW isolation.

The rod drop in the core shuts-off the core kinetics calculation and the decay heat calculation is activated (Figure 3).

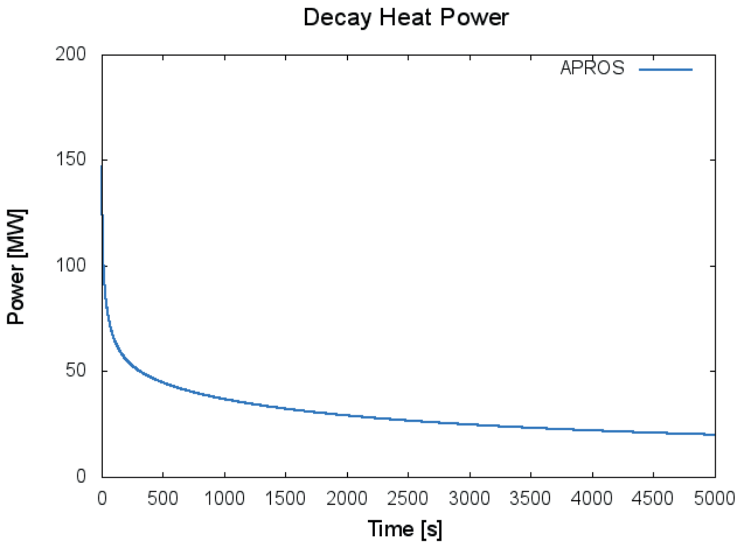


Figure 3: The decay heat curve calculated in APROS during the Reactor Trip transient.

The tubine trip actuates the closure of the turbine stop valve, which has 0.1s closing time, thus the turbine trip happens 0.1s after the Reactor Trip signal. The turbine trip signal also actuates the opening of the bank A and bank B of the SD that try to lower the T_{avg} to the T_{noload} (T=291.67°C) value by releasing steam to the condenser.

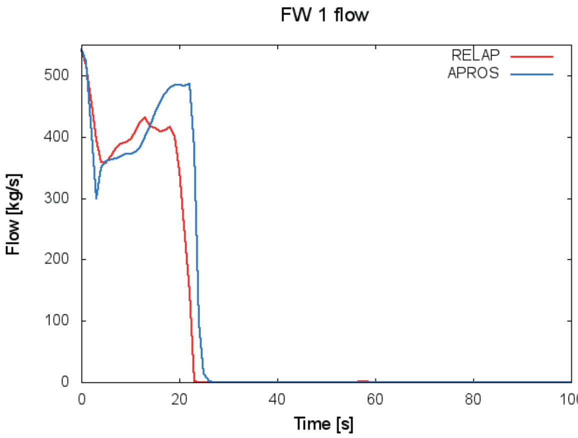
By releasing steam T_{avg} starts to decrease and when it reaches the LO- T_{avg} temperature (295.6°C), the second condition for the FW isolation is fulfilled and the FW isolation valve closes. Then with the delay of 5s the AFW motor pumps start and the AFW is used as secondary coolant. But as there is no exit for the steam after the turbine trip, the pressure starts increasing.

Case A – pressure rises until the PORV opening setpoint is reached and the PORV opens lowering the pressure. The steam pressure of the secondary circuit is therefore determined by the SG PORV opening and the on/off functioning of the AFW and that operation also maintains the SG level between 60% and 70%. And in case B the SD valves open when the T_{avg} temperature goes above T_{noload}. The long term goal of both scenarios is to remove the decay heat from the core.

5 ANALYSIS AND RESULT EVALUATION

The two scenarios, case A and case B were left to run in APROS and the calculated variables were sampled every second. All the relevant parameters were compared to RELAP5 results of the simulation, except those that did not change at all during the transient as the system did not start.

At the beginning of the transient the values calculated in APROS are very similar to those in RELAP5. The FW isolation happens 3 seconds later than in RELAP and the peak flow rate value is approximately 100 kg/s higher. Afterwards throughout the time of the transient the FW system remains isolated and the AFW system takes over regulating the SG level and cooling the core (Figure 4). The time sequence of the main events in the first minute was the same for case A and B, their comparison with RELAP5 is presented in Table 4.



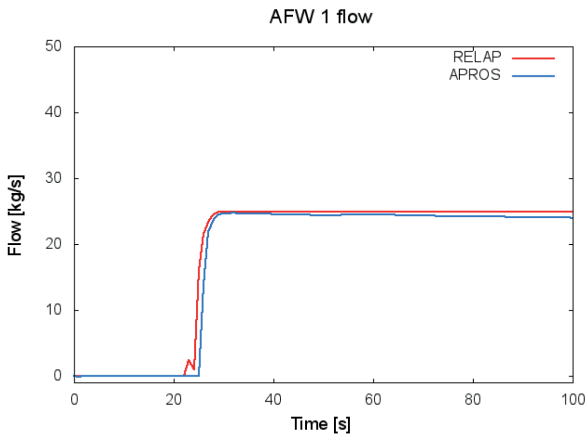


Figure 4: The comparison of the FW and AFW flow the first 100s of the transient.

Table 4: Comparison of the time sequence of main events.

Event	Time of event [s]			Comment
	RELAP5 case A	RELAP5 case B	APROS case A&B	
Reactor trip	0	0	0	
Turbine trip	0.06	0.03	0.1	on reactor trip
Main FW closure	17.91	17.87	21	LO-1 T_{avg} and reactor trip
AFW flow enabled	22.91	22.89	26	5 s delay after Main FW closure
AFW cycling on level enabled	32.91	32.91	36	10 s delay after AFW enabled
MSIV 1(2) isolation	60.09	/	60	60 s after turbine trip signal (case A only)

As stated in the previous paragraph the primary goal during this transient is the cooling of the core, which was successfully achieved in both cases (Figure 5).

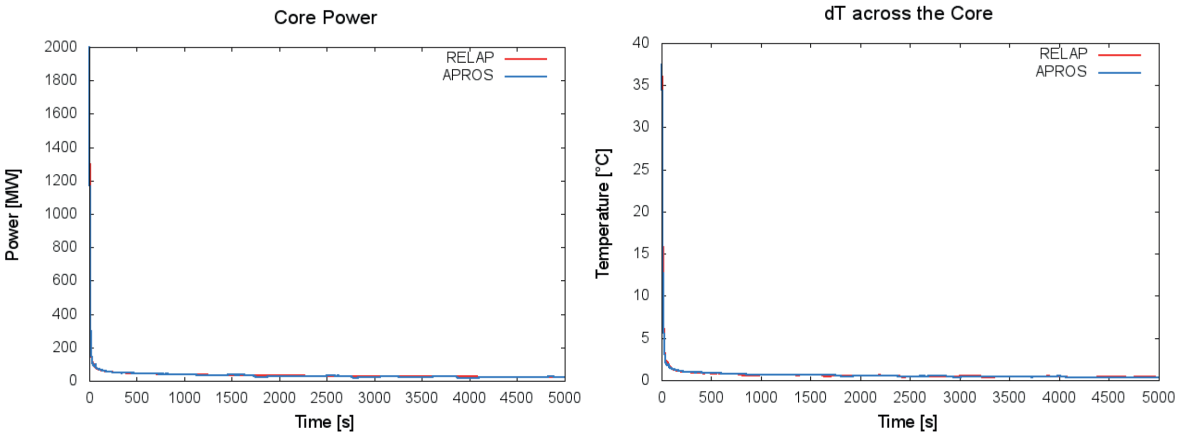


Figure 5: RELAP5 and APROS comparison of core power and dT across the core during the transient.

5.1 Case A analysis

After the MSIV closure the SD system is cut-off form the MS and the regulation of the pressure in MS is taken over by the SG PORVs. Just after the MSIV closure, the pressure of steam exiting from the SGs rises rapidly there is an earlier opening of the PORVs as the setpoint pressure 79.17 bar is reached already after approximately 350s, which is approximately 600s earlier than expected considering the RELAP5 results (Figure 6 left). This more rapid pressure rise at the beginning is because of the different condensation correlation calculation model used in APROS, where the Nusselt theory was chosen, contrarily in RELAP5 the maximum Nusselt and Shah is used. In total during the 5000s seconds of the transient the SG PORVs open 5 times as they do in RELAP5 simulation, but at the first opening the PORVs are open for a shorter period than in the later 4 times, and also the flow rate the first time is lower than the four later times (Figure 6 right).

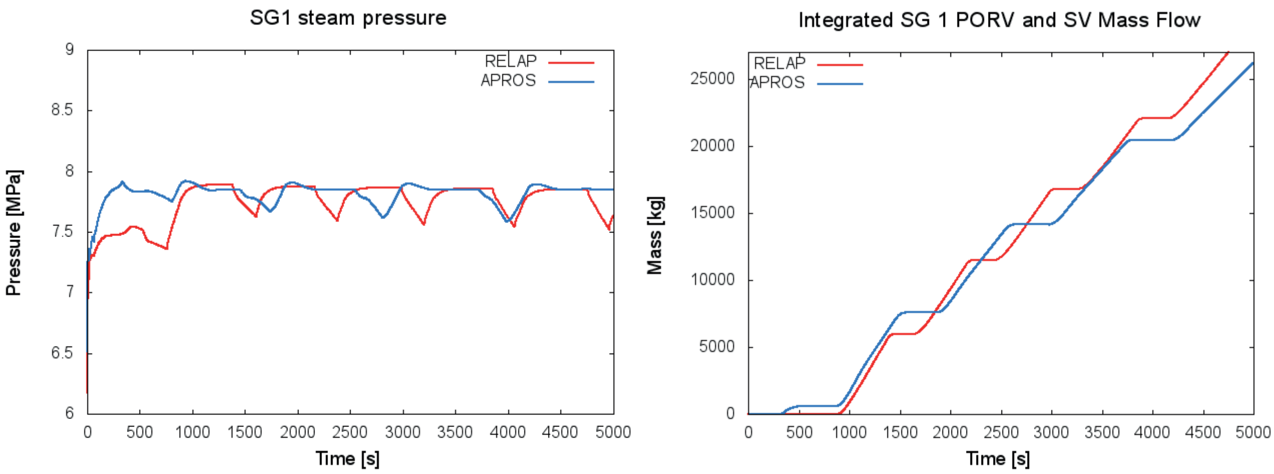


Figure 6: The comparison of the calculated values of the steam pressure exiting the SG 1 (left) and of the cumulative mass of the steam exiting through the SG 1 PORV (right).

In consequence of this different rate of pressure change there is also a different variation of the level change in the SGs instead of five 70% to 60% level drops as result in RELAP5, there are only four in APROS. Therefore there the times, at which the SG level drops to 60% and the AFW is activated, are different than in RELAP5. Additionally the AF pumps have been modelled including the heat-volume flow curve, therefore while they operate their flow rate is not constant (Figure 7).

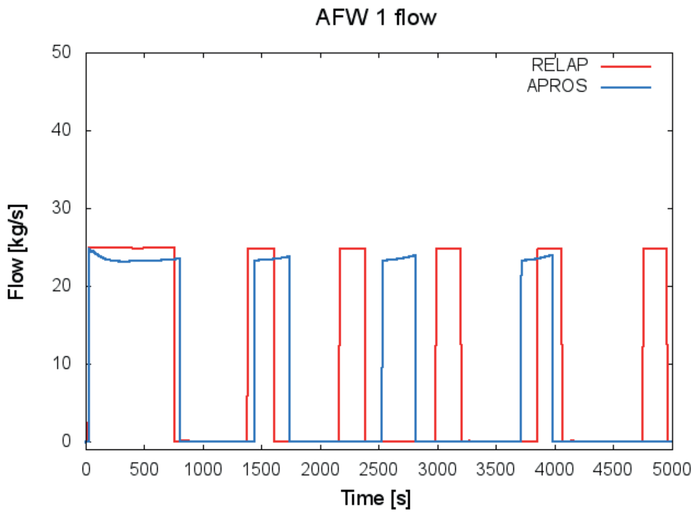


Figure 7: The comparison of the calculated values of the loop 1 AFW into the SG.

Due to this different rate of PORV opening and AFW cycling also other variables are phase-shifted, nevertheless their responses are similar to RELAP5's, as for example the heat flow through the SG. There the thermal heat transfer from the primary to the secondary circuit through the SG U-tubes is the very similar to RELAP5 results the first 20s, between the 20s and 60s the heat transfer in APROS is approximately 20 MW lower than in RELAP, most probably due to effect of a different condensation correlation type used in the models (Figure 8 left). Later during transient the thermal heat transfer is very similar, but there is also visible the phase difference (Figure 8 right).

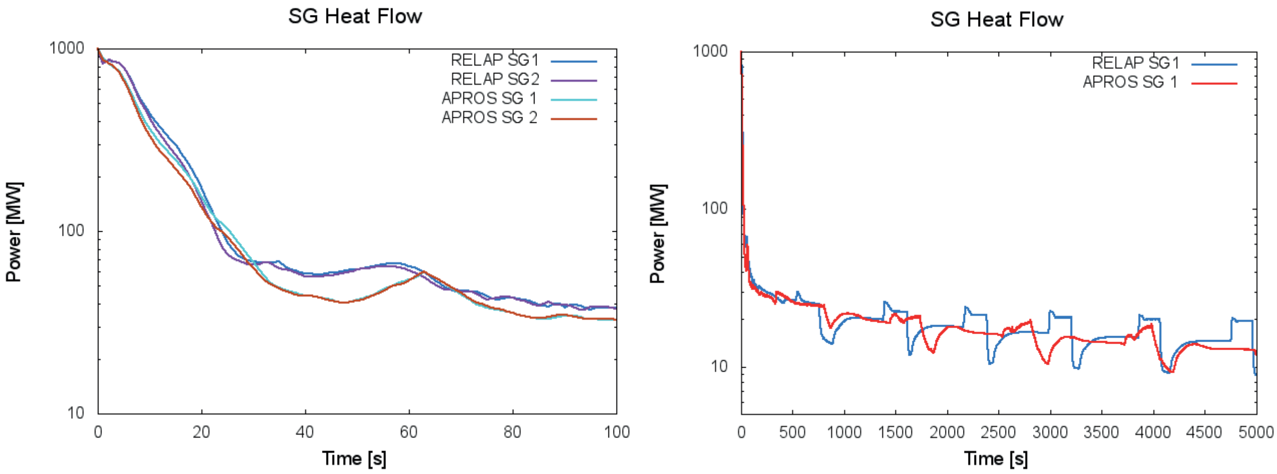


Figure 8: The comparison of the calculated heat flow through the SG U-tubes the first 100s of the transient (left) and throughout the transient (right).

At the beginning of the transient the T_{avg} was being lowered by the SD system and during that time the APROS calculated T_{avg} and consequently T_{hot} and T_{cold} are very similar to the respective RELAP5 results (Figure 9 left), but in later times when the PORVs take over the cooling there is discrepancy due to the phase difference explained above. The T_{avg} the temperature peaks are at approximately 295.5°C in APROS and a degree higher in RELAP5, and in both they decrease for about 0.5°C throughout the transient (Figure 9 right).

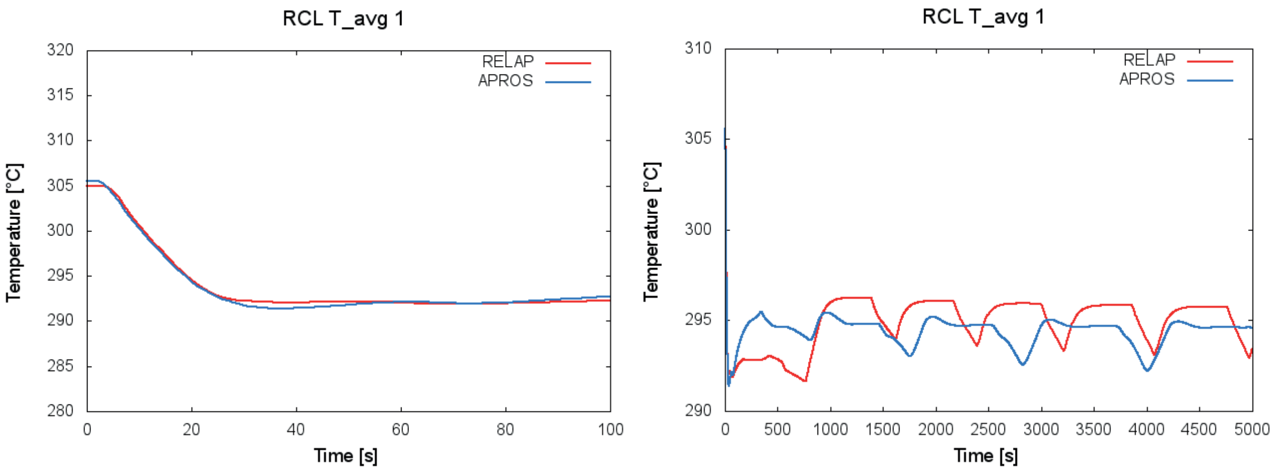


Figure 9: The comparison of the calculated T_{avg} value the first 100s of the transient (left) and throughout the transient (right).

The PRZ pressure peaks a little later than in RELAP5 and the PRZ level recuperates some time later than in RELAP5 (Figure 10 left and right), probably because of the some longer response times in regulation. Therefore we have a higher charging flow in APROS than in RELAP5, the letdown flows are fixed in both models.

The pressurizer heaters (Figure 11 left) and sprays (Figure 11 right) have similar responses in APROS as in RELAP but the spray peaks are much lower in APROS than in RELAP and the phase of the responses is shifted because of the reason described above. The RCS flow in APROS is very similar to that in RELAP, but again there is the same phase shift.

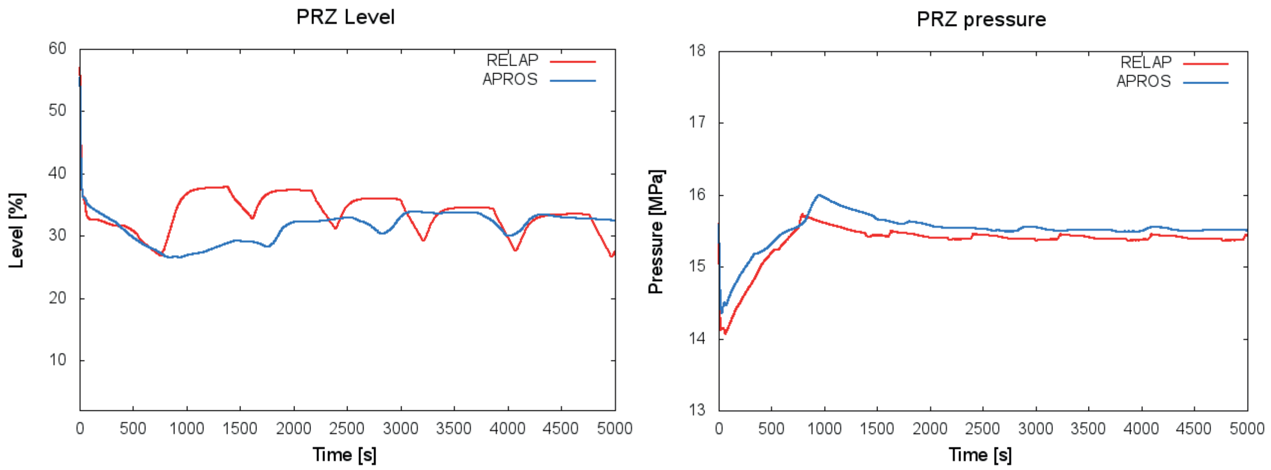


Figure 10: The comparison of the calculated PRZ level (left) and PRZ pressure (right).

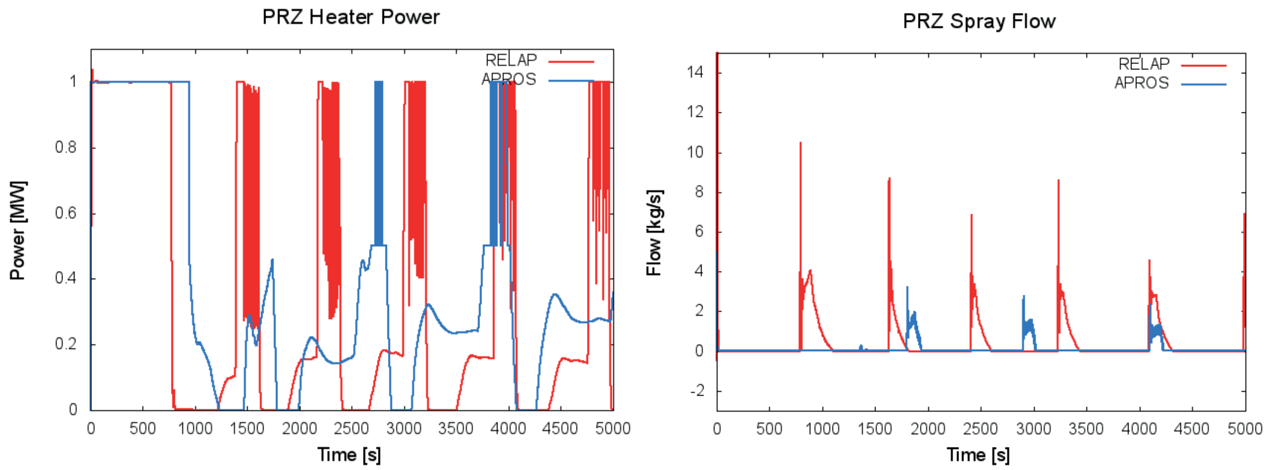


Figure 11: The comparison of the calculated PRZ heater power (left) and PRZ spray flow (right).

5.2 Case B analysis

In case B even after 60 s of the transient the SD system remains in operation and continues to release steam always, when T_{avg} is above T_{noload} in order to lower it (Figure 12 left). In case A and case B the SD opens 3s after the Turbine trip signal. SD flow peaks 4s after the reactor trip in both APROS and RELAP5, only in APROS it is approximately 100 kg/s lower, however it lowers the T_{avg} sufficiently, afterwards in both environments there are a few ripples that continue lowering the temperature, which in APROS are a bit higher than RELAP5. The cyclic operation of SD in APROS starts approximately 75s later than in the RELAP5 model. Most likely because of the reason stated earlier. The SD open cycles in APROS are longer than in RELAP5, but there are only 4 of them, in contrary there are 5 RELAP5 and they have slightly higher flow rates, less than 4 kg/s.

As the oscillations of the SD valves are directly linked to the T_{avg}, there are 5 shorter oscillations of T_{avg} in RELAP5 and 4 longer in APROS. The oscillation peak in RELAP5 is at 292.2°C and decreases for 0.2°C during the transient. In APROS T_{avg} peaks 0.3°C lower and decreases for less than 0.1°C during transient (Figure 12 right).

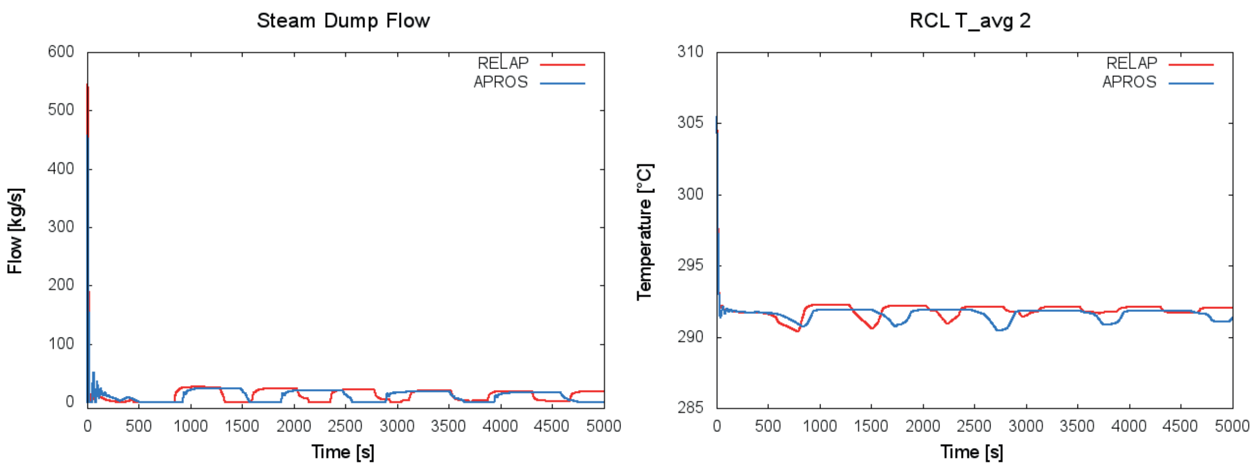


Figure 12: The comparison of the calculated SD flow (left), T_{avg} (right).

The steam pressure in the steam generator is in APROS calculations approximately 1 bar higher than in RELAP5, again here most likely because of the different condensation correlation calculation (Figure 13). The same oscillation pattern of that is given by the SD operation can be seen also in the responses of other variables as in case B they are governed by the SD and AFW cycling. As for example, the heat flow through U-tubes is presented on Figure 14.

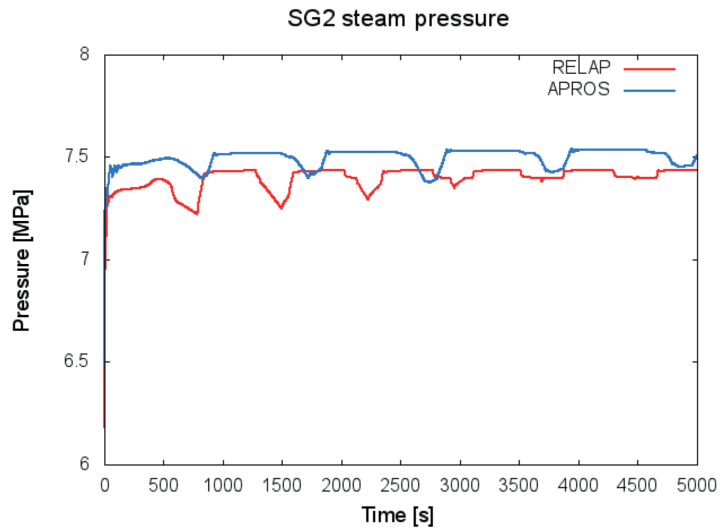


Figure 13: The comparison of the calculated steam pressure exiting the SG 2.

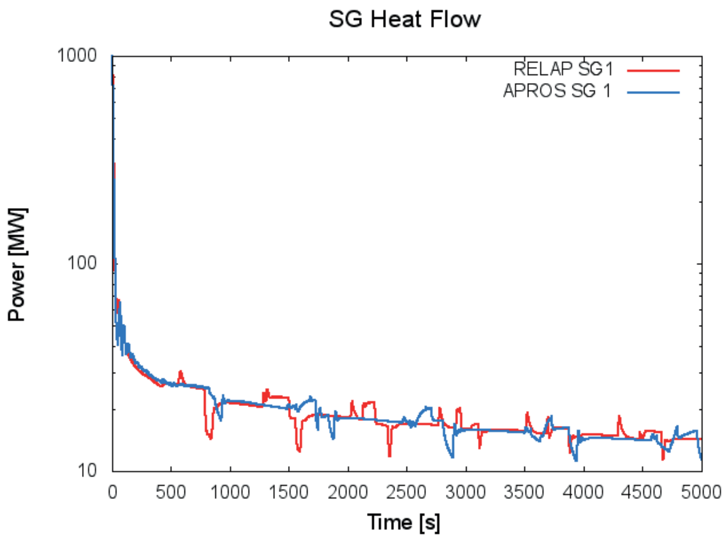


Figure 14: The comparison of the calculated heat flow through the SG U-tubes.

The PRZ level in case B (Figure 15 left) is higher in APROS calculations, but this is most likely due to the slightly higher spray flow rate throughout the transient (Figure 15 right).

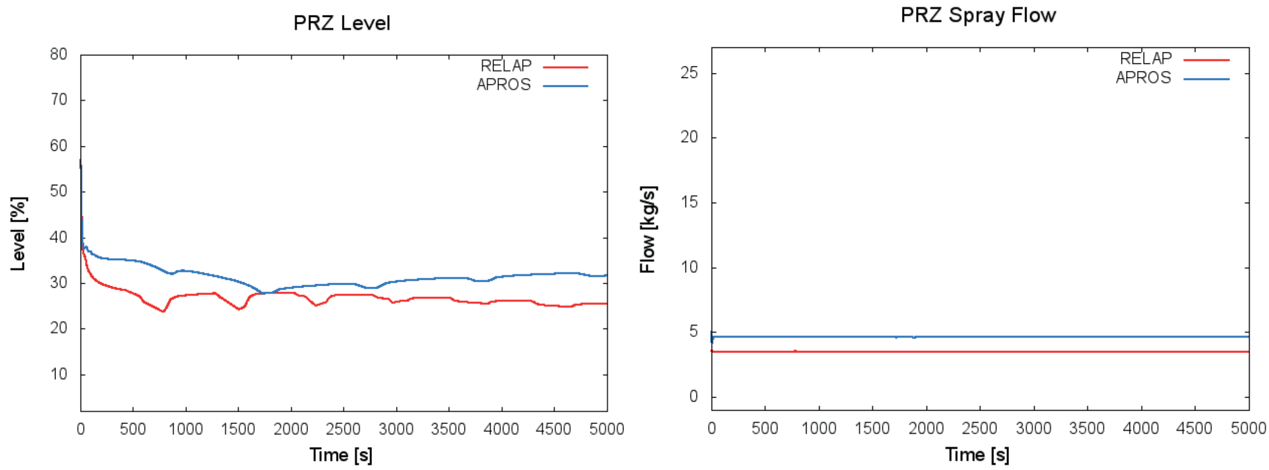


Figure 15: The comparison of the calculated PRZ level (left), PRZ spray flow (right).

6 CONCLUSION

In this Reactor Trip transient simulation two scenarios were analysed in order to observe the response of the SD valves and the SG PORVs. The APROS NEK simulations results were compared to the calculations obtained running the same transient on the RELAP5 model, as this model has already been validated. This analysis showed a satisfactory behaviour of our model. Many differences that arose were also expected as APROS and RELAP5 used different calculation methods. Nevertheless further transient analyses are planned, in order to validate different systems responses that have been incorporated into a wide-ranging model. Then further on, the model behaviour is to be compared to real plant responses. In the immediate future the APROS NEK model will be imported into the new version of APROS 6.05, and will therefore benefit of the more advanced programming options.

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NPP Krško Station Blackout Analysis after Safety Upgrade Using MELCOR Code

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ABSTRACT

The analysis of a Station blackout (SBO) accident in the NPP Krško including thermal-hydraulic behaviour of the primary system and the containment, as well as the simulation of the core degradation process, release of molten materials and production of hydrogen and other incondensable gases will be presented in the paper. The calculation model includes the latest plant safety upgrade with addition of Passive Autocatalytic Recombiners (PAR) and the Passive Containment Filter Venting (PCFV) system. The code used is MELCOR, version 1.8.6. MELCOR is an integral severe accident code which means that it can simulate both the primary reactor system, including the core, and the containment. The code is being developed at Sandia National Laboratories for the U.S. Nuclear Regulatory Commission.

The analysis is conducted in two steps. First, the steady state calculation is performed in order to confirm the applicability of the plant model and to obtain correct initial conditions for the accident analysis. The second step is the calculation of the SBO accident with the leakage of the coolant through the damaged reactor coolant pump seals. Without any active safety systems, the reactor pressure vessel will fail after few hours. The mass and energy releases from the primary system cause the containment pressurization and rise of the temperature. The newly added safety systems, PAR and PCFV, prevent the damage of the containment building by keeping the thermal-hydraulic conditions below the design limits. The analysis results confirm the capability of the safety systems to effectively control the containment conditions.

Results of the analysis are given with respect to the results of the MAAP 4.0.7 analysis for the same accident scenario. The MAAP and MELCOR codes are the most popular severe accident codes and, therefore, it is reasonable to compare their results. In addition, sensitivity calculations performed by varying most influential parameters, such as the hot leg creep failure, blockage of a pipe connecting the cavity and the sump, inclusion of a radionuclide package in the MELCOR, etc. are done in order to demonstrate correct physical behaviour and the accuracy of the developed NPP Krško MELCOR model.

Keywords: *station blackout, core degradation, MELCOR, PCFV, containment integrity*

1 INTRODUCTION

Following the lessons learned from the accident at the nuclear power plant Fukushima in Japan and according to the Slovenian Nuclear Safety Administration (SNSA) Decree No. 3570-11/2011/7 on September 1, 2011 [1], Nuclear Power Plant Krško (NEK) decided to take the necessary steps for upgrade of safety measures to prevent severe accidents and to improve the means to successfully mitigate their consequences. Consequently, the first modifications that NEK implemented during the Outage 2013 were the installation of Passive Autocatalytic Recombiners (PAR) and Passive Containment Filtered Vent (PCFV) systems.

The objective of the paper is to analyze the plant response following the station blackout (SBO) accident with the severe accident code MELCOR 1.8.6 in order to demonstrate its applicability to correctly simulate plant behaviour after implementation of the containment passive safety systems. Furthermore, results of the analysis will be given with respect to the results of the MAAP 4.0.7 analysis for the same accident scenario. The MAAP 4.0.7 calculation results were originally used for the designing of the PAR and PCFV systems and the SBO scenario analyzed herein is similar to those previous runs [2].

The SBO accident includes the loss of all AC power, main feedwater pumps, auxiliary feedwater pumps, safety injection (high and low pressure) pumps, containment sprays, reactor coolant pumps and containment fan coolers. No operator actions are modelled. The reactor coolant pump (RCP) seal injection flow provided by charging pump will be lost and the break at RCP will open. The break flow through the RCP seals of 21 gpm (0.0013 m^3/s) is assumed in the calculation [2].

The MELCOR code [3], [4] is a fully integrated, engineering-level computer code whose primary purpose is to model the progression of accidents in light water reactor nuclear power plants. A broad spectrum of severe accident phenomena in both boiling and pressurized water reactors is treated in MELCOR in a unified framework. Current uses of MELCOR include estimation of fission product source terms and their sensitivities and uncertainties in a variety of applications. The MELCOR code is composed of an executive driver and a number of major modules, or packages, that together model the major systems of a reactor plant and their generally coupled interactions.

Initially, the MELCOR code was envisioned as being predominantly parametric with respect to modelling complicated physical processes (in the interest of quick code execution time and a general lack of understanding of reactor accident physics). However, over the years as phenomenological uncertainties have been reduced and user expectations and demands from MELCOR have increased, the models implemented into MELCOR have become increasingly best estimate in nature. Today, most MELCOR models are mechanistic, with capabilities approaching those of the most detailed codes of a few years ago. The use of models that are strictly parametric is limited, in general, to areas of high phenomenological uncertainty where there is no consensus concerning an acceptable mechanistic approach.

2 MELCOR MODEL OF THE NPP KRŠKO

2.1 Models of the RCS and the Secondary System

The primary and secondary systems, including regulation systems and control volumes that represent boundary conditions, consist of 104 thermal-hydraulic control volumes, 125 flow junctions and 48 heat structures. The nodalization scheme of primary and secondary systems is shown in Figure 1.

Hot legs in each loop are modelled with two control volumes (CV), intermediate legs with four and cold legs with one volume. Reactor coolant pumps are defined by pressure heads and its respective volumes are added to adjacent control volumes. The pressurizer is modelled with two CVs (103 and 105), while the volume CV 104 represents the pressurizer relief tank where pressurizer power operated relief (PORV) and safety valves (SV) discharge steam. CV 109 and CV 209 represent accumulators. The steam generator (SG) inlet part is modelled with one volume, the outlet part also with one CV, and U-tubes with six control volumes. On the secondary side, SG downcomer is modelled with one CV, riser section with four and the SG separator and the dome with one CV. Auxiliary and main feedwater (MFW) flow are directed into the SG downcomer taking suction from the control volumes CV 503 and CV 513 for the MFW, and CV 504 and CV 514 for the AFW flow. SG safety and relief valves (FL375-FL380 for SG 1 and FL475-FL480 for

SG 2) discharge steam into CV 921 for the first and CV 922 for the second steam generator. Steam lines from the steam generators 1 and 2 are represented by control volumes 811 and 812. The steam header and the steam line to the turbine are represented with the control volumes 813 and 814, respectively. CV814 is connected to CV 901 (pressure boundary condition that simulates the turbine) by the valve that closes after the turbine trip. In the steady state calculation the valve opening is modulated to obtain the referent reactor coolant system (RCS) average temperature.

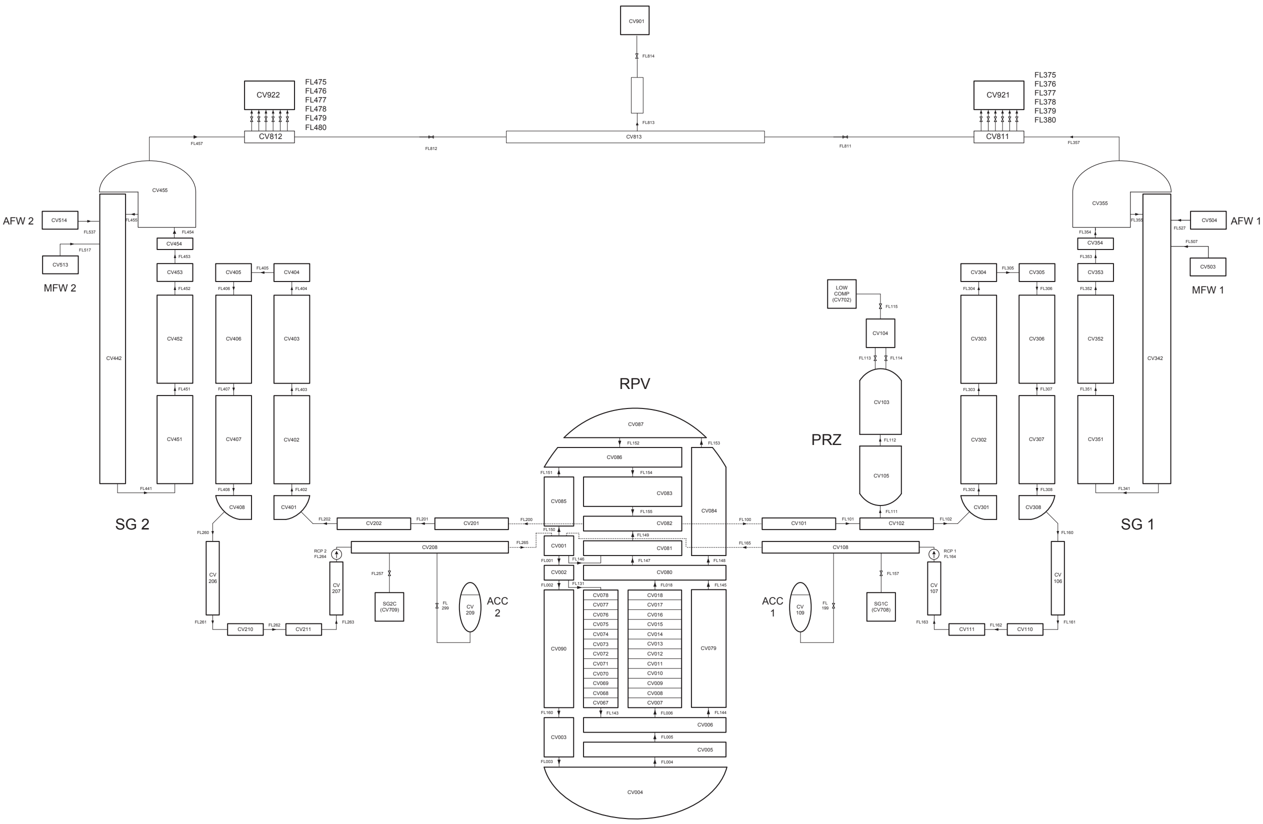

Figure 1: NEK nodalization for MELCOR 1.8.6 code calculation

2.2 Reactor Pressure Vessel and Core Models

The reactor pressure vessel (RPV) is modelled with 40 control volumes. The lower plenum is represented with 3 CVs, the downcomer with 5 CVs, the upper plenum with 4 CVs and the upper head with 2 CVs. One CV was used to represent RCCA guide tubes (CV 084). The flow inside the reactor core was represented with 12 control volumes (CV 007-018), as well as the baffle-barrel flow (CV 067-078). The guide tubes bypass inside the core was represented with one control volume (CV 079).

The height of one control volume inside the core is 0.3048 m, because the total core height is 3.6576 m.

Axial representation of the reactor core and the lower plenum used in the MELCOR COR package is shown in Figure 2. Seven radial rings are used to represent the RPV, 5 rings for the core, one ring for the region between the baffle and the barrel, and one additional ring in the lower plenum as requested by the code. The lower head is represented with more radial rings (10) for better prediction of the RPV wall temperature which is used to calculate the RPV rupture.

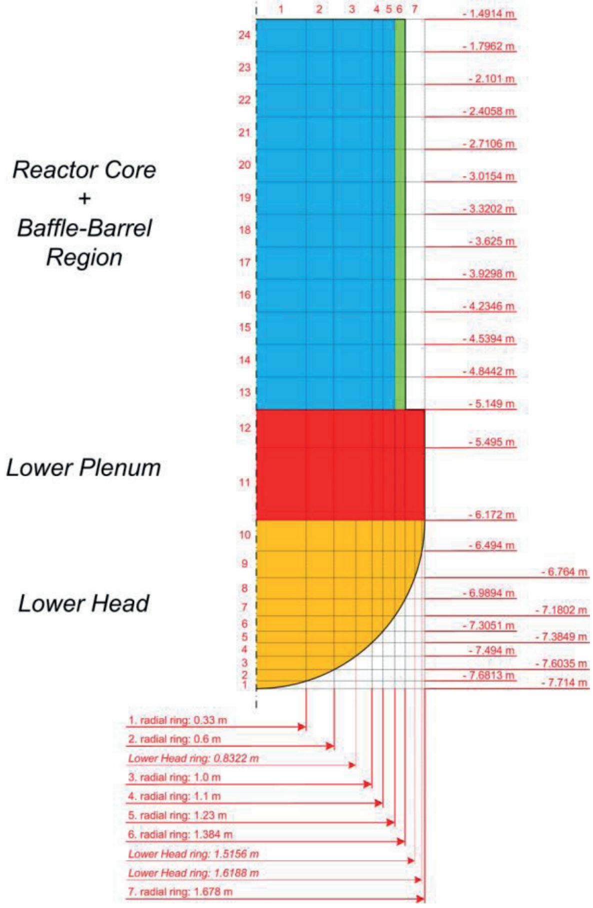

Figure 2: Axial representation of the reactor core and the lower plenum used in the COR package

2.3 Containment Model

The NEK containment model is based on the NPP Krško containment nodalization notebook [5] which contains detailed calculations of containment free volume and heat structures' dimensions, and data of the containment geometry. Containment nodalization is shown in Figure 3.

The containment building is represented with 12 control volumes:

1. CV 701 (containment dome) – cylindrical/spherical air space above the reactor pool, steam generators and pressurizer compartments,
2. CV702 (lower compartment) – lower compartment below the containment dome placed between SG1, SG2 and PRZ compartments excluding the reactor pool and the reactor pressure vessel area,
3. CV 703 (pressurizer compartment) – air space in the compartment that contains pressurizer and primary system safety and relief valves,
4. CV 704 (reactor cavity) – air space below the reactor vessel including the instrumentation tunnel,
5. CV 705 (annulus) – air space between the steel liner and the containment building,
6. CV 708 (steam generator 1 compartment) – air space in the SG1 compartment that contains components SG1 and RCP1,
7. CV 709 (steam generator 2 compartment) – air space in the SG2 compartment that contains components SG2 and RCP2,
8. CV 710 (reactor pool) – air space above the reactor vessel filled with water during the shutdown, otherwise empty,
9. CV 711 (around reactor vessel) – air space between the reactor vessel and the primary shield walls,

10. CV 712, 713, 714 (containment sump) – the lowest control volumes below the SG1 compartment and the lower compartment that contain recirculation and drainage sumps.

There are three additional volumes:

1. CV 706 – refuelling water storage tank,
2. CV 707 – connection between the upper compartment and the environment, added to control opening/closing of the PCFV relief valve,
3. CV 900 (environment) – a large volume (10^8 m^3) at constant temperature (307 K) and pressure (10^5 Pa).

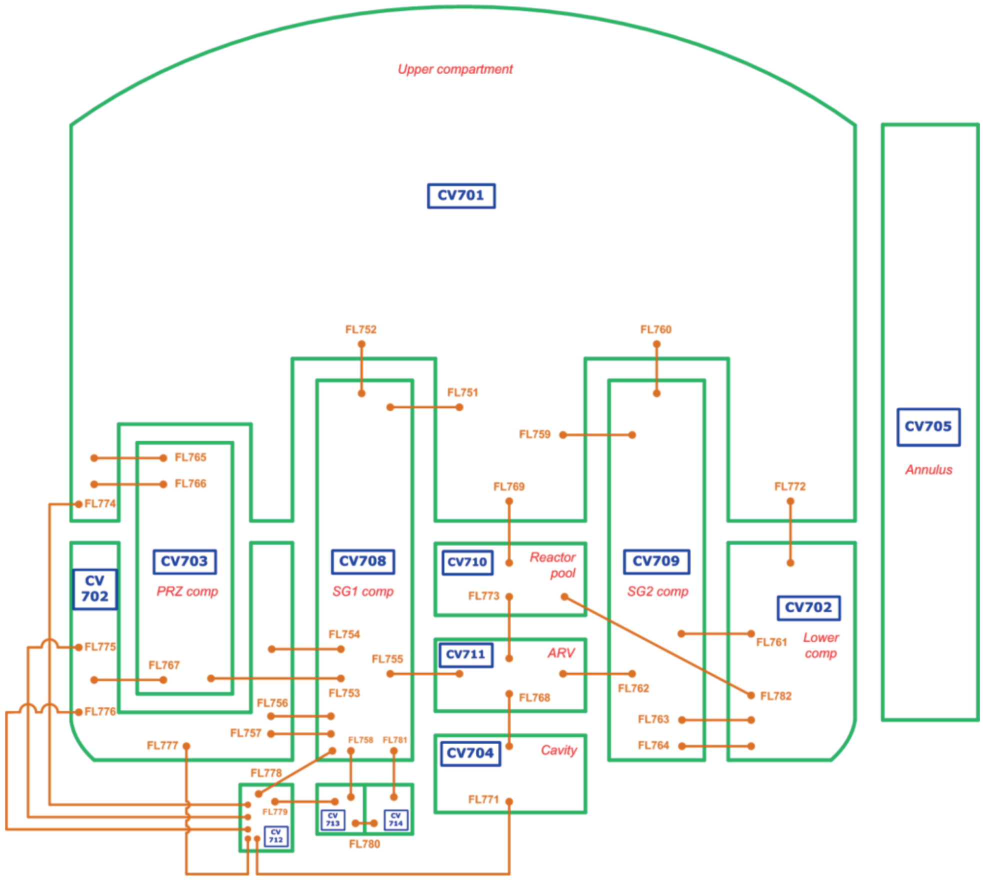

Figure 3: Containment NEK nodalization

Containment control volumes are connected by 30 junctions and heat sinks representing outside containment wall, internal walls, floors, polar crane, fan coolers, platforms, other miscellaneous stainless and carbon steel structures are modelled with 20 heat structures. Additional information about the containment model can be found in [6].

2.4 Transient Description

The analyzed transient is a standard SBO accident where no active components such as pumps and fans are available. The accident also involves release of coolant through damaged RCP seals with a leakage rate of 21 gpm (0.0013 m^3/s) per one reactor coolant pump. In addition, the letdown line is assumed to be isolated which consequently leads to the opening of the letdown relief valve to the pressurizer relief tank increasing the coolant loss from the RCS. Taking this into account, break flow areas are set to $1.8 \cdot 10^{-5}$ m^2 for the loop 1 and to $9.5 \cdot 10^{-5}$ m^2 for the loop 2 (which letdown line is attached to).

Unmitigated SBO sequence of events will include core degradation and melting, RPV rupture and release of corium in the containment cavity, interaction between the corium and concrete (MCCI), release of incondensable gases and containment pressurization. Passive autocatalytic recombiners are used to control hydrogen concentration and the PCFV system restricts containment pressure below the rupture limit.

3 ANALYSIS AND RESULTS

3.1 Thermal-Hydraulic Conditions in the RCS and the Core Meltdown

The results of the MELCOR calculation were compared with results of the MAAP code [7] calculation. On most of the diagrams, they were put together to highlight similarities and differences between the codes. The MAAP 4.0.7 code version and the NEK model of the plant [8] that includes PARs and the PCFV system based on the latest calculation performed by NEK [2] was used in the MAAP calculation. The intention of both calculations is to support licensing review process and demonstrate adequacy of introduced modifications as part of the NEK safety upgrade project.

The accident started with the closure of the turbine valve, trip of the RCPs and the reactor trip. The primary pressure (Figure 4) rose sharply at the beginning of the transient due to termination of the reactor coolant flow, but immediately afterwards, loss of coolant through the breaks at RCP seals lead to the primary pressure decrease. In the meantime, SG pressure oscillated around 8 MPa (Figure 5) due to periodical openings of SG safety valves. After 2000 s the primary pressure increased once again to 16 MPa and remained at that value until the hot leg pipe ruptured at 9700 s. The rupture was due to the creep thermal and mechanical stress and deformation caused by the hot vapour at high pressure coming from the reactor core. The pressure was maintained by the operation of pressurizer safety valves. The break flow rate was not large due to relatively small break area of the RCP pump seals. The limited leakage rate from the breaks, in combination with the loss of forced circulation in the RCS and the unavailability of the secondary heat sink, resulted in the primary system overheating (Figure 6) and the increase of the pressurizer pressure. Since the pressure, prior to the hot leg failure, never dropped to a value of 5 MPa, no accumulators' injection into the cold legs occurred in that period.

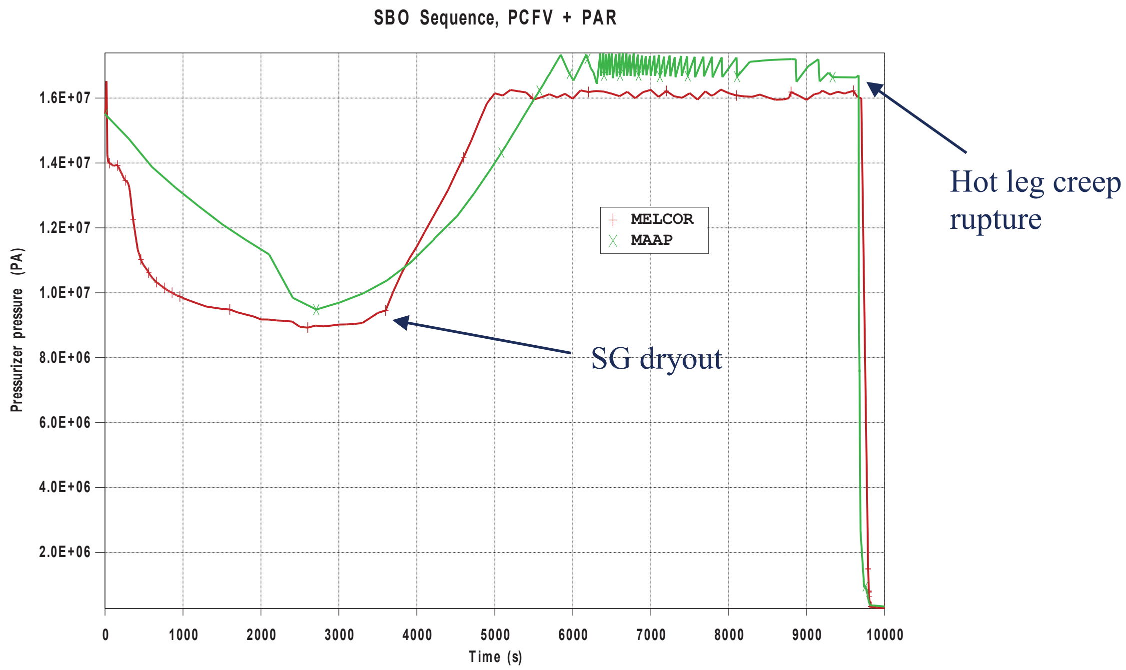

Figure 4: Pressurizer pressure

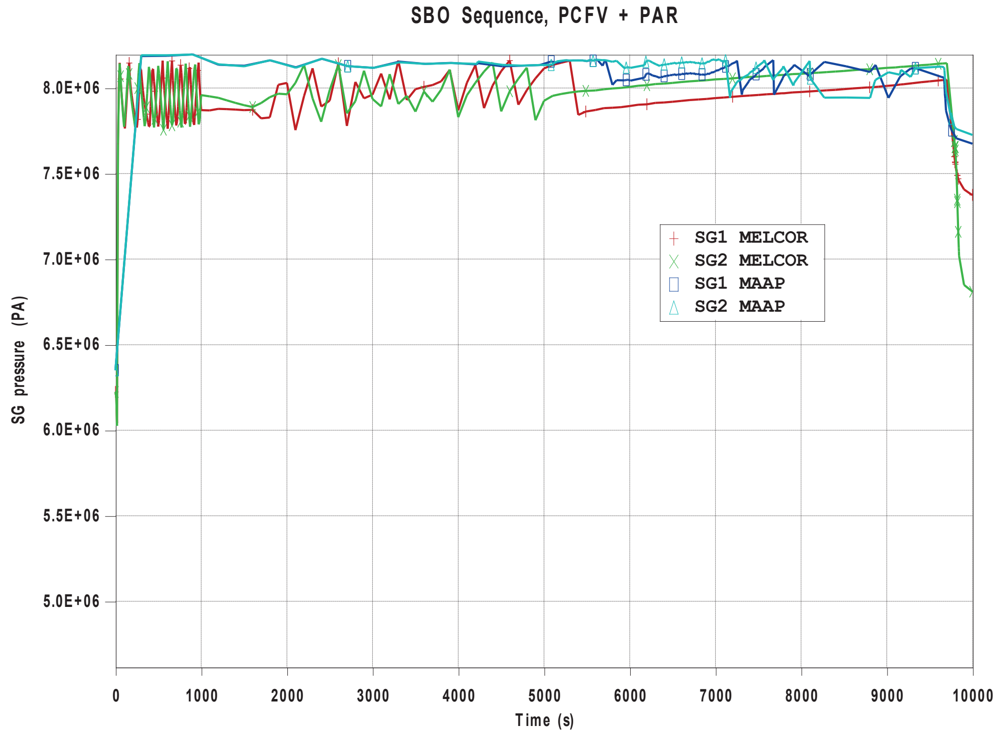

Figure 5: Steam generator pressure

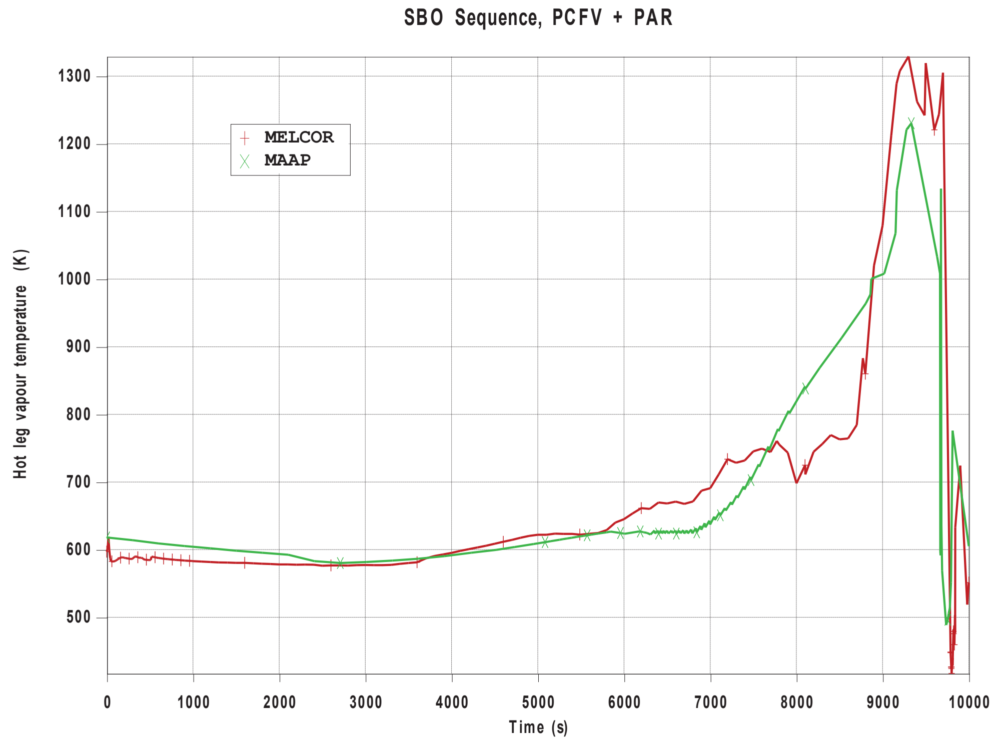

Figure 6: Hot leg vapour temperatures

As more and more water was discharged from the RCS to the containment, the core started to uncover and to heat up. The upper core levels uncovered at 6000 s, and the lower levels at 7500 s. The core heatup was additionally supported with oxidation of fuel rod cladding and other metallic materials. The mass of hydrogen produced during oxidation in the core was about 250 kg.

S. Šadek, D. Grgić, V. Benčik, NPP Krško Station Blackout Analysis after Safety Upgrade Using MELCOR Code, Journal of Energy, vol. 65 Number 3–4 (2016) Special Issue, p. 102-115

Since there was no water injection to stop the temperature increase, the core began to melt, first the inner fuel elements and later the outer ones. The process lasted about one hour. The melt front eventually propagated to the core boundary where the melt breached through the core baffle plates and relocated to the lower plenum. Relocation of the molten material to the RPV lower head caused heatup and failure of the reactor vessel wall at about 20000 s. The break was due to the creep failure as the consequence of the material fatigue when exposed to high thermal and mechanical stress. Damage of the reactor pressure vessel led to release of the corium into the containment cavity and start of the molten corium concrete interaction (MCCI). The mass of corium released from the vessel was around 80 tons.

3.2 Containment Behaviour and the Molten Corium Concrete Interaction

Heat transfer from the hot corium (~2000 K) to water in the cavity caused the water to evaporate. The released steam was a major contributor to containment pressurization (Figure 7). Cavity was initially empty, but the coolant released from the RCS drained into the containment sump and from there it entered the cavity. The cavity has dried out rather quickly (Figure 8). The time of the cavity dryout coincided with the point of termination of the fast pressure increase rate.

Connection between the sump and the cavity by the 4 inch pipe enabled water to enter and to flood the cavity from the start of the accident. About 60000 kg of water was present in the cavity before the vessel ruptured. The pipe was closed after the failure of the reactor pressure vessel and the release of the corium. The volume of corium is large and it is reasonable to assume that the narrow pipe will be blocked once when filled with the melt. The blockage means that there will be no more water flow from the sump to the cavity causing the fast water depletion in the cavity.

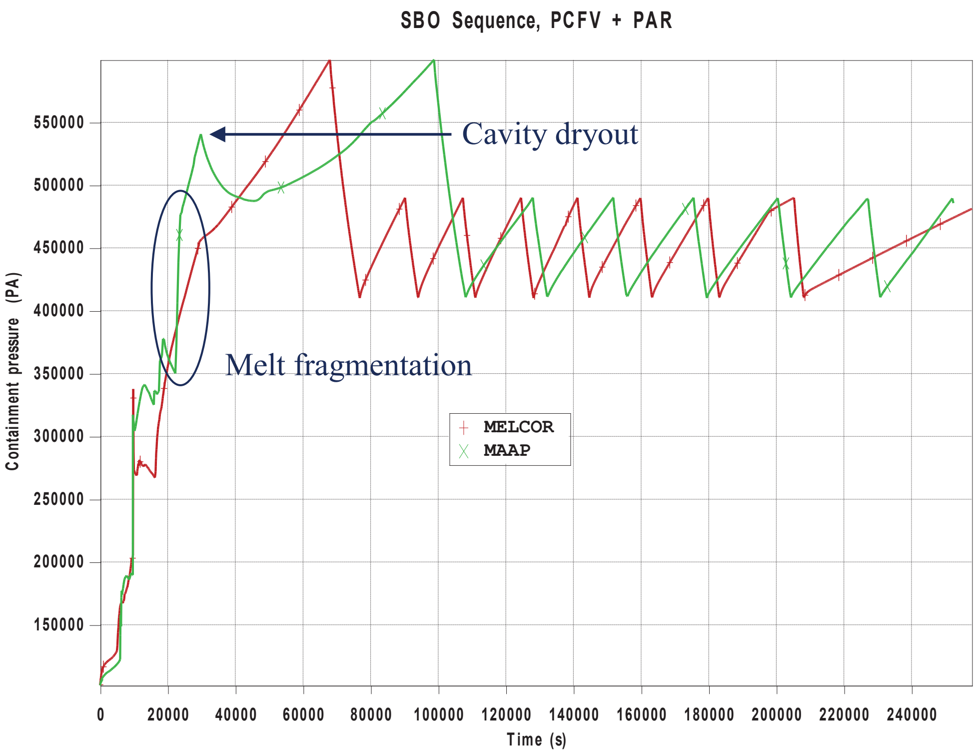

Figure 7: Pressure in the containment upper plenum

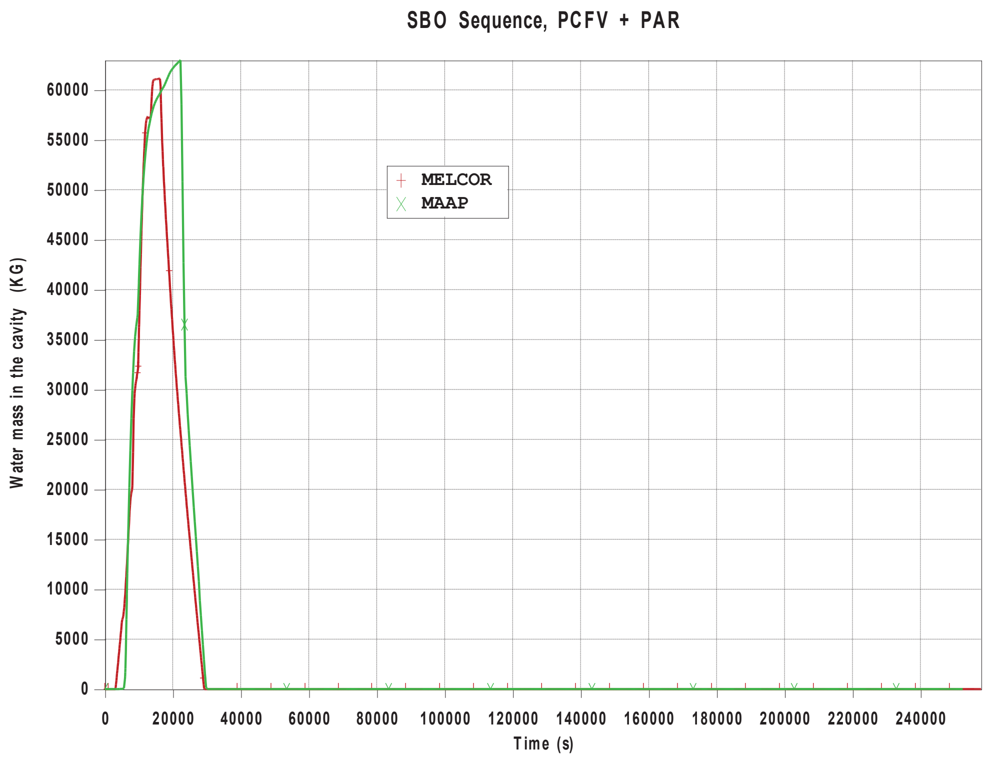

Figure 8: Mass of water in the reactor cavity

MELCOR calculated the first pressure peak to occur 30000 s earlier than MAAP. MAAP assumes that the melt is fragmented and quenched when it is submerged in water. MELCOR does not take the possibility of quenching into account which means that the melt is all the time at the high temperature. Therefore, after the water in the cavity evaporates, containment pressure and temperature (Figure 9) start to rise immediately in the MELCOR calculation, while in MAAP there is a time delay because the relocated material first has to be heated up by the decay heat and only after that the heat can be transferred in the containment atmosphere.

The pressure profile follows the PCFV system logic of controlling containment conditions. There are two components in the filtered venting line which behaviour depends on the pressure, the rupture disc and the relief valve. The rupture disc ruptured at 68000 s (MELCOR case) when the containment pressure reached 6 bar (the first pressure peak in Figure 7). Later, the pressure was cycling between 4.1 bar and 4.9 bar by the operation of the PCFV relief valve that had a hysteresis characteristic. The relief valve was closed when the pressure dropped below 4.1 bar and opened after it increased above 4.9 bar.

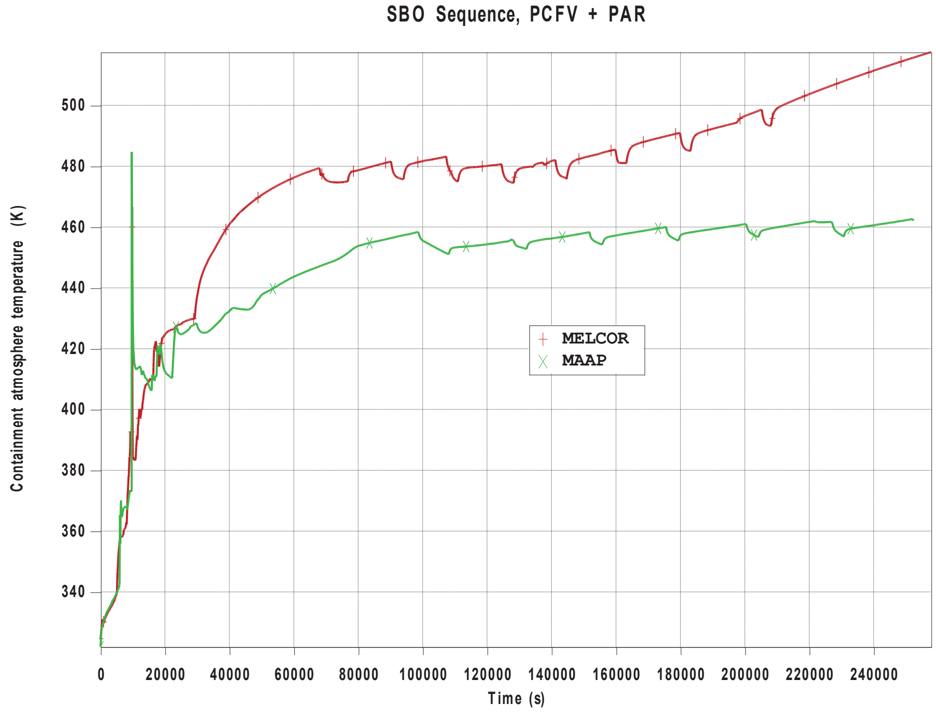

Figure 9: Temperature in the containment upper plenum

The reaction between the molten corium and the concrete (MCCI) erodes containment basemat (Figure 10) and results with production of incondensable gases: hydrogen, carbon monoxide and carbon dioxide. Hydrogen and CO are released during oxidation of iron contained in concrete rebar with steam, produced by evaporation of water bounded in the concrete, and CO_2, respectively. Carbon dioxide is released during decomposition of calcium carbonate ($CaCO_3$) into calcium oxide (CaO) and CO_2. Figure 11 shows total gas releases due to the molten corium concrete interaction as calculated by MELCOR. Those gases also contribute to overall containment pressurization.

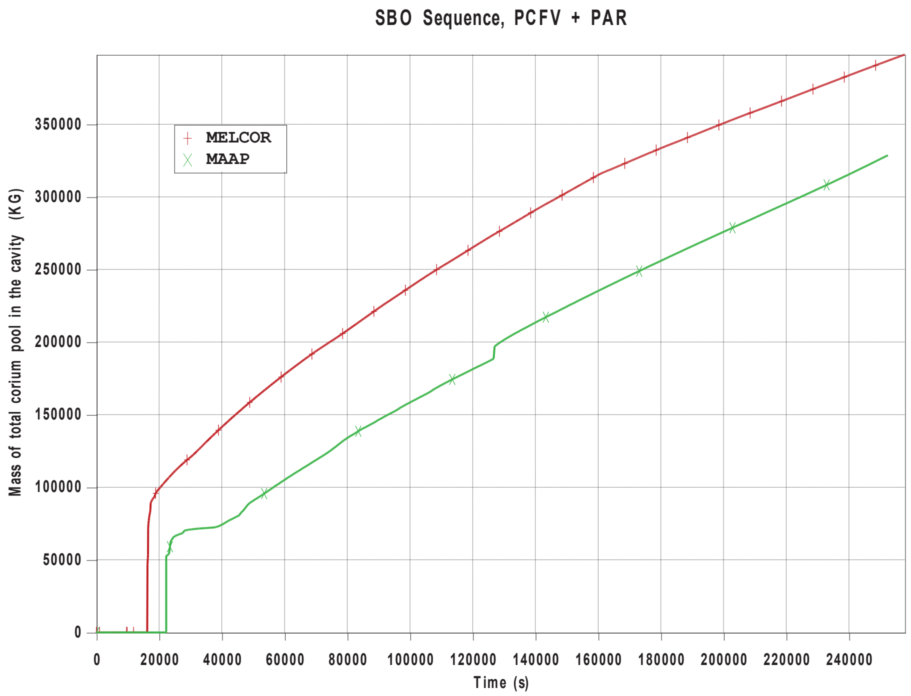

Figure 10: Mass of the total corium pool in the reactor cavity

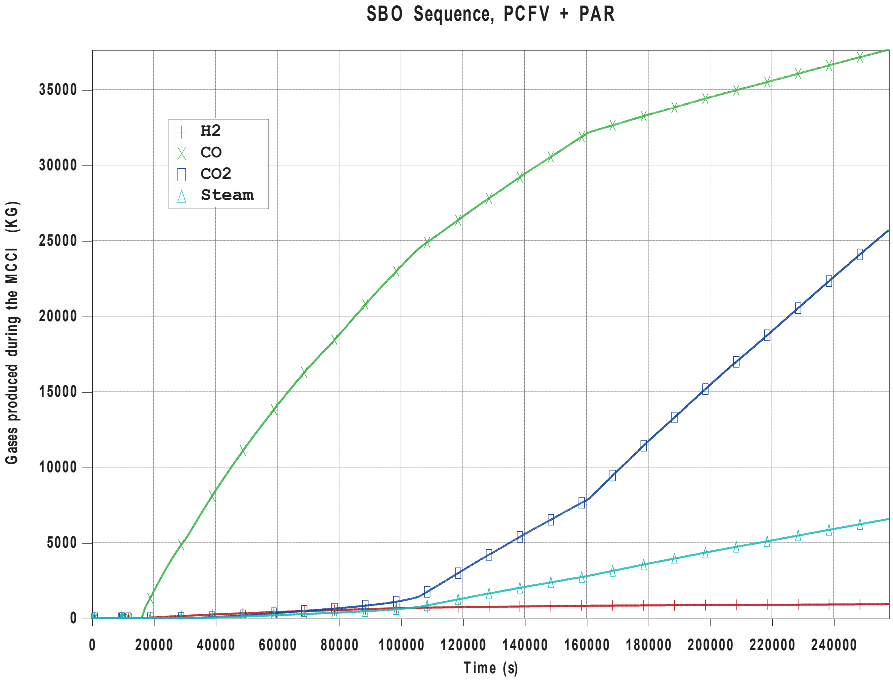

Figure 11: Gases released during the MCCI, MELCOR calculation

There are 22 PAR units installed in the containment. Passive autocatalytic recombiners are used to control hydrogen concentration in the containment by forcing the reaction between hydrogen and oxygen. PARs removed 850 kg of hydrogen which is about 80% of hydrogen produced by the oxidation in the core and the MCCI. Comparison between the hydrogen production and recombination rates is shown in Figure 12. PAR operation started when hydrogen mole fraction reached value of 0.02 and stopped after oxygen mole fraction dropped to 0.005.

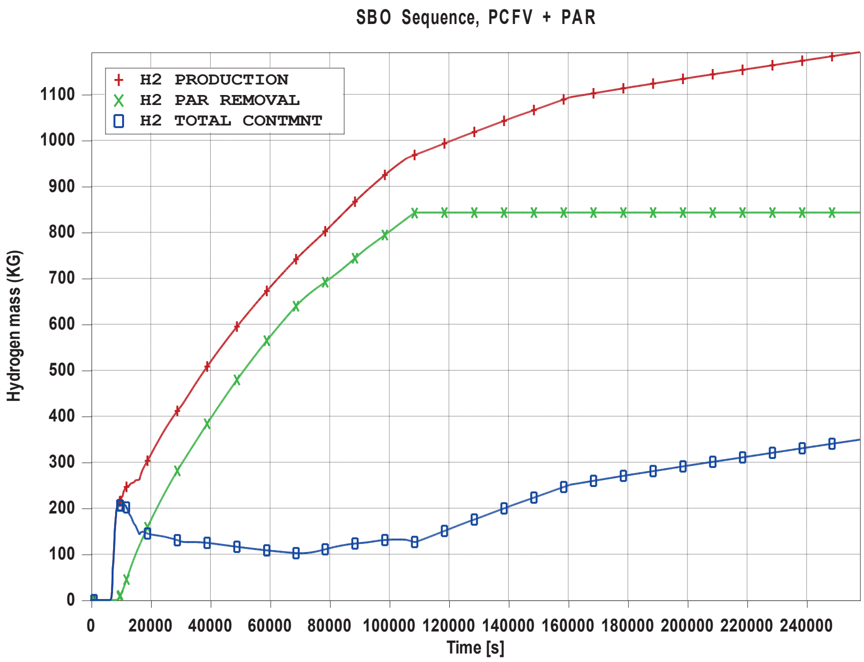

Figure 12: Hydrogen production, hydrogen mass removed by PARs and total hydrogen mass in the containment, MELCOR calculation

3.3 Sensitivity Calculations

The reference analysis was performed with the blocked pipe that connects the sump and the cavity. It was assumed that after the corium is released from the reactor vessel, it would block the pipe and stop the flow of water and steam. The sensitivity calculation with the pipe opened throughout the transient was performed to evaluate the influence of constant water flow between the sump and the cavity.

Containment pressure behaviour is much different if the flow in the pipe is not blocked (Figure 13). The reason is that water in the cavity is now depleted much later. The water evaporates at the same rate but it is being constantly added from the containment sump. Steam that condensates in the containment finishes eventually in the sump because all the drainages paths lead to the sump. Thus, the sump never dries out completely and as long as the upper elevation of water in the sump is higher than the bottom of the cavity, water will enter into the cavity. That water dries out and the steam pressurizes the containment. The pressure of 6 bars, when the PCFV rupture disk fails, is reached before the cavity dryout and so the whole sequence of opening and closing the PCFV relief valve takes place much earlier than in the previous sequence.

The MELCOR radionuclide (RN) package calculates fission product release and transport inside the circuits and the containment. Including the package, decay heat of released products will additionally heat up and pressurize the containment as can be seen in Figure 13.

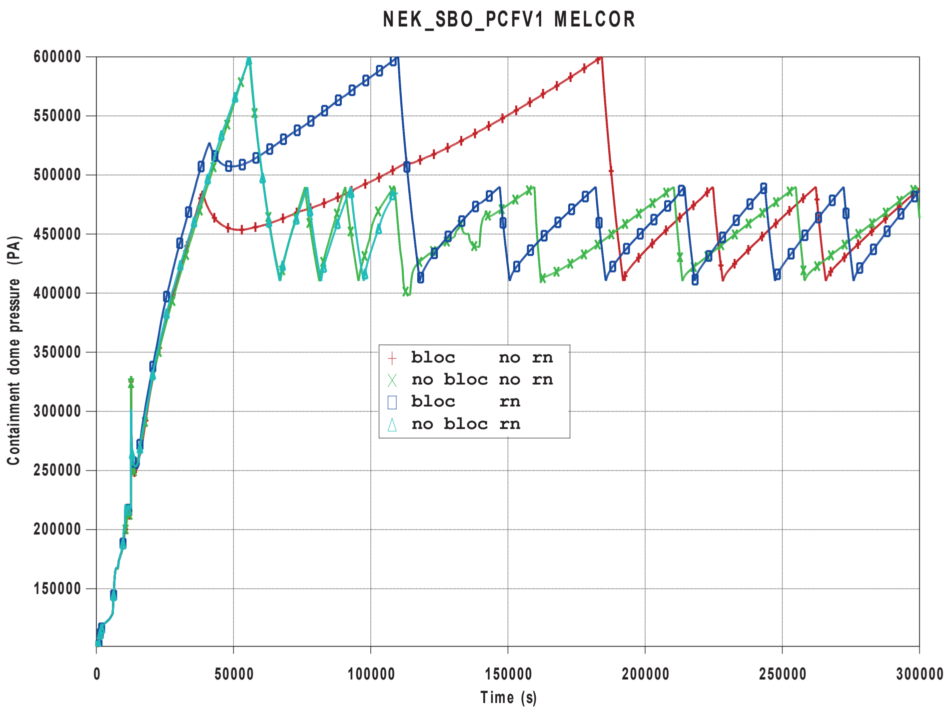

Figure 13: Pressure in the containment upper plenum, MELCOR calculation

In the SBO analysis the occurrence of the hot leg creep failure prior to the RPV rupture was assumed. Additional calculations were performed to check the influence of such assumption on the containment behaviour. After the introduction of the hot leg creep rupture, containment pressure rose faster, Figure 14. When there was no hot leg creep, the RCS was at a high pressure until the RPV failure. The accumulators then discharged water directly into the containment cavity through the break in the reactor vessel. More water in the cavity in the case with no hot leg creep failure provided more efficient cooling of the melt and less heat release in the containment. Thus, containment pressure and temperature increased slower when no hot leg creep failure was modelled.

S. Šadek, D. Grgić, V. Benčik, NPP Krško Station Blackout Analysis after Safety Upgrade Using MELCOR Code, Journal of Energy, vol. 65 Number 3–4 (2016) Special Issue, p. 102-115

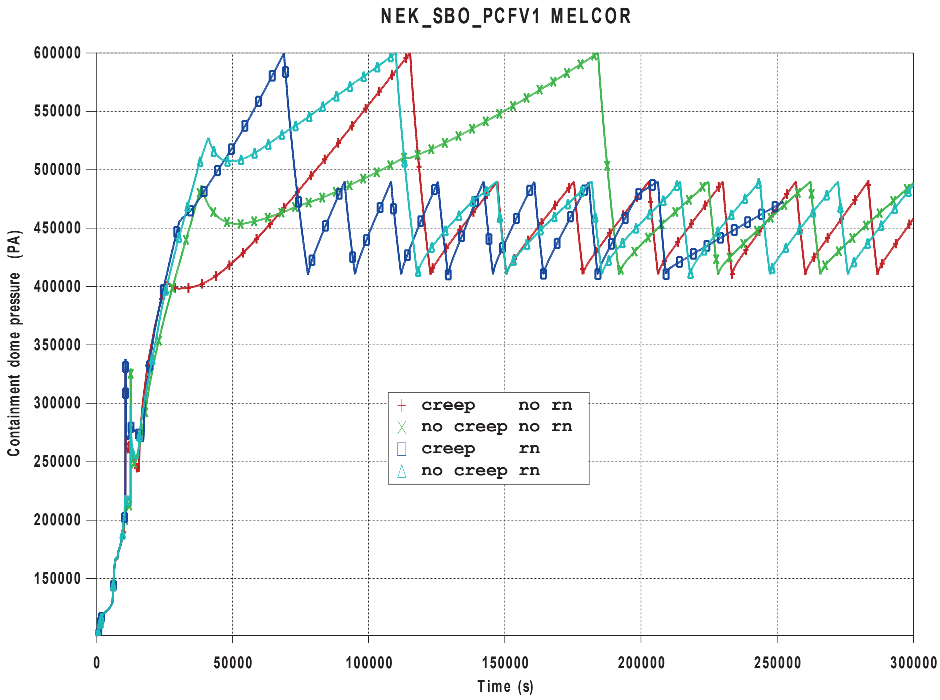

Figure 14: Pressure in the containment upper plenum, MELCOR calculation

4 CONCLUSION

A postulated station blackout accident at the NPP Krško with PAR and PCFV systems was analyzed with the MELCOR 1.8.6 code and its results were compared with the MAAP 4.0.7 calculation of the same transient scenario. The sequence of events included total core meltdown due to unavailability of safety injection, release of corium in the containment and containment pressurization by steam and incondensable gases and decay heat release from the melt. The PCFV system provided controlled release of containment inventory and kept the pressure at the system design limits. Passive autocatalytic recombiners reduced the hydrogen concentration by maintaining the chemical reaction between hydrogen and oxygen.

The MELCOR model of the primary and secondary systems, including regulation systems and boundary conditions, consisted of 104 thermal-hydraulic control volumes, 125 flow junctions and 48 heat structures. The reactor pressure vessel (RPV) was modelled with 40 control volumes. The reactor core and the lower plenum used were represented with 24 axial seven radial rings, 5 rings for the core, one ring for the region between the baffle and the barrel, and one additional ring in the lower plenum as requested by the code. The containment was divided in 12 thermal hydraulic control volumes. Three volumes instead of one were used to represent sump in order to realistically predict water drainage from other containment compartments and water supply to the cavity.

Calculation results support PCFV and PAR systems performance in controlling containment conditions during severe accidents. MELCOR and MAAP code predictions were similar except when parametric approach used in MAAP considerably influenced transient propagation. That was the case in predicting the first containment pressure peak which occurred later in the MAAP calculation due to the selection of melt fragmentation option in the cavity water. Fragmentation of the corium jet led to the fast melt cooldown and slower pressure increase once when the water evaporated. Models in MELCOR are mainly mechanistic and the plant behaviour is calculated based on the empirical correlations adopted in the code. In the analysed SBO scenario MELCOR did not calculate melt fragmentation and thus the pressure and temperature rise went faster. Nevertheless, overall system behaviour was similar to results obtained in the PCFV design calculation.

Sensitivity calculations that simulated the influence of a blockage of a 4 inch pipe connecting the sump and the cavity and the hot leg creep rupture on the containment thermal hydraulic conditions demonstrate the correct physical behaviour and the accuracy of the developed NPP Krško MELCOR model to predict plant behaviour with respect to varying initial and boundary conditions.

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Verification of GOTHIC Multivolume Containment Model during NPP Krško DBA LOCA

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ABSTRACT

New containment multivolume model of NPP Krsko for GOTHIC code is developed. It is based on plant drawings and other available data. It is supported by developed SketchUp 3D containment model. The model is subdivided in volumes following physical boundaries and clearly defined flow paths. All important concrete heat structures are taken into account. Metal heat structures are based on plant's SAR Chapter 6 licensing model. RCFC (Reactor Containment Fan Cooler) units are explicitly modelled as well as all main ventilation ducts. The model includes two trains of containment spray system. PARs (Passive Autocatalytic Recombiner) and PCFV (Passive Containment Filter Venting) filters added during plant safety upgrade project are part of the model too.

It was intention to use model for both DBA (Design Basis Accident) and for DEC (Design Extended Conditions) and BDBA (Beyond Design Basis Accident) calculations. Based on the same discretization and data, and on experience acquired during GOTHIC model development and use, containment models for MELCOR and MAAP integral codes are developed too.

As part of initial verification of the GOTHIC model containment DBA LOCA calculation is performed using SAR MER (Mass and Energy Release) data. The influence of different break positions on peak containment atmosphere pressure and temperature was studied. The results were compared against results obtained in single volume containment licensing model. Beside local effects due to different containment subdivision similar results are obtained when comparing containment dome from multivolume and the single volume in licensing model. Special attention was paid to distribution of water in lower part of the containment during recirculation phase. In this case much more valuable information are obtained in multivolume model with explicit volumes for main sump, recirculation sump and sump pit. Another point of interest was influence of containment spray duration on long term pressure and temperature behaviour. The intention was to study consistency of assumed different spray operation times used in safety analyses, EQ analyses and SAMGs and related consequences for plant operation during DBA LOCA.

Keywords: *containment, LOCA, GOTHIC, multivolume model, spray operation time*

1 INTRODUCTION

New containment multivolume model of NPP Krsko for GOTHIC code is developed/upgraded [1]. It is based on plant drawings and other available data. It is supported by developed Sketch-up 3D containment model. The model is subdivided in volumes following physical boundaries and clearly defined flow paths. All important concrete heat structures are taken into account. Metal heat structures are based on plant's SAR Chapter 6 licensing model. RCFC (Reactor Containment Fan Cooler) units are explicitly modelled as well as all main ventilation ducts. The model includes two trains of containment spray system. PARs (Passive Autocatalytic Recombiner) and PCFV (Passive Containment Filter Venting) filters added during plant safety upgrade project are part of the model too.

It was intention to use model for both DBA (Design Basis Accident) and for DEC (Design Extended Conditions) and BDBA (Beyond Design Basis Accident) calculations. Based on the same discretization and data (geometry data base), and on experience acquired during GOTHIC model development and use, containment models for MELCOR and MAAP integral codes are developed too. The objective of the paper is to verify multi compartment containment model response to DBA LOCA used in development of EQ (Environmental Qualification) conditions. The results are compared to single volume containment model developed for SAR type calculation. In addition, influence of longterm containment spray availability to pressure and temperature qualification profiles was studied.

2 GOTHIC MODEL OF THE NPP KRŠKO

2.1 Single compartment model

The computer code GOTHIC [2][3] was used for calculation of containment thermal hydraulic behavior. The code solves conservation of: mass, momentum, and energy equations for multiphase (vapor phase, continuous liquid phase, droplet phase) multicomponent (water, air, H2, noble gases) compressible flow. Constitutive relations predict interaction between phases for nonhomogenous, nonequlibrium flow. Heat structures are modeled as 1D heated or unheated structures. Hydraulic volumes use 1D, 2D, 3D or lumped approach. It is possible to simulate operation of engineered safety equipment: pumps, fans, valves and doors, vacuum breakers, spray nozzles, heat exchangers, heaters and coolers, controlled by trip logic based on: time, pressure, vapor temperature, liquid level, conductor temperature. The code is tested and qualified to perform calculations similar to this analysis.

Single compartment NPP Krsko containment model is developed for SAR and EQ type of analyses. The total free volume of the containment was modeled as one compartment, compartment number 1 on Figure 1. The containment annulus was modeled as another separate volume, compartment number 2. The values needed for model preparation are taken from USAR [4]. Initial conditions inside containment were chosen to maximize containment pressure (48.9 ^{o}C, 101.325 kPa, 30% RH). The temperature of the outside air was 34 ^{o}C.

Containment heat structures were modeled as 14 different heat structures, according to USAR. The heat structures 1 and 2 are used for representing steel liner (vessel cylinder and dome) and structures 3 and 4 are used for concrete containment wall (cylindrical and hemispherical part). All other heat structures are internal containment heat structures. HVAC cooper heat structure is additional passive heat structure. Material thermal properties were determined based on USAR data. The heat transfer coefficient was based on Uchida condensation correlation and natural convection heat transfer coefficient, for all heat structures exposed to the containment atmosphere. In addition, in case of LOCA, Tagami heat transfer correlation was used for internal metal heat structures during blowdown. For internal heat structures right side of the structure is isolated. For structures 3 and 4 right side is exposed to the environment with fixed temperature (34 ^{o}C) and fixed heat transfer coefficient of 11.36 W/m^{2}K.

Seven flow boundary conditions are present in the model (7F is hydrogen source when needed). Boundary conditions 1 and 2 were used for single sided breaks. In case of double side break boundary conditions 4 and 5 are used to model second side of the break. Two flow boundary conditions per break side can be used to split fluid flow in liquid and gas phases explicitly or, as was case in LOCA calculation, to model blowdown and reflood discharges separately. Boundary condition 6F was used only during LOCA recirculation phase to model part of the fluid removed from the sump by RHR system. Flow paths 1, 2, 6, 7 and 11 are associated with flow boundary conditions. Boundary condition 3F and flow path 3 (10 for recirculation) were used for modeling of spray flow from RWST tank (water temperature 37 °C). GOTHIC spray nozzle component was used at the end of flow path to convert all water flow to droplets. Constant spray mass flow rate was used (75.8 kg/s) during both injection and recirculation phase. Flow path 10 together with pump and spray nozzle 2N was used for modeling of spray line during recirculation phase of LOCA. Each reactor containment fan cooler was modeled separately as volumetric fan cooler (1Q) + standard single-pass, finned tube counterflow air-to-water heat exchanger (1H). The dimension of the tubes and fins, total heat transfer area and other RCFC operational data were realistically taken into account. Flow paths 4, 5, 8 and 9 are used to model flow of steam and air mixture over RCFC units 1 to 4 cooling surfaces. The influence of RCFC units was not taken into account during normal operation (before break) due to lack of the model for heat losses from the primary side. Actuation of both sprays and RCFC units is prescribed by (35 and 55 s respectively for analysed DEPSG MIN SI sequence) accident scenarios.

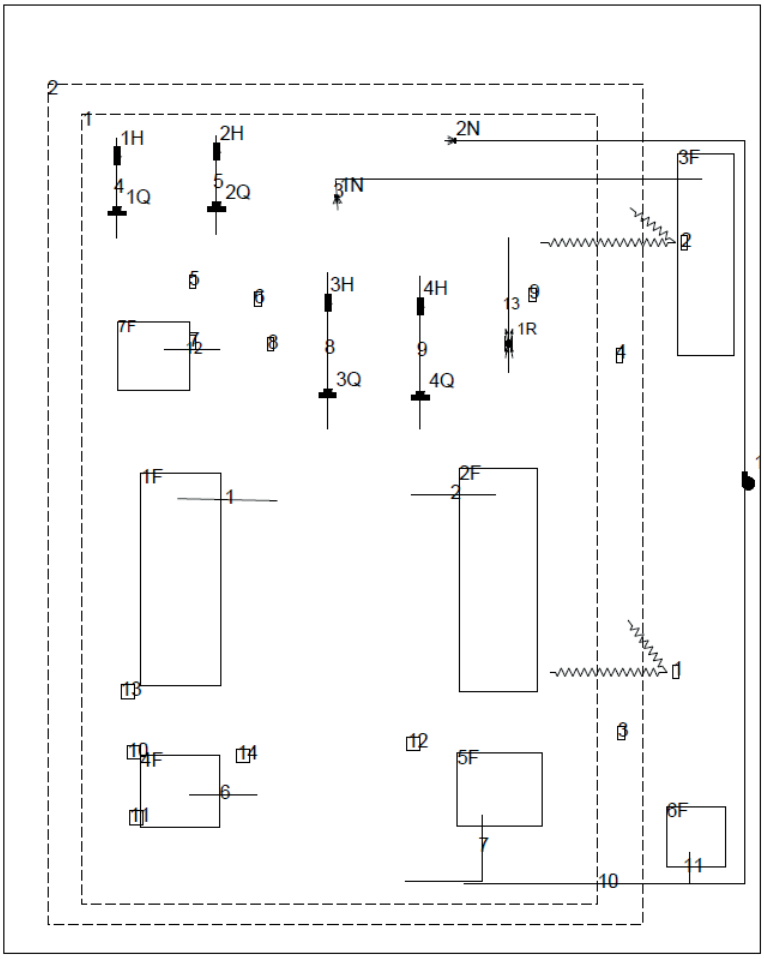

Figure 1: GOTHIC single compartment LOCA model

2.2 Multi compartment model

NPP Krško GOTHIC multi compartment model of containment is developed from described licensing model. Free volumes were recalculated during Krško modernization project as well as heat structures related data. All other containment engineering safety features are the same as used in licensing type of calculations. The explicit model of RCFC ducts is included in the model. The same flow boundary conditions are used in single and multi compartment model, but they are connected in different way.

Based on additional drawings available during Krško modernization project a new referent containment nodalization using 10 main control volumes/compartments was developed. Each control volume of the model is so defined to have clear physical boundaries and well defined openings based on physical containment compartment's boundaries, Figure 2.

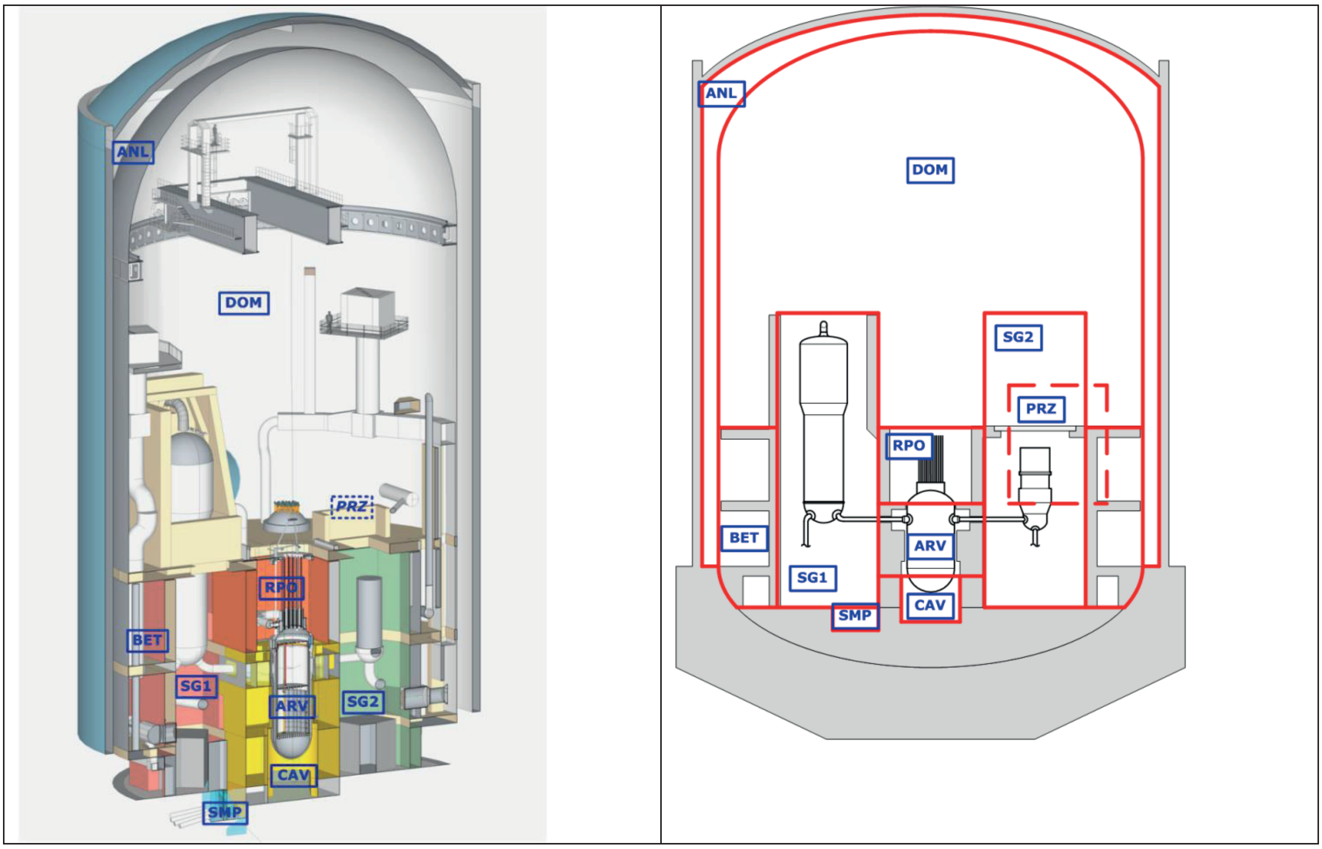

Figure 2: NEK containment layout and proposed basic subdivision

During development of the nodalization it was decided to build 3D geometry data base using SketchUp code. Whole model is decomposed in concrete heat structures and empty volumes following main containment elevations, Figure 3. Based on that main compartment volumes where developed, with enough data to enable further subdivision if needed. By using physical compartment boundaries in the development of the model local effect of different breaks can be better studied (e.g. LOCA can be within SG compartment or close to the reactor vessel), Figure 4.

As part of the model upgrade the connection between the sump and the cavity is better defined, and the sump is divided in three control volumes to reflect the real containment design, Figure 5. Additional junctions are connected to sump CVs to simulate accurately water drainage from other compartments into the sump, and the sump connections with the SG1 compartment and the cavity. All lower compartments are checked from point of flooding volume corectness.

The corresponding nodalization scheme is shown in Figure 6, together with bounding elevations. Containment dome (DOM) is control volume number 1. The annulus (ANA) space is control volume number 2. Steam generator 1 and 2 (SG1 and SG2) compartments are control volumes 3 and 4. The pressurizer compartment (PRZ) is volume 5. The reactor pool (RPO) and space around reactor vessel (ARV) are volumes 6 and 7. The reactor cavity (CAV) and containment sump (SMP) are volumes 9 and 10. The space outside listed volumes up to the containment liner and below elevation 115.55 m is called between (BET) and it is volume number 8. The reactor pool is modelled as a separate volume in order to take into account shutdown cases with opened vessel and pool filled with water. The total volume is almost the same as single compartment model described in licensing data. Concrete heat structures are based on real bounding walls and floors between the compartments. The metal heat structures are based on licensing model and are subdivided using engineering judgment between separate compartments in multi-volume model.

The flow paths in the model are based on real openings and communications between the compartments. More than one opening is used between the same volumes if they are located at different elevations to promote internal thermal mixing flow, what can be important for long term containment transients. When required, it is possible to join more flow paths in one equivalent flow path. The leakage (based on design leakage) flow paths exist between containment dome and annulus and between annulus and the environment. All main ventilation ducts were taken into account, what resulted in increase of used control volumes beyond the number used for main compartments. The nodalization can be in addition subdivided axially using floor elevations and laterally in two halves. Each volume can have internal axial elevations (useful in SG. PRZ and ARV compartments). The containment dome can be further subdivided using GOTHIC 3D capability in 2x2x4 subvolumes, Figure 7.

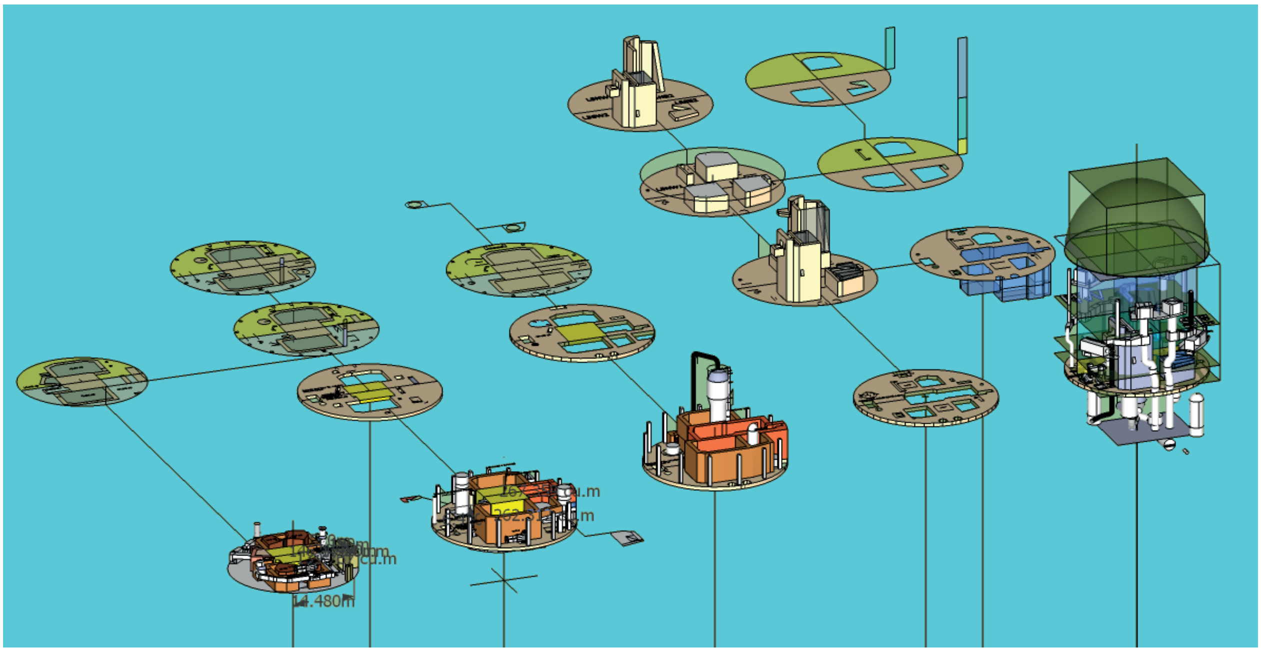

Figure 3: Decomposition of containment geometry

3 ANALYSIS AND RESULTS

3.1 DBA LOCA Calculation, single compartment model

DBA LOCA case, 0.6 Double Ended Pump Suction Guillotine break DEPSG with minimum SI (0.6 DEPS min SI, one RCFC and one spray train in containment), was used as representative case for thermal hydraulics testing of the model. For DEPSG MINSI LOCA mass and energy releases were taken from USAR Chapter 6.

For single compartment model 3.e6 s simulation time is used (the same as used in EQ calculation). Due to longer running time of the multi compartment model and rather small difference between single and multi-volume containment model, the simulation time for multi compartment model was limited to 1.e5 s.

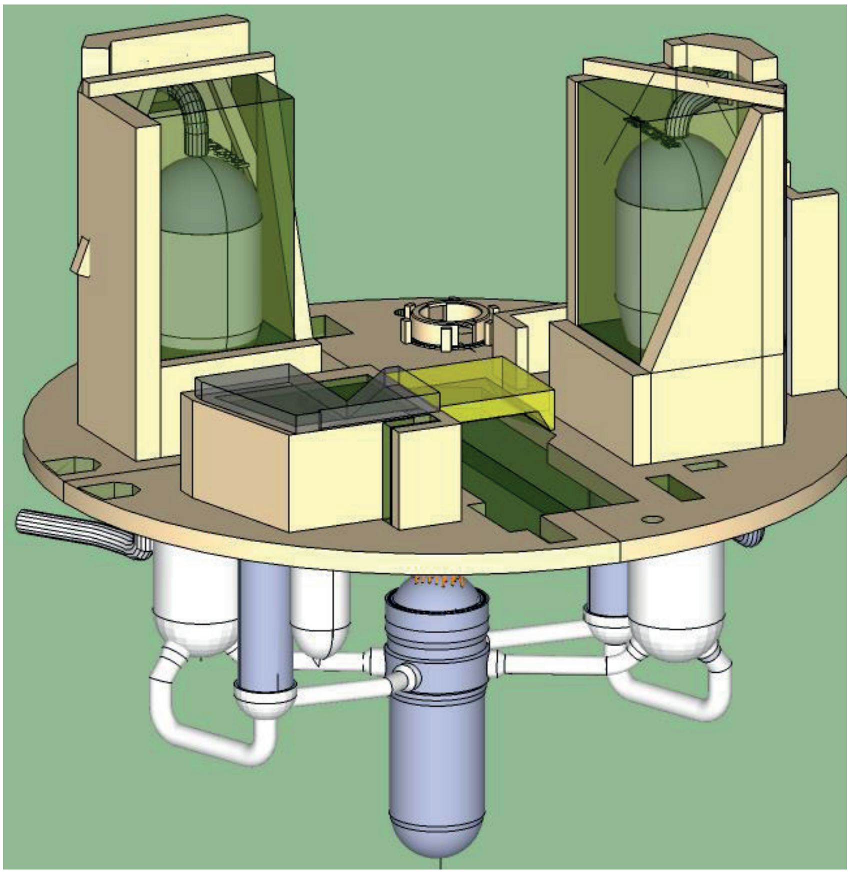

Figure 4: Break localization

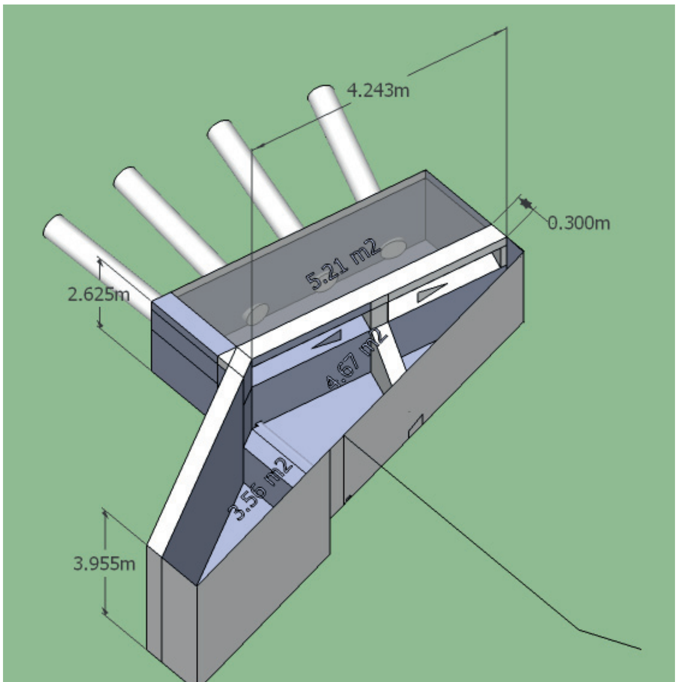

Figure 5: Containment sump geometry

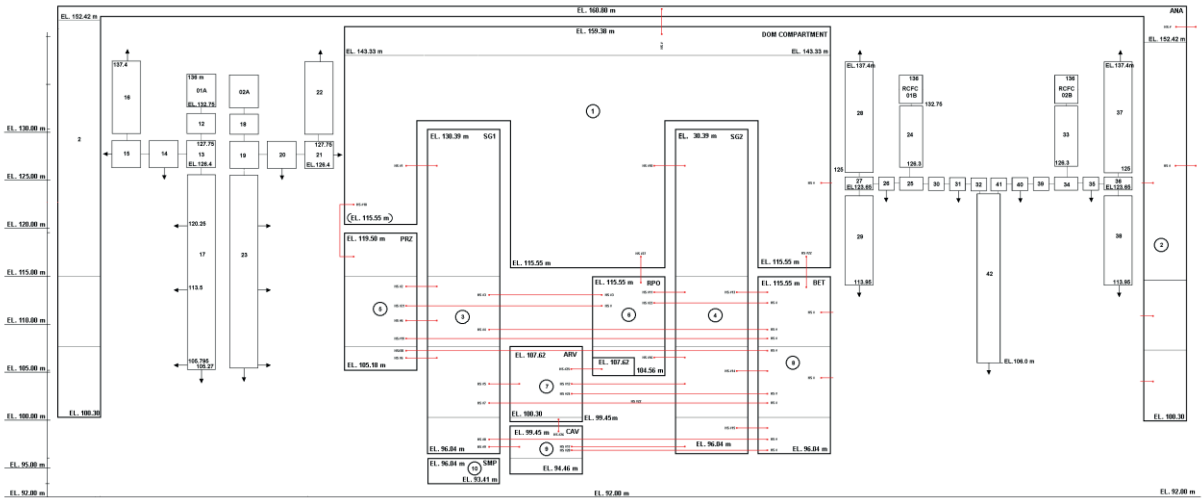

Figure 6: GOTHIC multivolume containment model

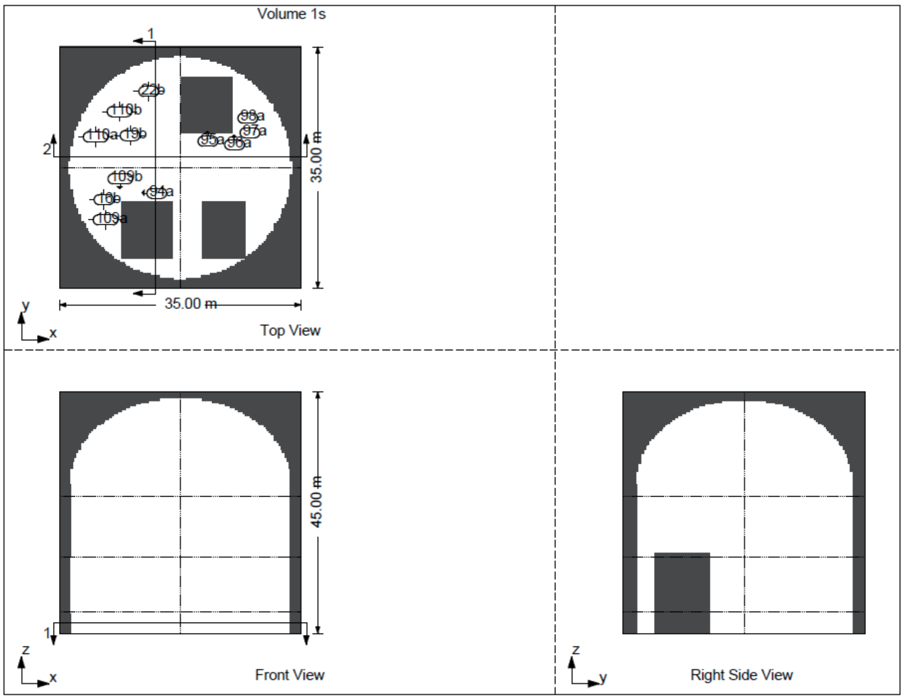

Figure 7: GOTHIC containment dome subdivision

Following scenarios were covered by single compartment model:

- spray switched off at 1day after break (EQ assumption), label: spray off 1 day,
- spray switched off per EOP (containment pressure below 1.6 kp/cm^2 gauge, 258.23 kPa), if before recirculation and shorter than minimum 2 hours per NUREG-0800 than limited to 2 hours, label: spray off 2 h,
- spray off, limiting case without spray after recirculation, label spray off,
- spray on, limiting case with spray on all the time, label: spray on.

For analysed sequence time to recirculation was 3292 s.

Containment pressure and temperature for different spray operation time are shown in Figure 8 and Figure 9, respectively. Up to the point of recirculation there is no difference in response, then both pressure and temperature start to increase after spray switch of. All values above profile obtained in original EQ calculation (spray off 1 day) can mean some invalidation of EQ envelopes. Pressure and temperature peaks are not affected due to their time position.

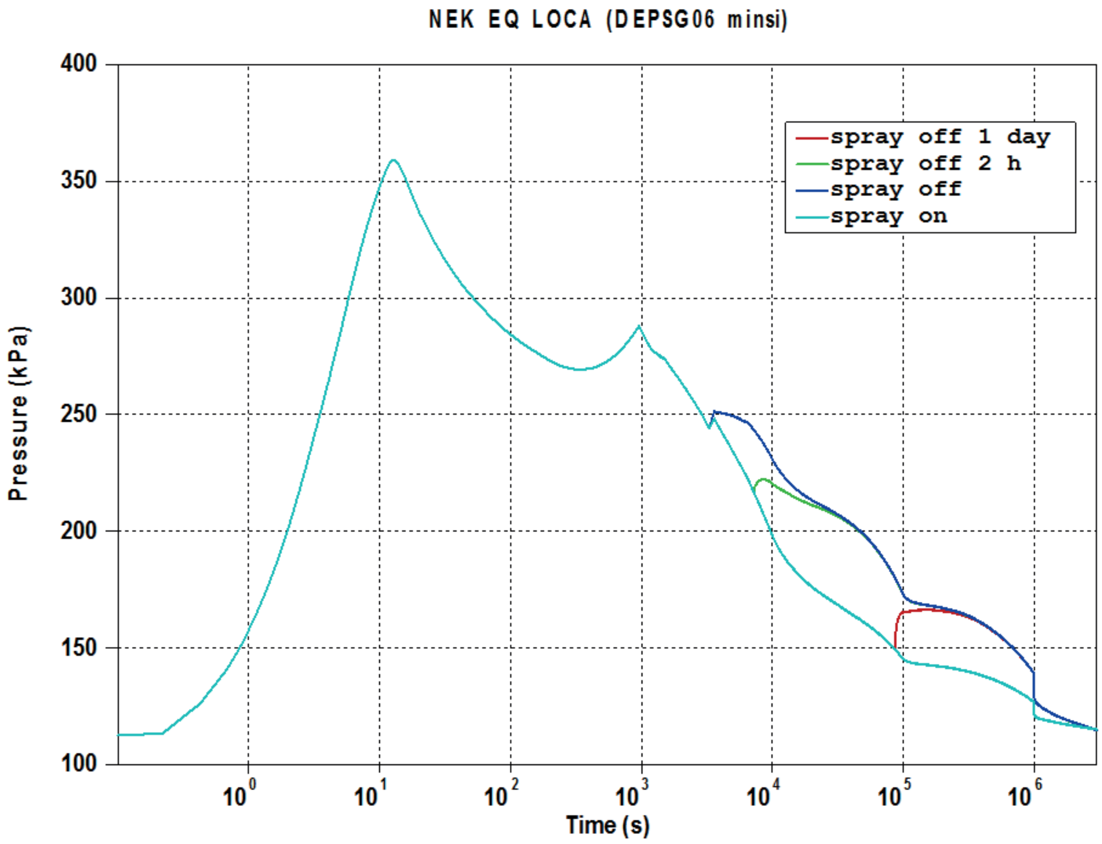

Figure 8: Pressure in containment, EQ LOCA

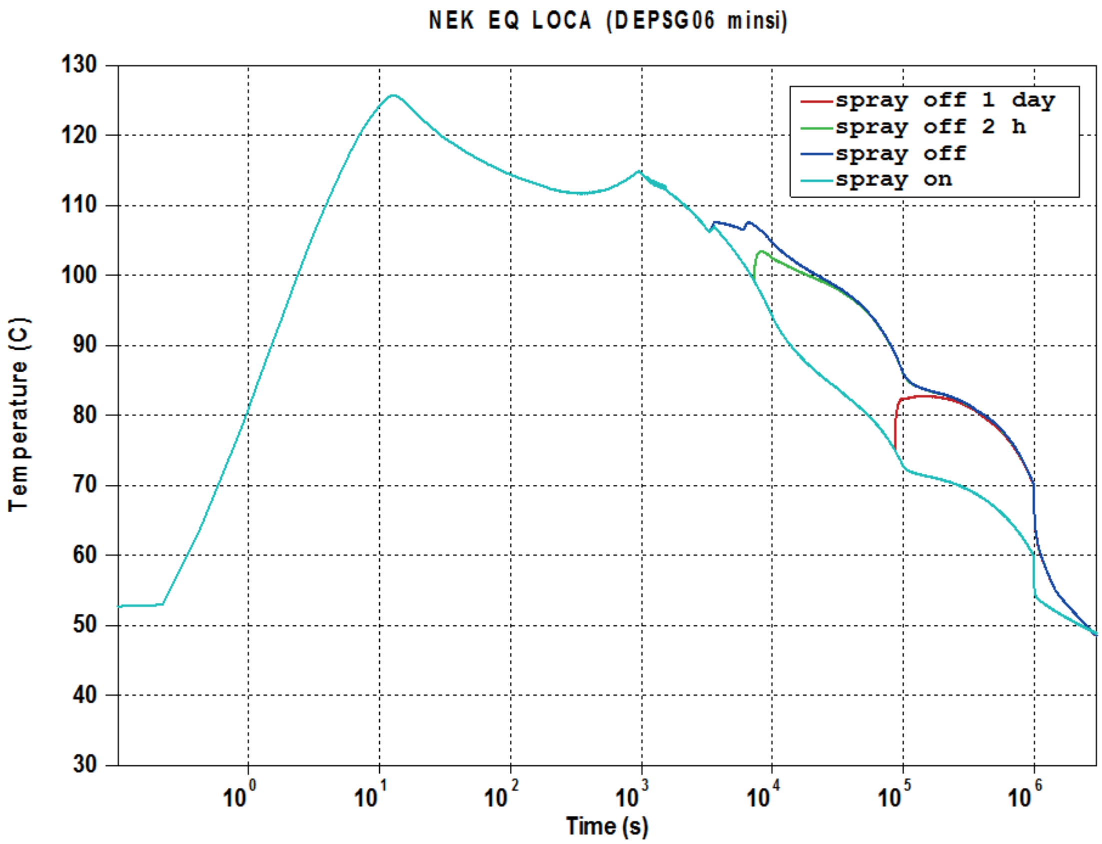

Figure 9: Temperature in containment, EQ LOCA

3.2 DBA LOCA Calculation, multi compartment model

For multi compartment containment model the same four cases are calculated as for single compartment model, but simulation time was up to 1.e5 s. The break location was in SG1 compartment and in free space above operating deck (to simulate situation close to the break within only volume in simple model). Only results for cases without spray actuation in recirculation were shown. Other cases results show similar trends. Only results within 16 containment dome volumes were used to follow the same logic as in single compartment model. In rest of the containment volumes the temperatures can be both lower and higher than in containment dome depending on location of the break. Pressure distribution is always rather uniform. Figure 10 shows pressure in lowest and highest subvolume of the containment dome (multi compartment model), for break in SG1 compartment and in containment dome, as well as referent single compartment pressure. Pressure response in multi compartment model is rather uniform except in early beginning of the transient. Both multi compartment responses give higher pressures than single compartment model (smaller effective free volume and condensation surfaces in multi compartment model). For break in containment dome due to similar reasons initial pressure increase is faster as well as later pressure decrease. Figure 11 shows gas temperatures for the same situation. Temperature distribution between subvolumes of multi compartment model is not uniform. Initial peak temperatures are again higher in multi compartment model than in single compartment model. Long term behaviour of the temperature for the break in SG1 is closer to the temperature behaviour in single compartment model what was not expected. The condensation heat transfer model is typically more effective in multi compartment model. The results with spray actuation in recirculation model show similar trends in multi compartment and single compartment model.

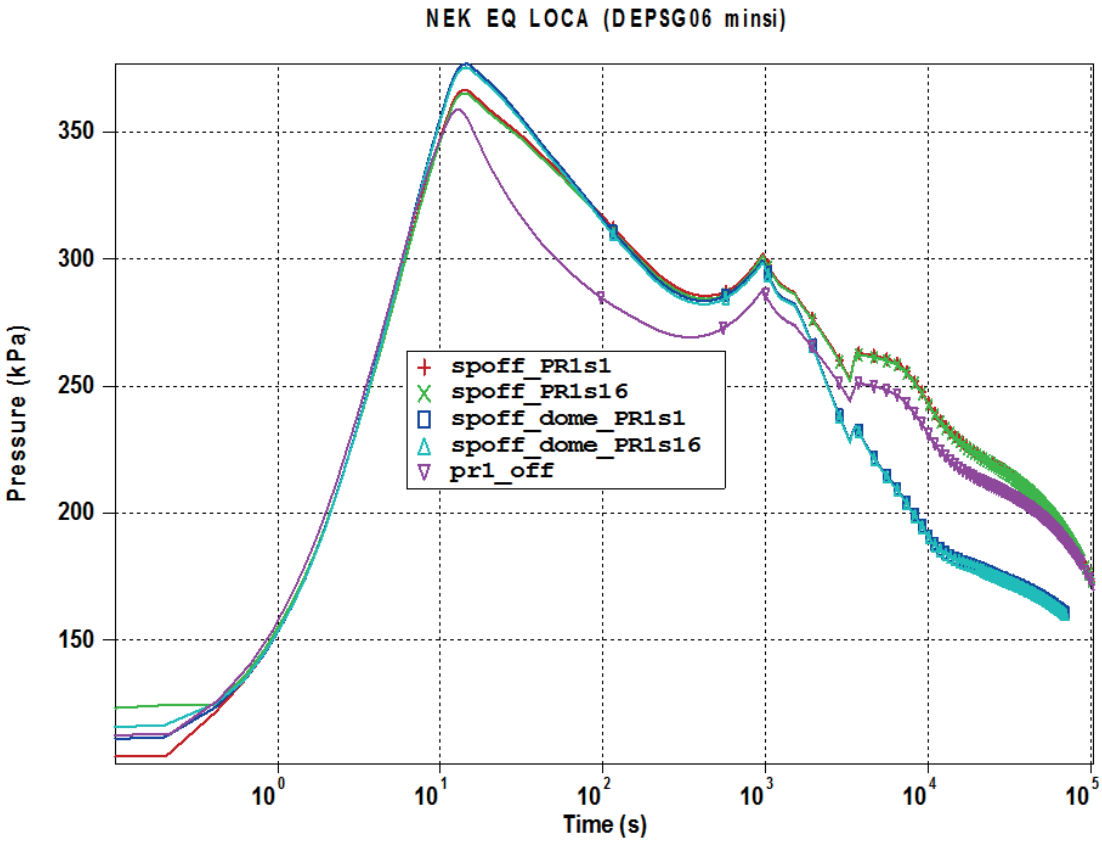

Figure 10: Pressure in containment upper plenum, EQ LOCA

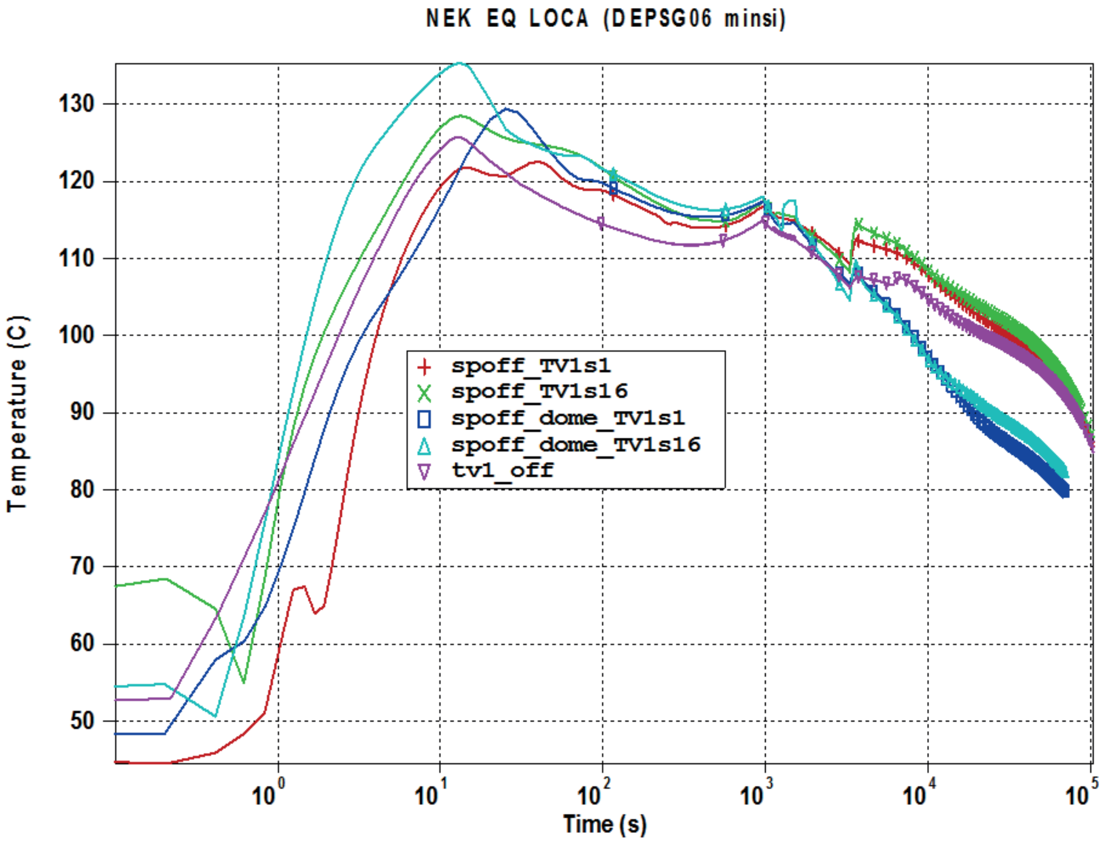

Figure 11: Temperature in the containment upper plenum, EQ LOCA

4 CONCLUSION

As part of initial verification of upgraded multi compartment GOTHIC model containment DBA LOCA calculation is performed using SAR MER (Mass and Energy Release) data for both single and multi compartment model. Beside local effects due to different containment subdivision, similar results are obtained when comparing containment dome values from multivolume and the single volume values from licensing model. Local redistribution of, mostly temperatures, is present when break location is changed in multi compartment model.

Another intention of the paper was to study consistency of assumed different spray operation times used in safety analyses, EQ analyses and SAMGs. Due to different assumed spray operation times (1 day in EQ, shorter for EOPs), based on the obtained results for spray influence, all values above profile obtained in original EQ calculation (spray off 1 day) mean some invalidation of calculated EQ envelopes (higher pressure and temperature values). Pressure and temperature peaks are not affected due to their time position.

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External Reactor Vessel Cooling Evaluation for Severe Accident Mitigation in NPP Krško

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ABSTRACT

The In-Vessel corium Retention (IVR) through the External Reactor Vessel Cooling (ERVC) is mean for maintaining the reactor vessel integrity during a severe accident, by cooling and retaining the molten material inside the reactor vessel. By doing this, significant portion of severe accident negative phenomena connected with reactor vessel failure could be avoided.

In this paper, analysis of NPP Krško applicability for IVR strategy was performed. It includes overview of performed plant related analysis with emphasis on wet cavity modification, plant's site specific walk downs, new applicable probabilistic and deterministic analysis, evaluation of new possibilities for ERVC strategy implementation regarding plant's post-Fukushima improvements and adequacy with plant's procedures for severe accident mitigation.

Conclusion is that NPP Krško could perform in-vessel core retention by applying external reactor vessel cooling strategy with reasonable confidence in success. Per probabilistic and deterministic analysis, time window for successful ERVC strategy performance for most dominating plant damage state scenarios is 2.5 hours, when onset of core damage is observed. This action should be performed early after transition to Severe Accident Management Guidance's (SAMG). For loss of all AC power scenario, containment flooding could be initiated before onset of core damage within related emergency procedure. To perform external reactor vessel cooling, reactor water storage tank gravity drain with addition of alternate water is needed to be injected into the containment. ERVC strategy will positively interfere with other severe accident strategies. There are no negative effects due to ERVC performance. New flooding level will not threaten equipment and instrumentation needed for long term SAMGs performance and eventually diluted containment sump borated water inventory will not cause return to criticality during eventual recirculation phase due to the lost core geometry.

Keywords: *in-vessel corium retention, external reactor vessel cooling, severe accident management*

1 INTRODUCTION

If a severe accident involving core damage is not arrested, at some point relocation of molten core material into the lower plenum of reactor vessel will occur. Reactor vessel integrity can be maintained by performing external reactor vessel cooling by retaining the molten material inside the reactor vessel, therefore avoiding significant severe accident negative phenomena connected with reactor vessel failure, such as:

- evaporation of water which is in contact with molten core debris which could result in containment overpressure;
- generation of additional flammable gases, as a consequence of molten core - concrete interaction (MCCI), which could threaten the containment;
- non-condensable gaseous buildup, as a consequence of MCCI, which could result in containment overpressure;
- additional radioactive aerosol production;
- reactor cavity steam explosions;
- basement floor concrete ablation, as a consequence of MCCI, which can threaten containment integrity;
- direct containment heating as a result of high pressure melt ejection of corium; and
- other accident phenomena connected with degradation of overall plant capabilities to mitigate the post-accident site releases and to restore controllable accident condition.

In case when ERVC performance is not successful to prevent reactor pressure vessel (RPV) failure, it will slow down boil off of reactor inventory, thus delaying the time of vessel failure. Gaining time could be crucial because it may be able to restore failed equipment back to service.

2 EVALUATION OF ERVC APPLICABILITY FOR NPP KRŠKO

An evaluation of current evidence, site specific analysis, and area of improvement in equipment and guidelines for IVR as an accident management strategy for NPP Krško will be presented. Also an overview of ERVC strategy for severe accident will be made. Areas of interest are:

- evaluation of NPP Krško current possibility for external reactor vessel cooling as a mean for severe accident mitigation, and
- eventual improvement of current "wet cavity" design and overall plant capabilities during accident mitigation.

Results of performed NPP Krško reactor cavity walkdown will be discussed and incorporated into the overall findings and conclusions.

2.1 Description of external reactor vessel cooling

The external reactor vessel cooling (ERVC) goal is to cool the RPV lower head from the outside. The objective is to prevent the RPV failure. As long as this objective is met, core debris remains inside RPV and therefore limits containment loads during severe accident. The goal can be satisfied by timely submerging the RPV lower head to a height some margin above the level which the core debris will reach inside the vessel following relocation from the original core boundaries. Nucleate boiling occurring on the outside vessel wall following core melt relocation will then remove sufficient heat from the debris to prevent vessel wall melt through.

Any actions required to achieve the flooding must be performed within a time window defined by the time interval between the start of the SAMG operations (i.e., Core Exit Thermocouple (CET) temperature > 650 °C (925 K)) and the predicted time of lower support plate failure, for the severe accidents under consideration.

In the long term, it is also necessary to compensate for the steaming due to the decay heat. For successful ERVC, focusing effect is on contact of metallic pool layer with RPV wall. It is expected that RPV wall on position 90 degrees will have the highest temperature and the creep deformation will lead to wall thickness reduction with partial wall melting. It will be potentially failure position of RPV, as seen in Figure 1. The focusing effect can be much reduced if the upper face of the metal layer is cooled on top (an example is injection into the core, spraying the upper portions of RPV from outside, or hot gases circulation with partial heat transfer to steam generators).

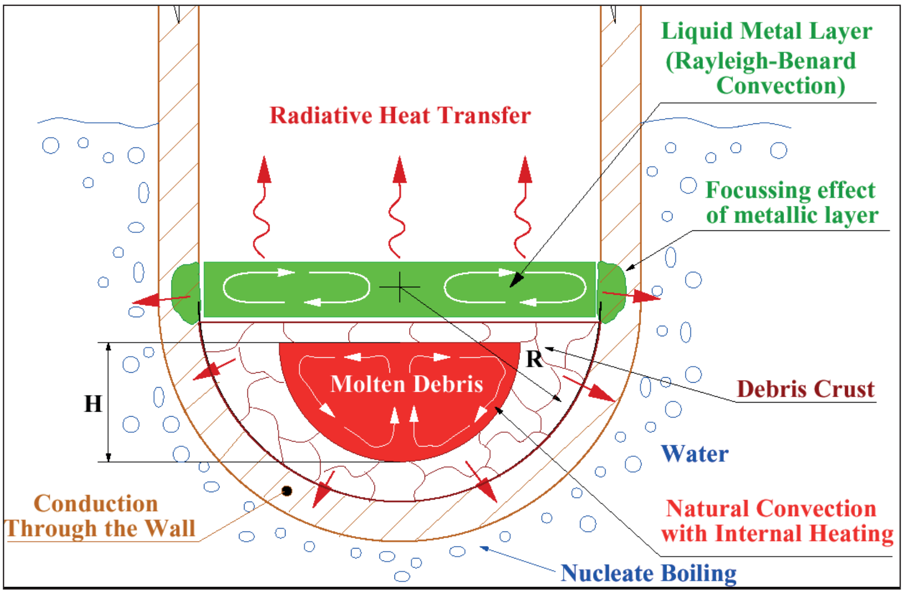

Figure 1: Basic principle and heat removal process of ERVC [1]

Once as successful ERVC strategy of vessel cooling has established IVR, heat removal from the containment stays crucial for long term severe accident mitigation.

2.2 History of ERVC development for NPP Krško

IPE Level 2

Summarized NPP Krško Individual Plant Examination (IPE) Level 2 [2] results based on the studies, experiments and analysis [3], represented that by performing ERVC strategy the heat transferred by nuclear boiling on the outer head wall surface can remove a large amount of heat, and that may be enough to prevent failure of the vessel wall due to melted corium thermal attack during the severe accident.

It was also concluded that if the reactor vessel could be timely flooded, this measure could prevent vessel failure in case of severe accident with relocated core. More practical considerations for NPP Krško included in [2] were:

- proposal of "wet cavity" modification at the plant by allowing free communication of water from the lower compartments to the RPV cavity, which will be beneficial to cool the debris if the vessel will fail, and
- with "wet cavity" design injection of total RWST volume would not flood the lower portion of the Krško vessel. Therefore, an accident management strategy to attempt to prevent vessel failure by IVR would require additional water to be injected into the containment than expected due to safeguards system operation.

In view of mounting evidences, it was recommended that NPP Krško considers vessel flooding as an ERVC severe accident strategy at a future date.

NPP Krško reactor cavity flooding evaluation

In the Reactor Cavity Flooding Evaluation Report [4] it was investigated and justified the flooding of the region below the reactor vessel as a means to mitigate a severe accident, and at the same time ensuring that such flooding will not have negative impacts on design basis accidents or on normal operations. The two main goals of RPV cavity flooding function were:

- containment floor concrete protection, and
- external RPV cooling.

An additional goal can is achieved when satisfying the main goals: scrubbing of fission product aerosols released from ex-vessel core debris.

NPP Krško wet cavity modification

Based on findings from [4], NPP Krško performed during 2001. modification 347-FD-L “Containment sump check valve removal”, where wet cavity design is adopted by simple removal of check valve, as seen in Figure 2. The modification requirements included external reactor vessel cooling analysis.

The objective of performing this modification would be to mitigate the consequences of a potential severe accident by:

- ensuring the presence of water in the cavity in the event of reactor vessel failure and core debris transport to the reactor cavity, to quench and cool the core debris, and thereby prevent the occurrence of long term molten core-concrete interactions,
- flood the outside of the reactor vessel before core melt relocation to the lower head, and thereby potentially prevent the failure of the reactor vessel, and
- to ensure an overlying water layer if core debris does enter the containment, to scrub fission products released from the debris.

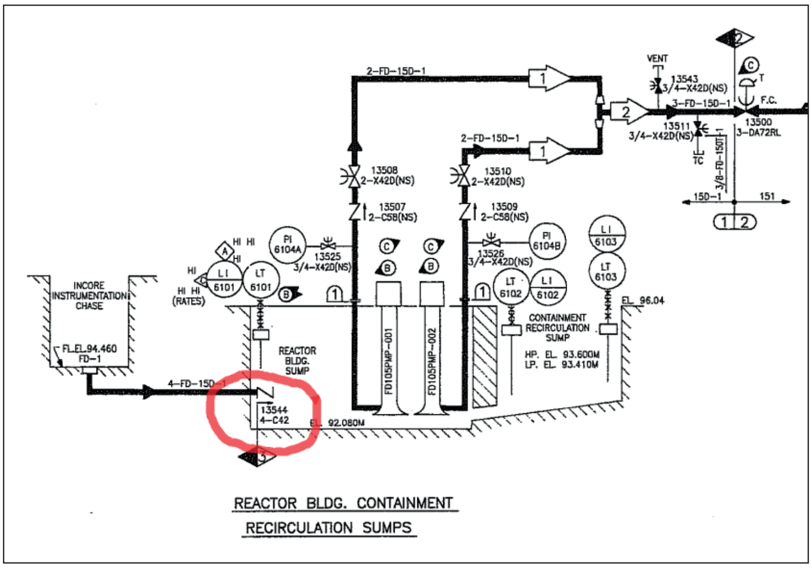

Figure 2: Containment sump check valve removal by plant modification 347-FD-L

Regarding that during wet cavity modification RWST was only source for containment injection, and analysis limitations for timely performance of ERVC strategy were applied, the RPV external cooling strategy was not implemented, despite deterministic benefits. Modification was accepted only for cavity floor concrete protection. See sections 3.8 and 3.9.

3 REQUIREMENTS AND LIMITATIONS FOR ERVC STRATEGY PERFORMANCE

3.1 Containment water level needed for IVR

The cavity water level shall be at least at the melted core level inside the RPV which is 1m above the RV Lower Head Bottom (inside RPV) plus a margin of 0.5 meter (see Figure 3).

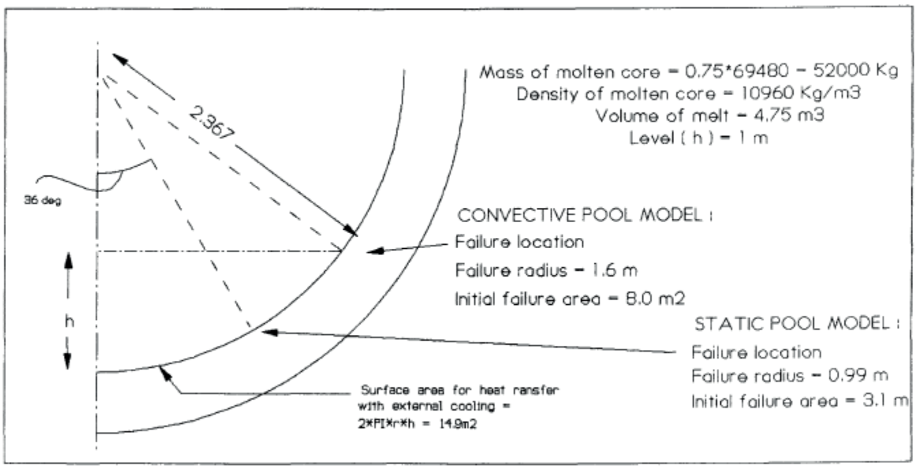

Figure 3: NPP Krško in-vessel corium pool geometry calculation [3]

This corresponds to a water height at elevation 99.2 m or approximately 152 m^3 of water in the cavity, or 1440 m^3 of water in the containment if the cavity and sump are connected.

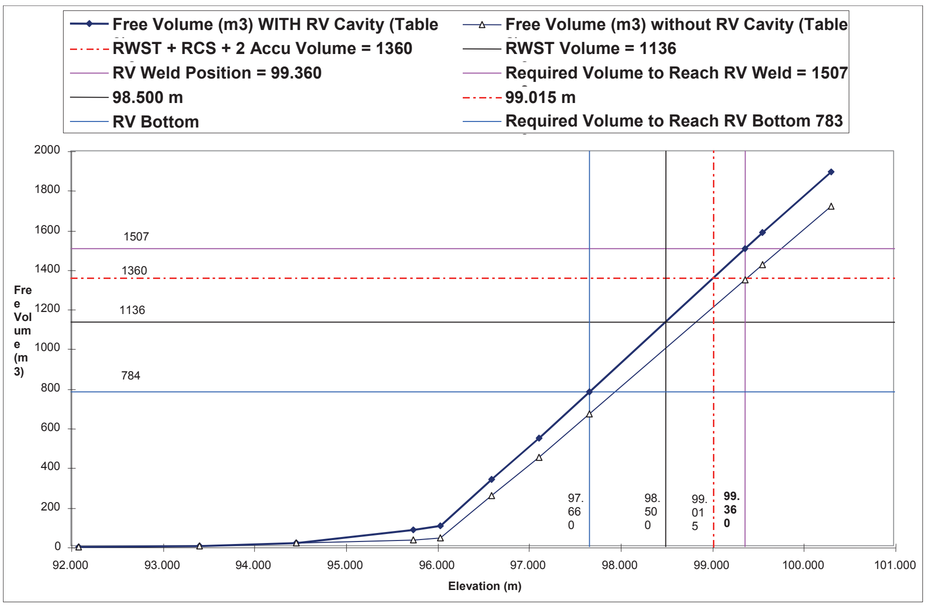

Figure 4: NPP Krško RB flooding level evaluation [4]

In Figure 4 and Table 1 are represented NPP Krško containment water volumes, elevations and plant instrumentation measured levels. Information's are collected from [4], [9], and [10].

Table 1: NPP Krško containment volumes, elevations, measured water levels

Level / RB plant elevation	**Related minimum volume**	**LI 6102/ LI 6103**	**Description (sp. - setpoint)**
92.080 m	-		RB sump bottom
93.410 m	-		Containment sump bottom
93.560 m	-	0 m	Containment recirculation sump level - bottom LI 6102 / LI 6102
94.460 m	-	0.9 m	Cavity floor bottom
95.500 m	75 m^3	1.94 m	Cavity concrete protection (SP – 2 m)
97.110 m	-	3.55 m*	Minimum recirc. sump operability water level *with (+) 0.3 m for EOP / SAMG sp.= 3.9 m
97.660 m	784 m^3	4.10 m	Bottom of the RPV
98.500 m	1136 m^3	4.94 m**	1. RWST useful volume 2. Flood level elevation FR-Z.2 **with (-) 0.3 m for EOP sp.= 4.6 m Ventilation opening to cavity (bottom)
99.015 m	1360 m^3	5.45 m	RWST + RCS + 2 SI ACC water volume Ventilation opening (top)
99.160 m	1440 m^3	5.60 m	RPV external cooling SP
99.360 m	1507 m^3	5.80 m	RPV weld
99.560 m	-	6 m	Containment recirculation sump level - top LI 6102 / LI 6103

3.2 RPV cavity flooding flowpath

Currently reactor cavity flooding in NPP Krško is performed through one 4 inch floor drain line (Figure 5).

Figure 5: RPV cavity floor drain opening protected with sieve (marked)

Afterwards, when containment level is sufficient high, water can enter through reactor compartment ventilation ducts (Figure 6). Therefore there are no flow limitations for timely performance of the cavity flooding strategy regarding containment to reactor cavity compartment injection.

Figure 6: Reactor compartment ventilation entry

3.3 Containment equipment flooding limitations

Regarding beyond design basis accidents and already adopted NPP Krško SAMG strategies [18], equipment flooding is predicted per SAMG [8] (see Table 2), where detailed information of potential effected equipment were made (as seen in Table 1).

Table 2: NPP Krško SAMG, SAG-8 Attachment "Loss of equipment and instrumentation", detail

Negative Impacts for Injecting Into the Containment (equipment and instrumentation location in RB below elevation 105)				
TAG NUMBER	**DESCRIPTION**	**TYPE**	**ELEVATION**	**MEC**
TE127	REGEN HX LETDN RTD	ELE	98.16	1
TE229	EXCES LETDN HX RTD	ELE	98.25	2
FE167	RCP 2 #2 SEAL ORIFICE	ELE	98.35	2
FE1008	RC DRN TANK DISC ORIF	ELE	98.63	2
FT1008	RC DRAIN TNK DISC FT	XMT	98.63	2
TE6530C	REACT SUPPORT PAD HI TEMP RTD	ELE	100.30	02B
TE6530D	REACT SUPPORT PAD HI TEMP RTD	ELE	100.30	02B
LT6102	**CNTMT RECIRC SUMP LT**	**XMT**	**100.30**	**13**

Affected instruments by additional flooding over el. 98.5 m (FR-Z.2) [21], [10] are part of reactor coolant drain tank (RCDT) flow measurement instrumentation, which are not necessary for containment sump recirculation, and are not essential for any further SAMGs strategy performance.

3.4 Pressurized Thermal Shock (PTS) issue for reactor vessel

In order to assure structural integrity of the Krško RPV for a postulated external flooding event, a stress and fracture mechanics analysis was performed (ref. [3], [4]), reflecting enveloping conditions in terms of internal pressure and temperature, and external temperature. The structural analysis focused on stability of postulated defects using fracture mechanics methods that are typically applied to demonstrate RPV integrity under PTS (Pressurized Thermal Shock) type of loading. Conclusion from [4] is that plant modifications, which allows vessel flooding to mitigate a severe accident, could not lead to PTS caused catastrophic vessel failure in case of vessel flooding during normal operation or design basis accident (DBA).

3.5 Reactor vessel insulation water ingression

NPP Krško IPE documentation suggests that NPP Krško reflective reactor vessel insulation would not impede the ingression of water needed for successful ERVC. The experiments referenced in plant specific IPE Level 2 [3] and EPRI [11] also show no effect of reflective insulation, with sustained nucleate boiling being maintained in cases performed with and without insulation.

Figures 7, 8 and 9, represents reactor cavity walkdown photos of NPP Krško reactor vessel bottom head insulation details.

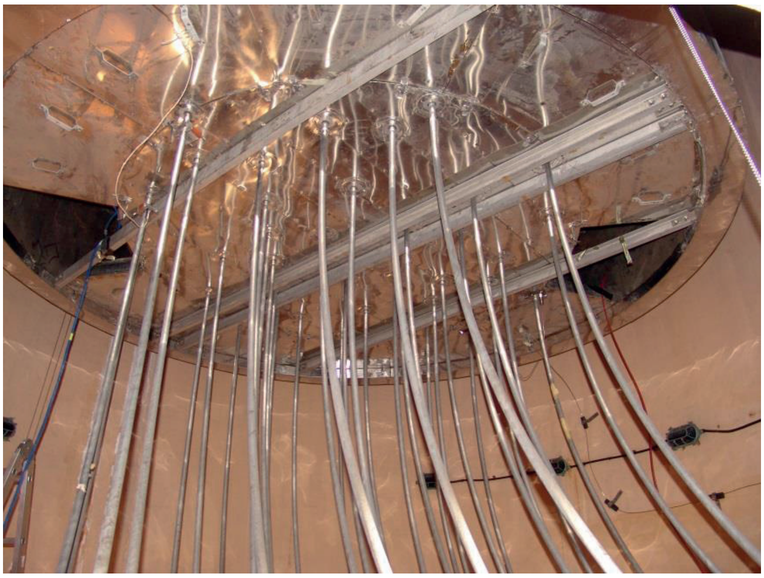

Figure 7: NPP Krško RPV insulation details (1) - inspection openings

Document Final Independent Review of NPP Krško Design Modification Package [14] reviewed plant modification 347-FD-L "Containment sump check valve removal". The conclusions are in accordance with findings from [4]. This report as a new input defines minimum water in-flow for successful ERVC strategy performance (8 m^3/hr). According to [3], various experimental evidences exist that insulation would not impede the ingression of water because it is not watertight.

Figure 8: NPP Krško RPV insulation details (2) – incore penetrations

Figure 9: NPP Krško RPV insulation details (4) – insulation - concrete gap

Regarding plant specific IPE Level 2 analysis [2], and EPRI SAMG TBR generic documentation [11] – additional insulation openings for NPP Krško are not needed.

3.6 Steam explosions

Steam Explosions phenomena for NPP Krško are addressed in phenomenological evaluation [3], and site specific report [24]. Approaches to the issue of steam explosions which have been used in various analyses have also been reviewed.

Based on the reviews, evaluations and caluculated results, ex vessel steam explosion, as a result of eventual vessel failure into the flooded reactor cavity, will cause no additional challenges to the containment integrity since:

- for containment pressurization due to steam generation, the potential for steam explosions has no impact, and
- ex vessel steam explosion shock waves during eventual vessel failure scenarios pose neglible threat to containment integrity.

3.7 Recriticality during severe accident conditions

Partially diluted or unborated containment sump water inventory, will not cause return to criticality of molten core debris during in-vessel recirculation phase or eventual ex-vessel quenching, because of lost core geometry [7].

3.8 Time window requirements for ERVC strategy performance

For successful ERVC strategy performance, flooding must be performed within a time window defined by the time interval between the start of the SAMG operations (i.e., Core Exit Thermocouple temperature > 650 °C) and the predicted time of lower support plate failure.

The time window was identified as follows [4]:

- Severe accident sequences to be considered are identified based on Plant Damage States (PDS). Damage states participating for the most dominating Core Damage Frequency (CDF) are considered for the time window calculation.

– For most dominant damage states, where ERVC usage is reasonable, the accident sequence analysis from the level 2 study was used to determine the time window between the Core Exit Thermocouple reaching 650 °C and the predicted time of core lower support plate failure (corresponding to the time of core melt relocation to the lower head).
– The minimum time window for most dominant PDS is selected. The results of this process are summarized in Table 3.

Table 3 represents MAAP calculated NPP Krško PDS scenarios with dedicated time for ERVC. Calculated time for ERVC performance (criterion per [4]), for successfully strategy performance is 2.5 hours for TEHNNN plant damage state (PDS) scenario (58% of total CDF participation - as was used for design of passive containment filtered vent system (PCFV) [20]).

Table 3: NPP Krško most dominant PDSs applicable to ERVC, with calculated time [15]

Most dominant PDS contributors to overall CDF	Frequency for NPP Krško PDS, summ. regarding releases	Core uncovered (CET>650 C*) (sec.) (A)	Core relocation begins (sec.) (B)	RV failed, (sec.) (C)	**Minimum time for ERVC performance 1) (B) - (A) [4] hours (h)**
TEHNNN	58,18%	6463	15472	21767	**2,5 h**
TEHANN	7,46%	6463	15472	21767	2,5 h **
TEHAYN	5,45%	1920	9183	13429	2,0 h
UXXXXB	2,95%	6680	16096	20546	2,5 h
SELAYN	2,76%	15298	31205	37579	4,2 h
WUUUUB	0,90%	69529	79687	84700	2,7 h

* After core uncovers, roughly 250 to 500 seconds (depending on overall scenario length) is needed that CET rises from 370 deg. C (saturation water temperature), to 650 deg. C (SAMG entry). Assumption is based on similar results from [4].

** TEHANN scenario differs from TEHNNN regarding containment injection after vessel failure. Therefore TEHANN results for in-vessel accident progression are the same as TEHNNN.

It should be noted that first Reactor Cavity Flooding Evaluation Report [4] (with too conservative result with time window for vessel flooding of 30 minutes) took into account only PSA results for specific, worst case (fastest) PDS scenario, and this was chosen to give a judgment for adoption of ERVC strategy. Deterministic benefits from appliance of ERVC strategy were not used. Meanwhile, NEK PSA model [15] was updated with initiating events for: internal flooding, internal fires, seismic events, high energy line breaks (HELB), and other external events. Recent codes calculation of accident scenarios are using different severe accident phenomenology, where some of effects have changed the accident sequences and overall results (as hot leg creep rupture effect which happens before high pressure vessel failure, which enables safety injection accumulators to be injected into the RPV, thus postponing time of vessel failure).

For further development, if early depressurization of RCS is performed before core damage (or when it is noted that core damage is imminent), or when still in EOP procedure, for TEHNNN PDS scenario for the time from CET > 650 °C and the predicted time of lower support plate failure extends time window for ERVC performance for additional 1 hour. Also, with early EOP flooding action, ERVC successful performance time window per criteria [4] could be extended to at least 4 hours [25].

3.9 **Curent plant availability for ERVC performance**

For ERVC performance water inventory needed is 1440 m^3, to cover bottom of reactor vessel up to elevation 99.2 m (or 5,6 m on containment recirculation sump level indication).

Current Refueling Water Storage Tank inventory (RWST) inventory (1250 m^3), together with spilled RCS and SI ACC (226 m^3), is not sufficient for performance of IVR strategy. Nevertheless, after performing post-Fukushima short term improvement plant modifications [5], [6], NPP Krško is available to inject additional water (see Table 5) into the containment through alternate flowpaths (containment spray lines, ECCS flowpath, RCP fire protection spray lines), including mobile equipment usage.

Injecting RWST up to main control room “RWST Empty” alarm (18%) [21] will inject 1110 m^3 of water. Basis for that alarm is that RWST vortexing regarding ECCS PMP operation will be prevented (if later in the accident they will become available). Also, for core damage to occur, core should remain “dry”, and it can be supposed that RCS has been spilled via PRZR safety / relief valve or through the RCS break (assuming the brake from RCS is spilled into the containment). RCS spilled total volume is roughly 226 m^3, including the accumulators. Together with RWST it would be 1360 m^3. Nevertheless by conservative assumption that RCS may spilled through IS LOCA or through SG tube rupture, this volume will not be taken into the account for ERVC purposes for flooding the containment.

NPP Krško current means for containment injection are:

- RWST gravity drain (1250 m^3) or RWST injection (1010 m^3) with ECCS or Containment Spray System to “RWST Empty” alarm (18%)

  AND

- additional 190 m^3 (430 m^3) needed to be injected by severe accident management equipment (SAME):
  - mobile AE900PMP-001/002 (FOX3) pump with capacity 96 m^3/h at 10 bar,
  - mobile AE900PMP-007 (HS450) pump with capacity 660 m^3/h at 12 bar.

It should be noted that for Extended Loss of AC Power (ELAP), containment flooding strategy by RWST gravity drain is currently performed within EOP “Loss of all AC power” procedure (before core damage). Applicable water sources are in Table 4.

Table 4: NPP Krško available water sources for inject into the containment (SAMG / EOP usage)

Source	Capacity	Water quality
RWST tank	1250 m^3	borated
BAT tank (2)	51 m^3	borated
WT tank 01	379 m^3	demineralized
WT tank 02	1000 m^3	demineralized
CY tanks (2)	879 m^3 each	demineralized
PW tanks (2)	1000 m^3 each	demineralized
FP tank	235 m^3	raw water
City water	unlimited	raw water
Condenser hotwell	not defined	raw water
CW tunnel	not defined	raw water
Sava river	unlimited	raw water

4 SEVERE ACCIDENTS MEASURES UPGRADE REGARDING ERVC PERFORMANCE

Severe accident assumes occurrence of a meltdown of the core, breaching the first barrier of the clad to release the radioactive fission products. If the accident progress further, the molten corium moves to the bottom head of the vessel. The bottom head will fail if the corium melt remains uncooled, thereby failing the second barrier to the release of radioactivity to the environment. The corium melt released to the containment may fail the containment in a short time if some energetic reactions, for example, hydrogen burn (detonation), steam explosion, or if direct containment heating occurs. If such energetic interactions do not occur, or are managed not to occur, the containment could fail later (by several hours or few days) due to the attack of the core melt on the concrete, which would release the non-condensable gases pressurizing the containment and possibility cause the melt-through of the basement. The containment structural failure represents failure of the third and the last barrier to the release of radioactivity to the environment.

Severe accident management guidelines (SAMGs) consist of actions (measures) that would prevent the failure of the barriers 1 (cladding), 2 (reactor pressure vessel) to 3 (containment). The first aim of SAMG is to prevent damage to the clad on the uranium fuel pellets.

This should be done by operator actions from severe accident control room guidance initial response, or after Technical Support Centre (TSC) becomes operable - by injecting into the secondary side, by depressurizing RCS and by injecting into the reactor vessel [8].

If that is not possible due to the inability to timely inject water to the vessel, the second barrier protection aim of SAMGs should become the prevention of the bottom head of the RPV by performing ERVC strategy. If that aim is not achieved due to either the inability of inject water to the vessel to quench and the melt pool in the lower head; the next (third) aim becomes prevention of the failure of the containment due to gaseous buildup, and/or the basement melt-through so that there is no significant release of radioactivity to the environment.

NPP Krško uses as a last mean for containment protection SAMGs severe challenge guidelines where usage of dedicated equipment is introduced from Phase 1 Safety Upgraded Project (SUP) implementation [6], [8]:

- Passive Autocatalytic Recombiners (PAR) - for mitigation of containment severe challenge due to buildup of flammable gases in containment, and
- Passive Containment Filtered Vent System (PCFV) for mitigation of containment overpressure severe challenges.

4.1 NPP Krško EOP Upgrades

Accident sequence with high contributing core damage frequency in PSA is loss of AC power initiator [20]. First procedurally containment injection action is in plant Emergency Operating Procedure (EOP) ECA-0.0 “Loss of All AC Power [21]. The basis is to establish RWST gravity flow to containment early enough to flood RPV if is evident that all attempts to establish decay heat removal by injecting into the secondary or primarily side are not successful and that core uncover is imminent. Also, benefit of that action is in the fact that containment pressure is low enough that RWST gravity drain is available. Decision for performing that action is made by Technical Support Center.

Also as a part of further SUP project there will be additional availability for early RCS depressurization with new Pressurizer PORV bypass valves [25]. Therefore, early RCS depressurization could prolong vessel failure for additional 1 hour.

So, by performing early RCS depressurization and containment flooding when there are no alternative means for beyond design basis accident mitigation, there is significant improvement in current plant chances for successful ERVC performance.

4.2 NPP Krško SAMG Upgrades

Currently, NPP Krško does not use ERVC strategy for in-vessel melt retention. Nevertheless early containment flooding strategy for reactor cavity flooding is adopted in NPP Krško SAMG's [8] for reactor cavity floor concrete protection to mitigate consequences of eventual vessel failure where MCCI could occur [4]. Means for containment flooding are usage of containment spray, injection through eventual RCS openings, or if AC power is lost - RWST gravity drain, as well as alternate mobile pumps, additional water sources with alternate containment injection flow paths for containment injection.

Regarding further development of generic SAMG's, containment flooding strategy to perform ERVC will also be used [26]. Also development of "Severe Accident Control Room Guideline - Loss of DC and Instrumentation" [24], proposes integrated RCS and containment flooding strategy for prevention of RPV failure, by application of ERVC strategy if core condition is unknown.

Main control room initial response guideline (SACRG-1)

Currently, NPP Krško SAMGs guideline SACRG-1 [8], or main control room (MCR) initial response to severe accident before technical support center (TSC) is activated, performs early containment flooding strategy to assure reactor cavity flooding up to recirculation sump level of 2 m [9]. Basis is reactor cavity basement floor protection [4]. Anyway, due to practically reasons, the operators are instructed to flood up to recirculation sump level of 3.9 m, or EOP / SAMG setpoint [10]; to establish containment sump level for strainers operability for usage of containment spray or emergency core cooling system in recirculation mode. Other actions, including attempts to inject into the RPV are followed later.

For timely performance of ERVC strategy, SACRG-1 guideline is most suitable. Although RWST gravity drain is possible during performance of Loss of All AC Power procedure (as TSC evaluated action for extended loss of AC power condition), certain containment flooding as a operators action will be performed during SAMG's SACRG-1 guideline for MCR.

Usage of alternate provisions (SAME equipment) and water sources will be needed, to fill containment with 1440 m^3. Beside 1250 m^3 of RWST inventory, additional 190 m^3 could be needed from other sources. Note that RCS and SI ACC inventory (which may or may be not spilled in sump) is not taken into the account (maximum 225 m^3 of inventory).

Therefore, if attempts to inject into the RPV had failed, or injection flow is insufficient to remove decay heat, suggestion is that SACRG-1 direct operators to establish ERVC to prevent / delay vessel failure, by injecting into the containment with alternative provisions.

Technical support centre severe accident guidelines (SAG's)

NPP Krško SAMGs guidelines [8], SAG-4 "Inject into containment" and SAG-8 "Flood the containment", performs strategy of injecting water into the containment by using all available means, as directed by technical support centre (TSC) evaluators.

The current purposes of injection into the containment are to [8]:

- prevent or mitigate the consequences associated with core-concrete interactions,
- scrub fission products released from ex-vessel core debris,
- allow ECCS recirculation (long term containment heat removal), and
- perform external cooling of RPV lower head (SAMG's generic [17]).

SAG-4 is suitable for performing ex vessel cooling strategy, even if it comes somehow late per SAMGs diagnostic flowcharts. If previous SAG-s strategies are not successful, flooding up to new level of 5.6 m will also allow more confidence for retaining core inside the vessel, even if unsufficient

RCS injection flow is established within SAG-3 “Inject into the RCS”. EPRI SA TBR [7], notes that insufficient injection flow less than 29 m^3/h (SAMG setpoint [F02], CA-1, [9], could accelerate core damage from core damage states OX/BD to EX. ERVC can assure that even for insufficient water injection (etc. insufficient STORE mobile pump characteristic for RCS decay heat removal with RPV injection), possible negative effects will be mitigated. Additionally, in combination with containment spray which will cool upper sections of RCS, ERVC strategy could provide more confidence in retaining the core inside the RPV.

Regarding SAG-8 “Flood the containment“, this strategy is performed when reactor vessel is already failed, so it is not suitable for ERVC strategy. SAG-8 flooding is to the elevation of [L03] (4000 m^3 of water) or [L03a] (9000 m^3 of water).

For containment flooding levels used in NEK SAMG [8], see Figure 10.

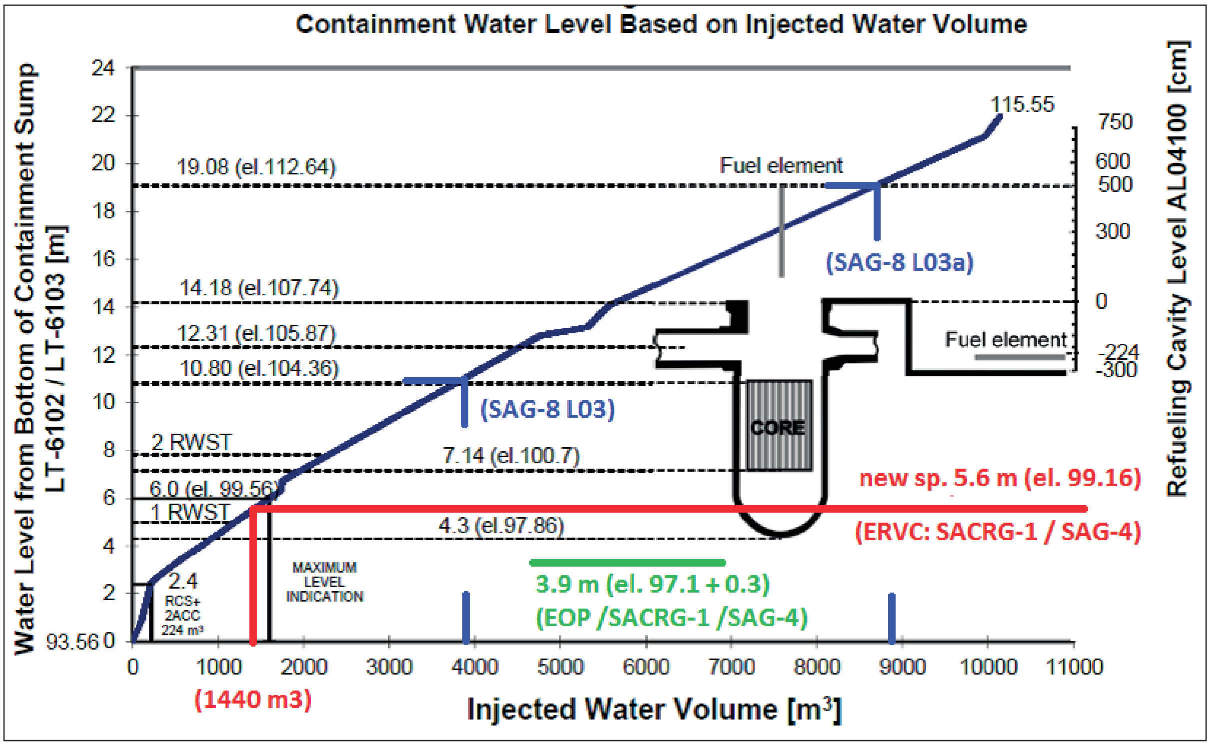

Figure 10: NPP Krško’s CA-5 “Containment Water Level and Volume” sketch [8] with marked EOP and SAMG containment flooding levels [9], [10]

4.3 Furtherer plant upgrade for ERVC performance

As a part of Safety Upgrade Program (SUP) modification 1029-RH-L, alternate containment injection flowpaths for variety of mobile pump usage with ERVC applicable performances will be installed. Also there will be installed additional borated water water tank, dedicated for beyond design basis accidents (design extended conditions) [6].

5 CONCLUSION

NPP Krško could perform external reactor vessel cooling (ERVC) strategy with reasonable confidence. Total containment water inventory need for ERVC performance is 1440 m^3, to cover bottom of reactor vessel up to elevation 99.2 m (or 5.6 m measured on containment sump level indicators) [27]. Strategies for performing this action for severe accident scenarios are:

- for Loss of all AC event - RWST gravity drain (up to 1250 m^3 - depending on containment backpressure), with additional alternate water inventory needed to be injected by AE equipment, and
- for AC available events - RWST injection with ECCS (total inventory of 1010 m3) up to “RWST Empty” alarm (18%), with additional 430 m^3 of alternate water inventory needed to be simultaneously injected by ECCS, DEC or AE equipment.

If RWST gravity drain is unavailable, containment injection by alternate equipment and water sources will take a place. Eventually spilled and partially evaporated RCS and SI ACC inventory (in total 226 m^3) is not taken into account.

Per probabilistic and deterministic analysis, time window for successful ERVC strategy performance during severe accident occurrence (after observed CET > 650˚C) is 2.5 hours for TEHNNN plant damage state (PDS) scenario with 58% of total CDF participation. This action should be performed early after transition to Severe Accident Management Guidance (SAMG). During loss of all AC power scenario leading to core damage, early containment flooding is performed before CET > 650˚C condition within EOP ECA-0.0 procedure, which extends time for successful strategy implementation.

There are no negative effects due to ERVC performance. New flooding level will not threaten equipment and instrumentation needed for long term SAMGs performance or hamper the ECCS recirculation capabilities. Eventually diluted containment sump borated water inventory will not cause return to criticality during recirculation phase because of lost core geometry [7].

ERVC strategy will also positively interfere with other severe accident strategies regarding corium retention inside reactor vessel. It will supplement the effect of insufficient individual equipment performances needed for successful severe accident mitigation. Example is combining ERVC with simultaneously insufficient RCS injection and/or minor containment spraying of upper RCS sections.

Changes in current NPP Krško SAMGs regarding ERVC strategy implementation could include: SACRG-1 “Severe Accident Control Room Guideline Initial Response”, and SAG-4 “Inject into Containment”.

Regarding further development of generic SAMG’s, containment flooding strategy to perform ERVC will be used. Development of Severe Accident Control Room Guideline - Loss of DC and Instrumentation, proposes integrated RCS and containment flooding strategy for prevention of RPV failure, by early application of ERVC strategy if core condition is unknown.

As a part of NPP Krško Safety Upgrade Program (SUP) modification regarding ERVC applicability, alternate containment injection flowpaths for AE equipment and additional borated water tank, will be introduced.

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International Context Regarding Application Of Single Failure Criterion (SCF) For New Reactors

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ABSTRACT

The paper provides an overview of the regulatory design requirements for new reactors addressing Single Failure Criterion (SFC) in accordance to international best-practices, particularly considering the SCF relation to in-service testing, maintenance, repair, inspection and monitoring of systems, structures and components important to safety.

The report [1] discusses the detailed comparison of the current SFC requirements and guidelines published by the IAEA, WENRA, EUR and nuclear regulators in the United States, United Kingdom, Russia, Korea, Japan, China and Finland. However, this paper presents the summary of work from [1] and 2major examples from IAEA and WENRA and applications for small and modular reactors.

1 INTRODUCTION

The Single Failure Criterion (SFC) ensures reliable performance of safety systems in nuclear power plants in response to design basis initiating events. The SFC, basically, requires that the system must be capable of performing its task in the presence of any single failure.

The capability of a system to perform its design function in the presence of a single failure could be threatened by a common cause failure such as a fire, flood, or human intervention or by any other cause with potential to induce multiple failures. When applied to plant's response to a postulated design-basis initiating event, the SFC usually represents a requirement that particular safety system performs its safety functions as designed under the conditions which can include:

- All failures caused by a single failure;
- All identifiable but non-detectable failures, including those in the non-tested components;
- All failures and spurious system actions that cause (or are caused by) the postulated event.

2 OVERVIEW OF INTERNATIONAL PRACTICE

2.1 IAEA application of Single Failure Criteria (SFC) and allowable outage time (AOT)

IAEA, in the major document related to the design of the nuclear power plants (SSR-2/1 as in the process of post-Fukushima upgrade [2]), defines under section 5 (General Plant Design) the single failure criterion in Requirement 25:

"*The single failure criterion shall be applied to each safety group incorporated in the plant design.*

5.39. Spurious action shall be considered to be one mode of failure when applying the concept to a safety group or safety system.

5.40. The design shall take due account of the failure of a passive component, unless it has been justified in the single failure analysis with a high level of confidence that a failure of that component is very unlikely and that its function would remain unaffected by the postulated initiating event." explaining that "*the single failure is a failure that results in the loss of capability of a system or component to perform its intended safety function(s) and any consequential failure(s) that result from it. The single failure criterion is a criterion (or requirement) applied to a system such that it must be capable of performing its task in the presence of any single failure*."

It should be noted that IAEA SSR-2/1 mentions the term "safety group" only in the Requirement 25 without definition and that in all other requirements only term "safety system" is applied. IAEA Safety Glossary [22] defines a "safety system" as a system important to safety, provided to ensure the safe shutdown of the reactor or the residual heat removal from the core, or to limit the consequences of anticipated operational occurrences and design basis accidents. Safety systems consist of the protection system, the safety actuation systems and the safety system support features. Components of safety systems may be provided solely to perform safety functions, or may perform safety functions in some plant operational states and non-safety functions in other operational states. Furthermore, IAEA Safety Glossary [22] defines a "safety group" as the assembly of equipment designated to perform all actions required for a particular postulated initiating event to ensure that the limits specified in the design basis for anticipated operational occurrences and design basis accidents are not exceeded. Per our understanding of IAEA glossary, single "safety system" is designed to perform its single safety function e.g. decay heat removal from core while "safety group" covers the few "safety systems" to perform all actions required for a particular postulated initiating event (Large Break LOCA).

Generally, based on the SSR-2/1, IAEA requires application of the single failure criteria (SFC) for all safety systems and it is covered by IAEA NS-G guidelines (e.g. NS-G-1.9, Design of the Reactor Coolant System and Associated Systems in Nuclear Power Plants or NS-G-1.10 Design of Reactor Containment System for Nuclear Power Plants, etc.). Generally, in applicable IAEA NS-G guides it is discussed that the all evaluations performed for design basis accidents should be made using an adequately conservative approach. In a conservative approach, the combination of assumptions, computer codes and methods chosen for evaluating the consequences of a postulated initiating event should provide reasonable confidence that there is sufficient margin to bound all possible The assumption of a single failure in a safety system should be part of the conservative approach, as indicated in SSR-2/1. Care should be taken when introducing ad equate conservatism, since:

– For the same event, an approach considered conservative for designing one specific system could be non-conservative for another;
– Making assumptions that are too conservative could lead to the imposition of constraints on components that could make them unreliable.

Allowable Outage Time (AOT)

Under Requirement 28 in SSR-2/1 (Operational limits and conditions for safe operation) it is stated that the design shall establish a set of operational limits and conditions for safe operation of the nuclear power plant. Para 5.44: The requirements and operational limits and conditions established in the design for the nuclear power plant shall include ([3], requirement 6):

a) Safety limits;
b) Limiting settings for safety systems;
c) Limits and conditions for normal operation;
d) Control system constraints and procedural constraints on process variables and other important parameters;

e) Requirements for surveillance, maintenance, testing and inspection of the plant to ensure that structures, systems and components function as intended in the design, to comply with the requirement for optimization by keeping radiation risks as low as reasonably achievable;

f) Specified operational configurations, including operational restrictions in the event of the unavailability of safety systems or safety related systems;

g) Action statements, including completion times for actions in response deviations from the operational limits and conditions.

Furthermore, Requirement 29 (Calibration, testing, maintenance, repair, replacement inspection and monitoring of items important to safety) in para 5.46 requires that where items important to safety are planned to be calibrated, tested or maintained during power operation, the respective systems shall be designed for performing such tasks with no significant reduction in the reliability of performance of the safety functions. Provisions for calibration, testing, maintenance, repair, replacement or inspection of items important to safety during shutdown shall be included in the design so that such tasks can be performed with no significant reduction in the reliability of performance of the safety functions. Para 5.47 provides the alternatives if an item important to safety cannot be designed to be capable of being tested, inspected or monitored to the extent desirable. Alternatives include a robust technical justification that incorporates the following approach:
(a) Other proven alternative and/or indirect methods such as surveillance testing of reference items or use of verified and validated calculational methods shall be specified.

(b) Conservative safety margins shall be applied or other appropriate precautions shall be taken to compensate for possible unanticipated failures

Additionally to requirements from IAEA SSR-2/1 [2], SSR-2/2 [3]((IAEA Safety Standard Series, SSR-2/2, Safety of Nuclear power Plants: Commissioning and Operations, Rev. 1 in preparation, 2014) defines that in, para 4.9, the operational limits and conditions shall include requirements for normal operation, including shutdown and outage stages, and shall cover actions to be taken and limitations to be observed by the operating personnel. Furthermore, para 4.12 requires that the operating organization shall ensure that an appropriate surveillance programme is established and implemented to ensure compliance with the operational limits and conditions, and that its results are evaluated, recorded and retained.

IAEA Safety Guide NS-G-2.2 [4] defines the requirements for plant safety limits, limiting safety systems settings, surveillance requirements and limits and conditions for normal operations. Under section 6 the requirements for the limits and conditions for normal operations are described in details.

Previously, IAEA had a document Safety Series document 50-P-1 (Application of the Single Failure Criteria, [8]). This document is outdated but there is still no new IAEA document superseded it. However, [8] in section 2 deals with the purpose of the single failure criterion with respect to the safety of a nuclear power plant. It also shows where the criterion has its limitations. The third section explains the difference between active and passive types of failure and the consequences of the failure characteristics for the application of the criterion. Examples are given of simple and more sophisticated component redundancy arrangements in a fluid system. The possibility of fail-safe designs and the role of auxiliary systems are also dealt with. The following section, which is supported by an extensive appendix on various methods to determine allowable outage times for redundant components, treats the important case of the reduction of redundancy during in-service maintenance and repair actions in operating nuclear power plants. Different maintenance strategies are discussed. Section 5 then considers that part of the definition of the single failure criterion which states that consequential effects of a single failure are to be considered as part of the failure. Section 6 provides an introduction to the problem of common cause failures. While the single failure criterion may be satisfied by redundancy of identical components, the

common cause failure of such components would nullify this redundancy. Exemptions from the application of the criterion are related to failure occurrence probability in Section 7. The methodology and the individual steps involved in a single failure analysis (SFA) are explained in the last section. A short commentary on the complementary use of probabilistic safety assessment (PSA) methods is also given. Permissible outage time in the context of single failure criteria is discussed in section 4.1.3. The basic requirements concerning permissible maintenance, test and repair times should be considered. They can be summarized as follows:
(a) If during maintenance, test or repair work, the assumption of a single failure would lead to a failure of the safety features; these activities are only permissible within a relatively short period without special measures being taken (e.g. replacing the function or rendering its operability superfluous). In most cases the time involved in the maintenance, test or repair procedure is so short as to preclude any significant reduction of the reliability of the safety feature concerned. Various methods (including probabilistic) can be used to determine an admissible outage period.

(b) If the resultant reliability is such that the safety feature no longer meets the criteria used for design and operation, the nuclear power plant shall be shut down or otherwise placed in a safe state if the component temporarily out of service cannot be replaced or restored within a specified time (stated in the technical specifications).

(c) Maintenance procedures on safety features over a longer period, during which the component concerned is not operable, are only admissible without special measures if in addition to the maintenance a single failure can be assumed without preventing the safety feature from fulfilling its safety function or if another available system can adequately replace the impaired function.

(d) Even if the single failure criterion is fulfilled during the maintenance procedure, the time for this procedure should be reasonably limited. (e) A PSA can be used to define the maintenance and repair times (time from the detection of the failure until the completion of the repair procedure), as well as the inspection concept. If this is done, the maintenance procedures should be defined so that they do not reduce the reliabilities of the safety features below the value required for the relevant PIEs and so that the probabilistic safety criteria, if established, are met.

Several methods can be used for the determination of permissible outage times. Important parameters are the degree of redundancy of the components or systems and the failure rate. The final goal is always the performance of a certain safety function, not primarily the availability of a particular component. The determination of the required degree of redundancy has to take this into account. It allows, therefore, not only for parallel trains of identical configuration but also for other systems which could perform the same function. Taking into account the need for reliability of safety systems and the desire for high operational availability, some countries consider it necessary in ensuring plant safety to require, along with the single failure criterion, additional redundancy for some specified safety functions in order to be able to cope with both ongoing maintenance or repair work and a simultaneous single failure. This requirement leads to an n + 2 degree of redundancy, for example 4 X 50% or 3 X 100% redundancy concepts. Another method used in many countries is to increase the redundancy of active components (e.g. pumps, valves) which require the most frequent maintenance. This leads in general to a 4 x 50% or a 4 x 100% redundancy concept for such components. It should also be noted that some countries as a result of probabilistic considerations introduce further equipment in addition to the single failure criterion requirements. This increases the level of redundancy of some safety groups required to cope with the relevant PIEs.

The question of common cause failure must also be considered, as described in Section 6 of [7]). The advantage of applying these concepts is not only a higher reliability of the safety systems but also a higher availability of the plant, because in the event of longer lasting repair activities additional measures such as power reduction or plant shutdown are not necessary. The choice between the possibilities is then also an economic matter; the investment costs must be compared with the anticipated savings connected with the improved availability of the plant.

Exception during testing and maintenance - Allowable Outage Time (AOT)

Detailed methodology for determination the surveillance test intervals and allowed outage times (AOT) of systems and components important to safety are not discussed in IAEA guides. However, under IAEA SSG-3[6] is discussed that the results of the PSA should be used in developing emergency procedures for accidents and to provide inputs into the technical specifications of the plant. In particular, the results of the PSA should be used to investigate the increase in risk after the removal from service of items of equipment for testing or maintenance and the adequacy of the frequency of surveillance or testing. The PSA should be used to confirm that the allowed outage times do not contribute unduly to risk and to indicate which combinations of equipment outages should be avoided. In the chapter „Risk Informed Technical Specifications (bullets 10.28 to 10.35) " it is discussed that The limiting conditions for operation give, for example, the requirements for equipment operability, the allowed outage times and the actions required (e.g. the testing requirements for redundant equipment). The allowed outage time for a particular system or component is the period of time within which any maintenance or repair activity should be completed. If the allowed outage time is exceeded, the technical specifications specify the actions that the plant operators should take. For example, if an allowed outage time is exceeded during operation at power, the requirement may be for the operators to reduce power or to shut down the plant. In addition, the requirements for equipment operability usually include limits on the combinations of equipment that can be removed for maintenance at the same time (usually referred to as configuration control). Insights from PSA can be used as an input to justify limiting conditions for operation and allowed outage times. Similarly it is discussed also for surveillance test periods, etc. Some details about practice of risk based AOT optimization is given in few older IAEA-TECDOCs documents [9], [11] and [11].

2.2 WENRA RHWG Safety Reference Levels related to SFC and AOT

A principal aim of the Western European Nuclear Regulators' Association (WENRA) is to develop a harmonized approach to nuclear safety within the member countries. One of the first major achievements to this end was the publication in 2006 of a set of safety reference levels (RLs) for operating nuclear power plants (NPPs) [15] . After the TEPCO Fukushima Daiichi nuclear accident, they have been further updated to take into account the lessons learned, including the insight from the EU stress tests. As a result a new issue on natural hazards was developed and significant changes made to several existing issues.

WENRA Rls cover the 19 areas (01 Issue A:Safety Policy, 02 Issue B:Operating Organisation,03 Issue C:Management System, 04 Issue D:Training and Authorization of NPP Staff (Jobs with Safety Importance), 05 Issue E:Design Basis Envelope for Existing Reactors, 06 Issue F: Design Extension of Existing Reactors, 07 Issue G: Safety Classification of Structures, Systems and Components, 08 Issue H: Operational Limits and Conditions (OLCs), 09 Issue I: Ageing Management, 10 Issue J: System for Investigation of Events and Operational Experience Feedback, 11 Issue K: Maintenance, In-Service Inspection and Functional Testing, 12 Issue LM: Emergency Operating Procedures and Severe Accident Manage-ment Guidelines, 13 Issue N: Contents and Updating of Safety Analysis Report (SAR), 14 Issue O: Probabilistic Safety Analysis (PSA), 15 Issue P: Periodic Safety Review (PSR), 16 Issue Q: Plant Modifications, 17 Issue R: On-site Emergency Preparedness, 18 Issue S: Protection against Internal Fires, 19 Issue T: Natural Hazards).

Single Failure Criterion is considered in several safety reference levels under Design Basis Envelope for Existing Reactors (Issue E), as shown below.

Demonstration of reasonable conservatism and safety margins

E8.2 The worst single failure (A failure and any consequential failure(s) shall be postulated to occur in any component of a safety function in connection with the initiating event or thereafter at the most unfavourable time and configuration.) shall be assumed in the analyses of design basis events. However, it is not necessary to assume the failure of a passive component, provided it is justified that a failure of that component is very unlikely and its function remains unaffected by the PIE.

Reactor and fuel storage sub-criticality

E9.7 At least one of the two systems shall, on its own, be capable of quickly rendering the nuclear reactor sub critical by an adequate margin from operational states and in de-sign basis accidents, on the assumption of a single failure.

Heat Removal Functions

E9.9 Means for removing residual heat from the core after shutdown and from spent fuel storage, during and after anticipated operational occurrences and design basis acci-dents, shall be provided taking into account the assumptions of a single failure and the loss of off-site power.

Reactor protection system

E10.7 Redundancy and independence designed into the protection system shall be sufficient at least to ensure that:

- no single failure results in loss of protection function; and
- the removal from service of any component or channel does not result in loss of the necessary minimum redundancy.

Emergency Power

E10.11 It shall be ensured that the emergency power supply is able to supply the necessary power to systems and components important to safety, in any operational state or in a design basis accident, on the assumption of a single failure and the coincidental loss of off-site power.

Alowable Outage Time (AOT)

The whole Issue H (Operational Limits and Conditions (OLCs)) deals with demonstration of OLCs to ensure that plants are operated in accordance with design assumptions and intentions as documented in the Safety Analysis Report (SAR). Among others, reference level H defines the unavailability of limits as:

H6.1 Limits and conditions for normal operation shall include limits on operating parame-ters, stipulation for minimum amount of operable equipment, actions to be taken by the operating staff in the event of deviations from the OLCs and time allowed to complete these actions.

H6.2 Where operability requirements cannot be met, the actions to bring the plant to a safer state shall be specified, and the time allowed to complete the action shall be stated.

H6.3 Operability requirements shall state for the various modes of normal operation the number of systems or components important to safety that should be in operating condition or standby condition.

Also, per H9.1 the licensee shall ensure that an appropriate surveillance program (*The objectives of the surveillance programme are: to maintain and improve equipment availability, to confirm compliance with operational limits and conditions, and to detect and correct any abnormal condition before it can give rise to significant consequences for safety. The abnormal conditions which are of relevance to the sur-veillance programme include not only deficiencies in SSCs and software performance, procedural errors and human errors, but also trends within the accepted limits, an analysis of which may indicate that the plant is deviating from the design intent. (NS-G-2.6 Para 2.11)*) is established and implemented to ensure compliance with OLCs and shall ensure that results are evaluated and retained.

In H10 non-compliances with defined OLCs requires the reports of non-compliance and corrective action shall be implemented in order to help prevent such non-compliance (taking into account that if the actions taken to correct a deviation from OLCs are not as prescribed, including those times when they have not been completed successfully in the allowable outage time, plant shall be deemed to have operated in non-compliance with OLCs.) in future.

Furthermore, the WENRA RHWG report on safety of new NPP designs [16] discusses some considerations based on the major lessons from the Fukushima Daiichi accident, especially concerning the design of new nuclear power plants, and how they are covered in the new reactor safety objectives and the common positions. The WENRA Objectives O1-O7 covers the following areas: O1. Normal operation, abnormal events and prevention of accidents, O2. Accidents without core melt, O3. Accidents with core melt, O4. Independence between all levels of Defence-in-Depth, O5. Safety and security interfaces, O6. Radiation protection and waste management and O7. Leadership and management for safety.

Within the WENRA Safety Objectives for New Nuclear Power Plants the words "reasonably practicable" or "reasonably achievable" are used. In this report the words Reasonably Practi-cable are used in terms of reducing risk as low as reasonably practicable or improving safety as far as reasonably practicable. The concept of reasonable practicability is directly analogous to the ALARA principle applied in radiological protection, but it is broader in that it applies to all aspects of nuclear safety. In many cases adopting practices recognized as good practices in the nuclear field will be sufficient to show achievement of what is "reasonably practicable".

The major change is refined structure of the levels of DiD (Defense in Depth) presented IN WENRA RHWG safety objectives for new NPP designs [16]. This document does not change the definition and usage of SFC according to WENRA RHWG safety reference levels for existing reactors [15] but discusses the some design expectations related to SFC. For example: while the postulated single initiating events analyses in combination with the single failure criteria usually gives credit on redundancy in design provisions of safety systems and of their support functions, addressing multiple failure events emphasizes diversity in the design provisions of the third level of DiD. Based on the [16], for DiD level 3.b, analysis methods and boundary conditions, design and safety assessment rules may be developed according to a graded approach, also based on probabilistic insights. Best estimate methodology and less stringent rules than for level 3.a may be applied if appropriately justified. However the maximum tolerable radiological consequences for multiple fail-ure events (level 3.b) and for postulated single failure events (level 3.a) are bounded by WENRA Objective O2 (accident without core melt).

Table 1 The refined structure of the levels of DiD proposed by RHWG

Levels of defence in depth		Objective	Essential means	Radiological consequences	Associated plant condition categories
Level 1		Prevention of abnormal operation and failures	Conservative design and high quality in construction and operation, control of main plant parameters inside defined limits	No off-site radiological impact (bounded by regulatory operating limits for discharge)	Normal operation
Level 2		Control of abnormal operation and failures	Control and limiting systems and other surveillance features		Anticipated operational occurrences
Level 3 [1]	3.a	Control of accident to limit radiological releases and prevent escalation to core melt conditions [2]	Reactor protection system, safety systems, accident procedures	No off-site radiological impact or only minor radiological impact [4]	Postulated single initiating events
	3.b		Additional safety features[3], accident procedures		Postulated multiple failure events
Level 4		Control of accidents with core melt to limit off-site releases	Complementary safety features[3] to mitigate core melt, Management of accidents with core melt (severe accidents)	Off-site radiological impact may imply limited protective measures in area and time	Postulated core melt accidents (short and long term)
Level 5		Mitigation of radiological consequences of significant releases of radioactive material	Off-site emergency response Intervention levels	Off site radiological impact necessitating protective measures[5]	-

(1) Even though no new safety level of defense is suggested, a clear distinction between means and conditions for sub-levels 3.a and 3.b is lined out. The postulated multiple failure events are consid-ered as a part of the Design Extension Conditions in IAEA SSR-2/1.

(2) Associated plant conditions being now considered at DiD level 3 are broader than those for existing reactors as they now include some of the accidents that were previously considered as "beyond de-sign" (level 3.b). For level 3.b, analysis methods and boundary conditions, design and safety assessment rules may be developed according to a graded approach, also based on probabilistic in-sights. Best estimate methodology and less stringent rules than for level 3.a may be applied if appropriately justified. However the maximum tolerable radiological consequences for multiple failure events (level 3.b) and for postulated single failure events (level 3.a) are bounded by WENRA Objective O2.

(3) The task and scope of the additional safety features of level 3.b are to control postulated common cause failure events as outlined in Section 3.3 on "Multiple failure events". An example for an additional safety feature is the additional emergency AC power supply equipment needed for the postulated common cause failure of the primary (non-diverse) emergency AC power sources. The task and scope of the complementary safety features of level 4 are outlined in Section 3.4 on "Provisions to mitigate core melt and radiological consequences". An example for a complementary safety feature is the equipment needed to prevent the damage of the containment due to combustion of hydrogen released during the core melt accident.

(4) It should be noted that the tolerated consequences of Level 3.b differ from the requirements con-cerning Design Extension Conditions in IAEA SSR-2/1 that gives a common requirement for DEC: "for design extension conditions that cannot be practically eliminated, only protective measures that are of limited scope in terms of area and time shall be necessary".

(5) Level 5 of DiD is used for emergency preparedness planning purposes.

The WENRA RHWG safety objectives for new NPP designs[16] does not deal with safety demonstration of the SFC. However, it points that the demonstration of physical impossibility, based on engineered provisions, can be difficult. Care must be taken to recognize that some claims for practical elimination may be based on as-assumptions (e.g. non-destructive testing, inspection) and those assumptions need to be acknowledged and addressed. For engineered provisions this can be done by excluding the certain feature from the design making further development of accident scenario impossible (accident sequence cut-off).

It should be noted that the level of defense are varying according different international guidelines as a basis to develop an evaluation basis for SFC criteria. See Table 2 bellow.

<u>Exception during testing and maintenance - Allowable Outage Time (AOT)</u>

However, WENRA RHWG safety objectives do not discuss application of the SFC in the context of determination of the allowable outage times (AOT) for redundant components. There is no recommendation how to treat the the reduction of redundancy during in-service maintenance and repair actions in operating nuclear power plants

2.3 Level of Depth in Defence (DiD) according different guidelines as a basis to develop an evaluation basis for licensing

Table 2 Level of DiD according different guidelines as a basis to develop an evaluation basis for licensing

Level of Defence	Initiating Event Frequency / yr	IAEA, SSG-2 [5], NOTE 2	EUR[17]	WENRA Note 1	STUK[20],	US-NRC[13]	ASME Service Levels
1	f=1	Normal Operation	DBC 1, Normal Operation	Normal Operation	DBC 1, Normal Operation	Normal Operation	A
2	$f>10^{-1}$	Anticipated Operational Occurances	DBC 2 Incidents	Anticipated Operational Occurances	DBC 2, Anticipated Operational Occurances	Anticipated Oper-ational Occurances (AOO)	B
3	$10^{-1}<f<10^{-2}$		DBC 3, Accidents of low Frequency	Design Basis Accidents 3.a Postulated Single Initiating Events			C
	$10^{-2}<f<10^{-4}$	Design Basis Accidents			DBC 3, Class 1 postulated accidents $10^{-2}<f<10^{-3}$ DBC 4, Class 2 postulated accidents $f<10^{-3}$	Design Basis Accidents (DBA) (Limiting Faults)	
	$10^{-4}<f<10^{-6}$	Beyond Design Basis Accidents	DBC 4, Accidents of very low Frequency	Design Basis Accidents 3.b Postulated Multiple Initiating events	DEC A		D
4a	$10^{-6}>f$	Severe Accidents	Complex Sequences	DEC A for which prevention of severe fuel damage in the core or in the spent fuel storage can be achieved;	DEC B	Beyond Design Basis Accidents	N/A
4b			Severe Accidents	DEC B with postulated severe fuel damage.	DEC C	Severe Accidents	
5			Severe Accidents	Accident with significant release of radioactivity to the environment			

Note 1: It should be noted that DiD for associated regulation was not assessed toward the initiating event frequency. The presented categorisation was made based on analogy with IAEA SSR-2/1. It was generally required that a list of PIEs shall be established to cover all events that could affect the safety of the plant. From this list, a set of anticipated operational occurrences and design basis accidents shall be selected using deterministic or probabilistic methods or a combination of both, as well as engineering judgement. The resulting design basis events shall be used to set the boundary conditions according to which the structures, systems and components important to safety shall be designed, in order to demonstrate that the necessary safety functions are accomplished and the safety objectives met.

Note 2 Regarding the IAEA SSG-2, please note that it is meant to apply for all the operating reactors in the world and that IAEA tends to come with guidelines which are acceptable for all reactor types and and all member states. In comparison to EUR, for example: EUR is meant for new reactors to be built in EU member countries. Furthermore: the limit / target of 1E-05 /yr from Canadian REGDOC 2.4.1 (section 8.2.3) is not necessarily directly comparable to the target of 1E-04 /yr in the IAEA's SSG-2 (Table 2). Canadian limit relates to "design basis accidents" (DBA). IAEA's target relates to "postulated initiating events" (PIE).

The "DBA" involves the "PIE" and allows / tolerates a single failure (provided that SFC is applied in the design, which should normally be the case). (For example: design basis LOCA followed by a failure of one ECCS train is still a design basis accident, if ECCS was designed according to the SFC.) The probability of a single failure (train level) by the "rule of thumb" can be taken as 1E-02 for a train with motor-driven pump, or 1E-01 for a train with a turbine-driven pump. Thus, when the IAEA SSG-2 says that PIE with freq. > 1E-04 /yr shall be enveloped by the design basis, it means that any accident sequence with frequency in the range 1E-06 - 1E-05 per year or higher (1E-04 /yr x (0.01 to 0.1)) shall produce no consequences larger than design basis consequences (concerning, for example, dose limits).

Table 2 was created by combining few sources which are not fully comparable but certain analogy was done. For illustration, please see below the original tables from SSG-2[5] and EUR rev D [17]:

SSG-2 Deterministic Safety Analyses for Nuclear Power Plants (2009)

Occurrence (1/reactor year)	Characteristics	Plant state	Terminology	Acceptance criteria
10^{-2}–1 (expected over the lifetime of the plant)	Expected	Anticipated operational occurrences	Anticipated transients, transients, frequent faults, incidents of moderate frequency, upset conditions, abnormal conditions	No additional fuel damage
10^{-4}–10^{-2} (chance greater than 1% over the lifetime of the plant)	Possible	Design basis accidents	Infrequent incidents, infrequent faults, limiting faults, emergency conditions	No radiological impact at all, or no radiological impact outside the exclusion area
10^{-6}–10^{-4} (chance less than 1% over the lifetime of the plant)	Unlikely	Beyond design basis accidents	Faulted conditions	Radiological consequences outside the exclusion area within limits
$<10^{-6}$ (very unlikely to occur)	Remote	Severe accidents	Faulted conditions	Emergency response needed

EUR Reference: Rev. D Section 2.1.8.2

Design Basis Category	Definition	Frequency of initiating event (per year)	Acceptance Criteria	
			Plant parameters	Radio-active releases
1	Normal Operation*		• Process parameters within Normal Operation* range of Technical Specifications*	Table 1
2	Incidents*	$f > 10^{-2}$	• Process parameters within applicable acceptance criteria	Table 1
3	Accidents (low frequency)	$10^{-2} > f > 10^{-4}$	• Acceptance criteria for Category 3 (1) • Limited Fuel Damage* (3) • Shutdown for Inspection* may be necessary	Appendix B
4	Accidents (very low frequency)	$10^{-4} > f > 10^{-6}$	• Acceptance criteria for Category 4 (1) • Core coolable geometry retained • Plant restart may be impossible • Peak clad temperature: 1204 °C (4, 5) • Local clad oxidation: 17% (4, 5) • Radial average peak fuel enthalpy at hot spot: 837 kJ/kg (4, 6, 7)	Appendix B

2.4 SINGLE FAILURE CRITERION APPLICATION IN NEW SMALL REACTOR DESIGNS

In tha last decade there was a lot of discussion related to the implementation of so called “small rectors” (SR) and “small modular reactors” (SMRs). To establish some context, it may be pointed that IAEA provides the following definitions concerning the “sizes” of the reactors:

- Small-sized reactors: < 300 MW(e)
- Medium-sized reactors: < 700 MW(e)
    - o Upper power limit may change as the current Large-sized reactors are being designed for up to 1700 MW(e).

Until recently, several dozens of Design Concepts of SRs and SMRs have been developed in Argentina, China, India, Japan, the Republic of Korea, Russian Federation, South Africa, USA, and several other IAEA Member States.

According to the definition of its role in the on-going SRs and SMR process, IAEA:

- Coordinates efforts of Member States to facilitate the development of SRs and SMRs by taking a systematic approach to identify key enabling technologies to achieve competitiveness and reliable performance of SRs and SMRs, and by addressing common issues to facilitate deployment;
- Establishes and maintains international network with international organizations involved on SRs and SMRs activities;
- Ensures overall coordination of Member States experts by planning and implementing training and by facilitating the sharing of information/experience, transfer of knowledge ;
- Develops international recommendations and guidance on SMRs, focusing on addressing specific needs of developing countries.

By definition, SRs and SMRs should have the following advantages:

- Fitness for smaller electricity grids;
- Options to match demand growth by incremental capacity increase;
- Tolerance to grid instabilities;
- Site flexibility;
- Other possible advantages;
- Lower capital cost but perhaps higher capital cost per MWe;
- Shorter and more reliable construction;
- Easier financing scheme;
- Enhanced safety;
- Reduced complexity in design and human factors;
- Suitability for process heat application.

IAEA developed the guidance for preparing user requirements documents for small and medium rectors and their application [25], although without clear design requirements. It is mentioned that the technical requirements should indicate that the design of a given new facility has to be in conformance with applicable rules, regulations, codes and technical standards. IAEA-TECDOC-1451 [26] discusses innovative small and medium sized reactors including, very briefly, design features, safety approaches and R&D trends. However, the mentioned document does not provide clear information regarding SMRs design requirements and, consequentially, does not mention SFC at all. Similarly to IAEA-TECDOC-1451, the IAEA-TECDOC-1485 [27], as well as TECDOC-1536 [28], discusses advantages of SMRs design only partially and without specific design requirements.

IAEA report NP-T-2.2 [24] discusses the design features for achieving defence in depth in 10 different designs of small and medium sized reactors where the part devoted to the application of SFC was very limited. In this document there is no mention of SFC as a specific design requirement from the IAEA. The latest IAEA documents discussing the advances in small modular reactor technology developments, [29], mentions, for the few applications, that the defence in depth (DID) concept is based on Western European Nuclear Regulators Association (WENRA) proposal and includes a clarification on multiple failure events, severe accidents, independence between levels, the use of the SCRAM system in some DID Level #2 events and the containment in all the Protection Levels. The safety systems are duplicated to fulfil the redundancy criteria, and the shutdown system is diversified to fulfil regulatory requirements. Application of SFC is not discussed at all.

In USA some utilities are considering licensing small modular reactor designs using the 10 CFR Part 52 combined license (COL) or early site permit (ESP) processes. The U.S. Nuclear Regulatory Commission (NRC) expects to receive applications for staff review and approval of small modular reactor (SMR)-related 10 CFR Part 52 applications as early as by the end of 2015. The NRC has developed its current regulations on the basis of experience gained over the past 40 years from the design and operation of large light-water reactor (LWR) facilities. Now, to facilitate the licensing of new reactor designs that differ from the current generation of large LWR facilities, the NRC staff seeks to resolve key safety and licensing issues and develop a regulatory infrastructure to support licensing review of these unique reactor designs. Toward that end, the NRC staff has identified several potential policy and technical issues associated with licensing of small LWR and non-LWR designs. The current status of these issues may be found in the series of related Commission documents (http://www.nrc.gov/reactors/advanced.html). The NRC staff has also assembled a list of stakeholder position papers identifying stakeholder documents that communicate opinions to the staff on technical or policy issues. Additionally, the NRC's Office of Nuclear Regulatory Research has engaged in an extensive program focusing on nine key areas of anticipatory and confirmatory research in support of licensing reviews for advanced reactors. The NRC also interacts with its international regulatory counterparts to share information. In August 2012, the NRC provided to Congress a requested report (Advanced Reactor Licensing) addressing advanced reactor licensing. The report addresses the NRC's overall strategy for, and approach to, preparing for the licensing of advanced non-LWR reactors. The report addresses licensing applications anticipated over the next two decades, as well as potential licensing activity beyond that time. It focuses on the licensing of nuclear reactor facilities for commercial use and illustrates regulatory challenges that may occur if various advanced reactor initiatives evolve into licensing applications. During 2012, DOE (Department of Energy) instituted an Advanced Reactor Concepts Technical Review Panel (TRP) process to evaluate viable reactor concepts from industry and to identify R&D needs. TRP members and reactor designers noted the need for a regulatory framework for non-light water advanced reactors. The TRP convened in spring 2014 reiterated the need for a licensing framework for advanced reactors:

- 10 CFR 50 requires applicants to establish principal design criteria derived from the General Design Criteria (GDC) of Appendix A.
- Since the GDC in Appendix A are specific to light water reactors (LWRs), this requirement is especially challenging for potential future licensing applicants pursuing advanced (non-light water) reactor technologies and designs.
- NE and NRC representatives agreed in June 2013 to pursue a joint licensing initiative for advanced reactors.

Overall purpose of this initiative is to establish clear guidance for the development of the principal design criteria (PDC) that advanced non-LWR developers will be required to include in their NRC license applications.

In the meantime, while USA NRC was still defining the position related to the licensing review of SMRs, the American Nuclear Society (ANS) issued in 2010 the Interim Report of the American Nuclear Society President's Special Committee on Small And Medium Sized Reactor (SMR) Generic Licensing Issues [23] which, among other issues, discusses the application of single failure criterion (SFC). Report mentions that the current SFC may not be appropriate to risk-informed safety assessments since it defeats the fundamental purpose of a risk analysis, given that all components, regardless of safety classification, have the opportunity to fail in a probabilistic assessment. SFC can be used to assess the importance of components and structures for design improvement, should the consequence be significant, but should not be mandatory. This SFC discussion is based on the the rigorous application of risk analysis in a plant design where the important design-basis events can be deduced from the event and fault trees. In addition, safety classification of systems, structures, and components can be directly determined from the analysis, as can reliability requirements for component performance and the need for inspection, test, and surveillance based on component importance. The risk-informed assessment also allows for explicit treatment of uncertainties, which conventional deterministic analysis largely ignores by applying "margins" and "conservatisms" intended to bound these unknowns. The risk assessment methodology allows for a more transparent understanding of the safety basis of reactors.

Finally, ANS concluded that a key element to development and implementation of innovative reactors is the use of a risk-informed framework, coupled with a demonstration test program upon which to issue DCs. Thus, the American Nuclear Society President's Special Committee on SMR Generic Licensing Issues (SMR Special Committee) recommends immediate development of a rulemaking to establish a new risk-informed, technology-neutral licensing process with a license-by-test element, to allow innovative designs to be developed and deployed more efficiently in the longer term.

None of other regulatory frameworks related to the SFC application discussed in sections 2.1-2.3 deals with the application of SFC specifically for the SMRs, from which it can be reasonably concluded that current regulations for large commercial NPPs (including the SFC application) will be in place until new regulations become available.

Canadian regulatory requirements for design of small reactor facilities [30] (RD-367, Design of Small Reactor Facilities) defines the "small reactor facility" as a reactor facility containing a reactor with a power level of less than approximately 200 megawatts thermal (MWt) that is used for research, isotope production, steam generation, electricity production or other applications. For reactors with power level above 200MWt Canadian regulatory requirements from REGDOC 2.5.2 [21] (Design of Reactor Facilities Nuclear Power Plants) are applicable. Differing to the all other regulatory approaches discussed above, Canadian regulatory requirements for design of small reactor facilities [30] in section 7.8.2 clearly defines that all safety groups shall be designed to function in the presence of a single failure. Each safety group shall perform all safety functions required for a PIE in the presence of any single component failure, as well as:

- all failures caused by that single failure;
- all identifiable but non-detectable failures, including those in the non-tested components;
- all failures and spurious system actions that cause (or are caused by) the PIE.

Each safety group shall be able to perform the required safety functions under the worst permissible systems configuration, taking into account such considerations as maintenance, testing, inspection and repair, and equipment outage. Analysis of all possible single failures and associated consequential failures shall be conducted for each element of each safety group until all safety groups have been considered. Such requirement is similar for the current large commercial nuclear power plant.

With above overview and discussion in mind, it is considered recommendable for the CNSC to investigate the risk-informed and performance-based alternatives to the single-failure criterion, such as those studied and described in [14], in order to identify potential alternative or complementary risk-informed approaches with respect to the SFC, for use in the new requirements for SMRs.

2.5 SFC Summary Table

Table 3 summarizes the approaches discussed in [1] in a limited scope due to the fact that all regulatory requirements related to the AOT and associated SFC are not written and defined in the same manner. Nuclear industries (utilities, NPPs, etc.) have developed procedures how to response to regulatory requirements and, typically, national regulators accept or refuse proposed application for relaxing the AOTs or SFC.

Table 3 Summary Table

Regulatory Position	**SFC applied to safety group or individual system**	**What systems have to meet SFC?**	**Is SFC applied during planned maintenance?**	**Is SFC applied during a repair within AOT?**	**Is SFC applied to passive components?**	**Is SFC applied in addition to assuming failure of a non-tested component?**
IAEA	Safety system	General approach: systems which prevent radioactive releases in environment. Because of different designs, system names and description it can be related to: • Reactor Protection System • Engineering Safety Feature Actuation System • Core Decay Heat Removal System • Emergency Core Cooling System • Containment decay heat removal system • Containment Isolation System • MCR Habitability System • Emergency AC/DC power • Safety System Support System (Component Cooling Water, etc.)	Not discussed directly in regulations. The allowable periods of safety systems inoperability and the cumulative effects of these periods should be assessed in order to ensure that any increase in risk is kept to acceptable levels.		General approach is that the fluid and electric systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming Passive Equipment functions properly) nor (2) a single failure of a Passive Equipment (assuming Active Equipment functions properly) results in a loss of capability of the system to perform its Safety Functions. *Exemption for passive components exists if justification of high standard and quality design and maintenance is possible.*	Not discussed directly in regulations. See 4th column on left side. In other words it means that if assessment of potential failure of any single component designed for the function in stand-by (non-tested) system shows the increase in risks above acceptable levels such test/maintenance should be excluded.
WENRA	Safety system					
EUR	Assembly of Equipment (combination of systems and components that perform a specific function)					
US NRC	Safety system					
Finish (STUK)	Safety system		Not discussed directly in regulations. The PRA shall be used to determine the surveillance test intervals and allowed outage times of systems and components important to safety. Actually, it is similar with above. YVL B.1 discusses actually the two failure criteria: • (N+1) failure criterion shall mean that it must be possible to perform a safety function even if any single component designed for the function fails. • (N+2) failure criterion shall mean that it must be possible to perform a safety function even if any single component designed for the function fails and any other component or part of a redundant system – or a component of an auxiliary system necessary for its operation – is simultaneously out of operation due to repair or maintenance. Some systems need to satisfy criteria (N+1) and some (N+2)			YVL B.1 discusses actually the two failure criteria as described in 4th column on the left side for Finish (STUK).
UK	Safety system		See IAEA, WENRA, EUR, US NRC above.			See IAEA, WENRA, EUR, US NRC above.
Japan	Structure, System and Components (SSCs)					
Korean	Safety system					
Russian	Safety features (safety systems elements)					
China	Safety system					
Canadian	Safety group/Safety system		A request for an exception during testing and maintenance should be supported by a satisfactory reliability argument covering the allowable outage time			Actually, similar to text for IAEA, WENRA, EUR, US NRC above even that section 7.6.2 of REG-DOC-2.5.2 [21] refers to the old IAEA, Safety Series No. 50-P-1 [8] which was withdrawn without applicable replacement.

3 CONCLUSION

The most important conclusion based on the presented work is that nuclear industry and regulation applications either to single failure criteria (SFC) or Defence in Depth (DiD) are not well harmonized. A bigger additional effort should be done to establish more strict and harmonized design requirements regard either SFC or DiD to improve safety of nuclear installation in future.

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Evaluation of Impact of NEK Safety Upgrade Program Implementation on the Reduction of Total Core Damage Frequency

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ABSTRACT

The nuclear accident in Japanese plants in 2011 has initiated a quick response from the countries utilizing a nuclear power. Krsko Nuclear Power Plant (NEK) was required by Slovenian Nuclear Safety Administration (SNSA) to perform consequential actions to reduce the risk of severe accidents and their consequences as low as feasible. NEK have analyzed the response to the severe accidents, and based on the results of this analysis, proposed the measures to be implemented in the shortest possible time.

In response to SNSA request NEK has developed the Safety Upgrade Program (SUP) which contains a comprehensive set of measures for plant safety improvements. SUP Phase 1 has been implemented during the Outage 2013.

This paper presents the work done on the evaluation of impact of plant modifications planned for implementation in Phases 2 and 3 of NEK Safety Upgrade Program on the quantitative figure of merit for the plant risk significance – total core damage frequency (CDF). A significance of the most valuable measures in terms of contribution to the CDF reduction was determined and a new fractional contribution of initiators categories was obtained.

The quantification of total CDF was made by employment of RiskSpectrum PSA code and running a predefined set of analytical cases for all initiator categories covered by the scope of NEK PSA model.

The basis for analysis was NEK's most recent at-power PSA model which reflects the plant status with plant modernization implemented in SUP Phase 1. The analysis was done in two steps. In the first step plant modifications planned for SUP Phase 2 were addressed and in the second step plant modernization measures and upgrades planned in SUP Phase 3 were modeled in addition to those already modelled in the first step.

Keywords: *Probabilistic Safety Assessment, PSA, Core Damage Frequency, Krsko, NEK, Safety Upgrade Program*

1 INTRODUCTION

Before the Fukushima accident, a certain plant modernization were in progress in NEK, such as installation of the third Emergency Diesel Generator, for power supply of safety systems, contributing to an increase in safety and at the same time also supports initiatives for modernization after the Fukushima accident. Thus, based on its own analysis, as well as on recommendations of international organizations and nuclear regulatory bodies, NEK has taken certain short-term and long-term actions.

Within the frame of short-term actions, the mobile equipment was purchased (e.g., diesel generators of different rated power, air compressors, water pumps, vehicles for transportation of mobile equipment). The modifications on some of the existing systems were done in order to allow connection of new mobile equipment to adequate connection points.

Within the frame of long-term actions, and due to the requirements from the SNSA Decree [1], NEK have analyzed the response to the severe accidents. Based on the results of the analysis documented in NEK technical report [2], the measures to be implemented in the shortest possible time were proposed. This results of analysis from [2] have shaped plant modernization program for severe accidents prevention and mitigation of their consequences, formally known as NEK Safety Upgrade Program, Rev. 0, [3]. In the meanwhile SUP was updated and reviewed, and the most recent design bases for additional structures, systems and components, as well as the list of plant modernization proposals were documented in SUP, Rev. 2, [4]. Document [5] establishes common understanding of design inputs, requirements and assumptions for all SUP modifications.

The paper analyzes the impact of the implementation of plant modifications in SUP Phases 2 and 3 on the total CDF, considering both internal and external initiators. The intention was to obtain a new total CDF which will reflect the impact of plant modifications taken integrally, as well as to determine a significance of the most valuable measures in terms of contribution to the total CDF reduction and obtain a new fractional contribution of initiators' categories.

2 METHODOLOGICAL APPROACH

The methodology used in this report is essentially the methodology used for evaluation of the risk from the plant in power operation mode. The methodology for assessment uses probabilistic approach, including the necessary engineering judgments. The Probabilistic Safety Assessment (PSA) is a methodology used to quantify risk based on the reliability of consequence-limiting equipment. NEK PSA procedure [6] defines methodology for application of a PSA. A PSA evaluation could be performed at several levels of scope. Two levels are used in the NEK PSA – Level 1 and Level 2. The scope of this paper is focused on the Level 1 PSA modelling and quantification.

Within the scope of this paper the total CDF, due to both external and internal initiators, was evaluated. The analysis was performed by employment of the second generation of Windows OS based RiskSpectrum PSA (RS PSA) code, version 1.2.1.1 with built-in RiskSpectrum Analysis Tool, version 3.2.3.1 [7]. Consequence-type analyses were performed by RS PSA for different initiators categories contributing to the total CDF. Analyzed initiators categories include:

- Internal initiating events (IIE) – 16 categories,
- Internal fire events,
- Internal flooding events,
- High energy line break (HELB) events,
- Seismic events and liquefaction, and
- Other external events (OEE) covering: aircraft accidents, turbine-generated missiles, external fire, external flooding, industrial and military accidents, pipeline (gas), release of chemicals, severe winds, transport. accidents, turbine missiles, glaze ice, extreme drought.

The evaluation was done in two consecutive steps. In the first step the plant modifications planned for implementation in SUP Phase 2 were modelled in PSA model and quantified. In the second step, in addition to the plant modifications modeled in the first step, the plant modifications planned for implementation in SUP Phase 3 were modelled and quantified.

Finally, the evaluation is concluded with results of quantification of NEK PSA model posterior to the implementation of both Phases 2 and 3, in which plant modifications, as seen and planned at the time of preparation of this paper, are addressed integrally.

3 EVALUATION

The starting point and the basis for the evaluation is NEK baseline at-power PSA model "NEKC28" (referential or baseline model), documented in technical report [8], and it reflects the plant status with modifications from SUP Phase 1 implemented, i.e. Passive Containment Filtering Vent System and Passive Autocatalytic Recombiners.

The total CDF obtained by quantification of referential PSA model "NEKC28" is estimated at 4.69E-5 /rcryr (baseline total CDF). These results in terms of contribution of all initiators' categories are, as well as the fractional contribution of initiator categories to baseline total CDF are illustrated by blue bars in Figure 1 and Figure 2 (baseline total CDF) in section 3.2.

3.1 Evaluation NEK Safety Upgrade Program Phase 2

The purpose of the evaluation performed in this section is to obtain insight on the impact of plant modifications planned for implementation in Phase 2 on the total CDF, as compared to the baseline total CDF. Plant modifications planned for implementation in Phase 2 were modelled in PSA model "NEKC28". Limitations and exceptions during modelling of modifications in Phase 2 were noted to the associated plant modification, where appropriate. Modelled plant modifications encompass the following:

1. SUP Phase 2 plant modifications per [4]:
    a. Construction of Emergency Control Room (ECR) and Technical Support Center (TSC) in BB1 building (Phase 2c).
    b. Additional PRZR PORV Bypass valves for RCS pressure relief (Phase 2d).
    c. Upgrade of flooding protection of NSSS island (Phase 2a).
    d. Upgrade Operating Support Center (OSC) with additional emergency power supply capacities and conditions for long term presence of operating personnel during accident (Phase 2b). These modifications were not addressed in PSA model since OSC is not a system for performing a safety function and directly mitigating a sequence leading to the core damage.
    e. Alternative Spent Fuel Pool (SFP) cooling (additional sprinklers for SFP cooling and connections for mobile heat exchanger). This modification was not addressed in PSA model; it should be reflected in the NEK SFP PSA model.
    f. Additional A-RHR heat exchanger for alternative long-term RCS / containment cooling and decay heat removal. An attempt was made to address the impact of installation of Alternative RHR system (A-RHR) on the CDF in NEK PSA model, and the result has shown no change in CDF. The importance analysis was carried out for the existing RHR system, and the highest risk decrease factor (RDF) estimated at 1.01 was obtained for a basic event "RHR MDPs failure to start due to CCF", by running the importance analysis of RH system's "No low pressure recirculation flow" top gate. Since this risk decrease potential is very small, installation of an additional RHR pump / train would have negligible contribution to further risk reduction (unchanged total CDF). The reason for this lies in the fact that a typical mission time of 24 hours, used in the standard PSA model, is considered to be sufficient to reach stable state after the accident. As the evolution of the accident takes more than 24 hours, the impact on CDF may not be demonstrated for long term low pressure recirculation mode. Consequently, the importance and benefit of installation of an alternative RHR train is not "visible" through the CDF metric which is "driven" by 24 hour mission time.
2. Installation of the shielding of ESW pumps' motors from the spray source.

    In the ESW Pumphouse, shielding can be accomplished with placing sturdy removable steel jacketing around the expansion joints that would protect the pumps from potential water spray. It has to be noted at this point that this modification as such is not a part of SUP Phase 2. Although, this modification is not explicitly stated as a SUP modification, it was identified during NEK analyses of potential safety improvements [2], as a measure with impact on CDF reduction with "High" risk significance, and as such has to be addressed in the PSA model.

The modifications stated in the paragraphs above were modeled in "NEKC28" model, and the resultant model is named "PNV2".

3.1.1 Analysis and Quantification Results (SUP Phase 2)

The results obtained by running the consequence analysis cases on "PNV2" model are provided in Table 1, Figure 1 and Figure 2 (red bars). They shows an impact on CDF reduction (per initiator category) of plant modifications defined by the scope of Phase 2 as compared to the baseline CDF obtained by the referential PSA model (with SUP Phase 1 addressed).

Table 1 Comparison of the CDF per Initiators' Group (Phase 2 Addressed vs. Phase 1 Addressed)

Initiators' Group	*Baseline CDF [/rcryr] (NEKC28)*	*CDF Posterior to SUP Phase 2 [/rcryr] (PNV2)*	*Abs. Delta CDF [/rcryr]*	*Delta CDF Rel. to Baseline CDF due to Initiators' Group [%]*	*Total CDF Reduction Factor (RF)*	*Delta CDF Rel. to Baseline Tot. CDF [%]*
IIE	1.22E-5	1.22E-5	0.00E+00	0.0%	1.0	0.0%
FIRE	1.26E-5	2.90E-6	-9.70E-6	-76.9%	4.3	-20.7%
FLOOD	4.88E-6	6.71E-7	-4.21E-6	-86.3%	7.3	-9.0%
HELB	1.48E-6	1.46E-6	-1.51E-8	-1.0%	1.0	0.0%
SEISMIC	1.12E-5	1.10E-5	-2.19E-7	-2.0%	1.0	-.5%
OEE	4.54E-6	3.73E-6	-8.06E-7	-17.8%	1.2	-1.7%
TOTAL	**4.69E-5**	**3.20E-5**	**-1.49E-5**	**-31.9%**	**1.47**	**-31.9%**

Based on the quantification of "**PNV2**" model, the total CDF posterior to the implementation of Phase 2 was estimated at **3.20E-5 /rcryr**, which represents a **reduction of 32%** (reduction factor of 1.5) as compared to the baseline total CDF (4.69E-5 /rcryr). The measures planned for implementation in Phase 2 with most dominant contribution to baseline total CDF reduction are:

- Construction of ECR – contribution to baseline total CDF reduction of 20.7%,
- Installation of ESW pumps shielding against spraying – contribution to baseline total CDF reduction of 9.0%,
- Upgrade of flooding protection of NSSS island – contribution to baseline total CDF reduction of 1.7%.

Consequently, the construction of ECR significantly reduces CDF due to Internal Fire to the value of 2.9E-6 /rcryr (reduction of 77% relative to the baseline CDF from internal fire events). Similarly, installation of SW pumps shielding against spraying reduced CDF due to Internal Flooding to the value of 6.7E-7 /rcryr (reduction of 86% relative to the baseline CDF from internal flooding events). The installation of flooding protection of NSSS island reduces CDF due to Other External Events (in particular "External Flooding" category) to the value 3.7E-6 /rcryr (reduction of 18% relative to baseline CDF from other external events).

CDF due to IIEs was calculated to be unchanged since the plant modernization measures in Phase 2 do not affect plant systems responsible for mitigation of sequences leading to CD due to internal initiators. Similarly, calculated CDF due to HELB as well as CDF due to seismic events are practically unchanged.

3.2 Evaluation of NEK Safety Upgrade Program Phase 3

The purpose of the evaluation performed in this section is to get an insight of the impact of the implementation of Phase 3 on the total CDF, as compared to both baseline total CDF and total CDF posterior to the implementation of Phase 2.

Plant modifications planned for implementation in Phase 3 were modelled in "PNV2" model. Limitations and exceptions during modelling of modifications in Phase 3 were noted to the associated plant modification, where appropriate. Plant modifications subject to Phase 3 [4] are:

1. Installation of Alternative Safety Injection (A-SI) pump and associated Alternative Borated Water Tank (A-BWT) for RCS injection with borated water (primary injection) in BB2 building (Phase 3a).
2. Installation of Alternative Auxiliary Feedwater (A-AF) pump and associated Alternative Condensate Water Tank (A-CYT) with water inventory for SG injection (secondary injection) in BB2 building (Phase 3b).
3. Construction of interconnections between BB1 and BB2 buildings and interconnections between BB2 building and NSSS island, which are seismically designed and resistant to liquefaction. This is not explicitly listed as a modification in Phase 3 per [4], but there is a requirement in [5] that equipment and interconnections from new DEC systems to the existing systems equipment shall be designed to meet seismic performance requirements during and after a DEC earthquake with Peak Ground Acceleration (PGA) intensity of 0.6*g*.
4. Plateau for mobile equipment seismically designed for 0.6*g* PGA for mobile equipment with mobile diesel generator mounted with seismic isolation.

The modifications listed above were modeled in the PSA model "PNV2", and the resultant model named "PNV3" reflects the plant status with both Phases 2 and 3 modifications addressed.

3.2.1 Analysis and Quantification Results (SUP Phase 3)

The results obtained by running the same set of consequence analysis cases are provided in Table 2, Figure 1 and Figure 2 (green bars). They reflect an impact on CDF reduction (per initiator categories) of implementation plant modifications in Phase 3 as compared to CDF obtained by model with Phase 2 addressed.

Table 2 Comparison of the CDF per Initiators' Group (Phase 3 Addressed vs. Phase 2 Addressed)

Initiators' Group	*CDF Posterior to SUP Phase 2 [/rcryr] (PNV2)*	*CDF Posterior to SUP Phase 3 [/rcryr] (PNV3)*	*Abs. Delta CDF [/rcryr]*	*Delta CDF Rel. to Baseline CDF due to Initiators' Group [%]*	*Total CDF Reduction Factor (RF)*	*Delta CDF Rel. to Baseline Tot. CDF [%]*
IIE	1.22E-5	2.22E-6	-9.98E-6	-81.8%	5.5	-21.3%
FIRE	2.90E-6	1.18E-6	-1.73E-6	-59.5%	2.5	-3.7%
FLOOD	6.71E-7	2.86E-8	-6.42E-7	-95.7%	23.4	-1.4%
HELB	1.46E-6	1.05E-7	-1.36E-6	-92.8%	14.0	-2.9%
SEISMIC	1.10E-5	4.81E-6	-6.17E-6	-56.2%	2.3	-13.2%
OEE	3.73E-6	3.63E-6	-1.00E-7	-2.7%	1.0	-.2%
TOTAL	**3.20E-5**	**1.20E-5**	**-2.00E-5**	**-62.5%**	**2.67**	**-42.6%**

Based on the quantification of "**PNV3**" model, the total CDF posterior to the implementation of Phase 3 was estimated at **1.20E-5 /rcryr**, which represents a significant reduction of 63% (a reduction factor of 2.7) as compared to the total CDF obtained posterior to Phase 2, and reduction of 43% as compared to the baseline total CDF (4.69E-5 /rcryr). The measures planned for implementation in Phase 3 with the most significant contribution to reduction of baseline total CDF are:

- Installation of additional pumps (A-AF and A-SI) and associated tanks (A-CYT and A-BWT) – contribution to baseline total CDF reduction of 21.3%.
- Construction of interconnections between BB1 and BB2 buildings and interconnections between BB2 building and NSSS island, which are seismically designed to withstand PGA of 0.6*g* and resistant to liquefaction and mobile diesel generator mounted with seismic isolation – contribution to baseline total CDF reduction of 13.2%;

Consequently, the installation of alternative pumps A-AF and A-SI, as well as associated tanks A-CYT and A-BWT in BB2 building, would significantly reduce CDF due to IIEs to the value of 2.2E-6 /rcryr (reduction of 82% relative to the baseline CDF due to IIEs). Similarly, construction of seismically designed and liquefaction resistant interconnections between buildings BB1 and BB2 and interconnections between BB2 building and NSSS island would lead to reduced CDF due to seismic events to the value of 4.8E-6 /rcryr (reduction of 56% relative to the baseline CDF from seismic events).

In order to obtain an insight of the cumulative effect of plant modifications planned in both Phase 2 and 3 a comparison of contributions from all initiator categories to the total CDF between is summarized and presented in Table 3.

Table 3 Comparison of the CDF per Initiators' Group (Phase 3 Addressed vs. Phase 1 Addressed)

Initiators' Group	*Baseline CDF [/rcryr] (NEKC28)*	*CDF Posterior to SUP Phase 3 [/rcryr] (PNV3)*	*Abs. Delta CDF [/rcryr]*	*Delta CDF Rel. to Tot. CDF due to Initiators' Group [%]*	*Total CDF Reduction Factor (RF)*	*Delta CDF Rel. to Baseline Tot. CDF [%]*
IIE	1.22E-5	2.22E-6	-9.98E-6	-81.8%	5.5	-21.3%
FIRE	1.26E-5	1.18E-6	-1.14E-5	-90.7%	10.7	-24.4%
FLOOD	4.88E-6	2.86E-8	-4.85E-6	-99.4%	170.4	-10.3%
HELB	1.48E-6	1.05E-7	-1.38E-6	-92.9%	14.1	-2.9%
SEISMIC	1.12E-5	4.81E-6	-6.39E-6	-57.0%	2.3	-13.6%
OEE	4.54E-6	3.63E-6	-9.06E-7	-20.0%	1.2	-1.9%
TOTAL	**4.69E-5**	**1.20E-5**	**-3.49E-5**	**-74.5%**	**3.92**	**-74.5%**

The **total CDF** evaluated on the basis of implemented plant modifications in Phase 1, as well as scope of modifications planned in Phases 2 and 3 is estimated, as mentioned before, at about **1.20-5 /rcryr**. Cumulative reduction of total CDF is about 3.5E-5 /rcryr, which is a significant decrease of total CDF for **75%** (reduction factor of nearly **4**), when compared to the baseline total CDF (32% in Phase 2 plus 43% in Phase 3). (Note: Overall RF of nearly 4 equals to product of RF of 1.5 after Phase 2 and RF of 2.7 after Phase 3.)

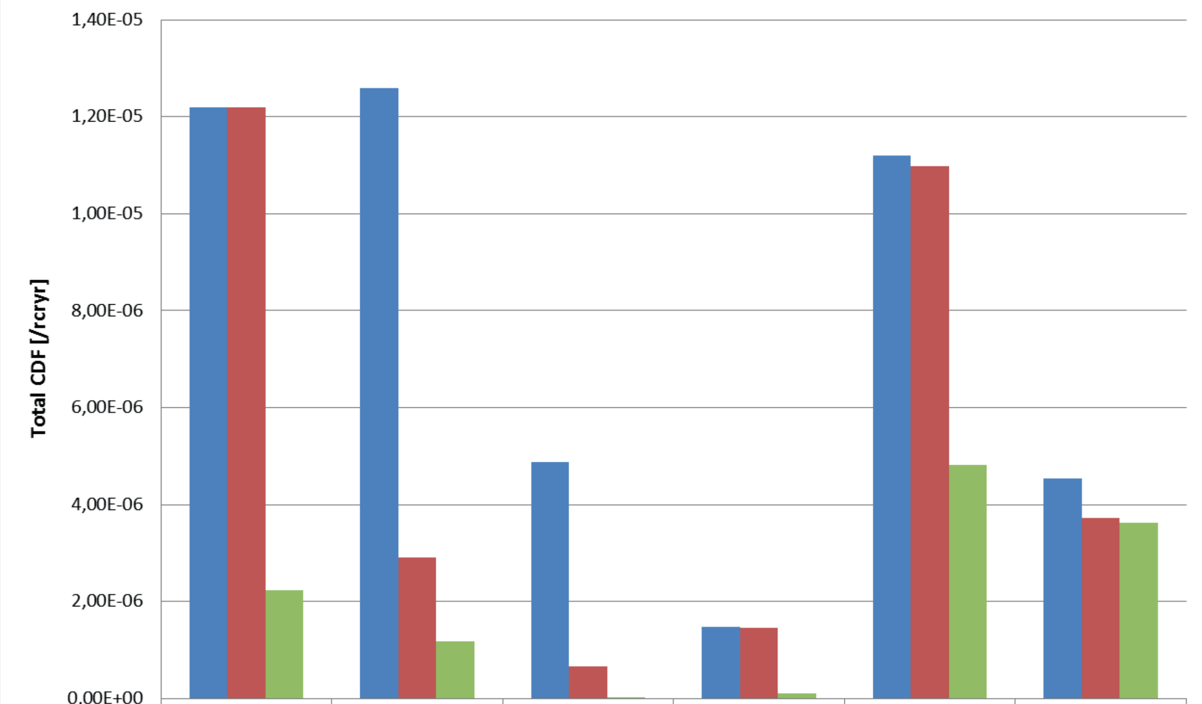
Upon a completion of Phases 2 and 3 the most dominant total CDF reduction is observed for internal fire events in absolute value of 1.1E-5 /rcryr, which represents reduction of baseline total CDF for 24%. This reduction is primarily due to installation of the Emergency Control Room as part of BB1 project in Phase 2 (21%).

The second largest contributor to the reduction of total CDF are IIEs, for which a reduction about 1.0E-5 /rcryr is obtained, which represents reduction of baseline total CDF for 21%. All of it comes from the modifications in Phase 2, primarily due to installation A-AF and A-SI pumps and associated tanks (A-CYT and A-BWT) in BB2 building.

The third largest contributor to the reduction of total CDF are seismic events, for which an absolute reduction is about 6.4E-6 /rcryr, which represents reduction of baseline total CDF for 14%. This comes from the modifications in Phase 3, due to construction of an interconnections between BB1 and BB2 buildings and interconnections between BB2 building and NSSS island, which are seismically designed to withstand PGA of 0.6*g* and are resistant to liquefaction.

The fourth largest contributor to the reduction of total CDF are internal floods, for which an absolute reduction is about 4.9E-6 /rcryr, which represents reduction of baseline total CDF for 10%. This reduction is primarily due to installation of ESW pumps shielding against water spraying in Phase 2 (9%).

Figure 1 provides comparison of CDF profiles for all three analyzed cases (posterior to Phase 1 (baseline), posterior to Phase 2 and posterior to Phase 3). Figure 2 illustrates the comparison of fractional contributions CDF per contributors (initiators' groups) for the same three analyzed cases.



	IIE	INT. FIRE	INT. FLOOD	HELB	SEISMIC	OEE
Baseline	1,22E-05	1,26E-05	4,88E-06	1,48E-06	1,12E-05	4,54E-06
SUP Phase 2	1,22E-05	2,90E-06	6,71E-07	1,46E-06	1,10E-05	3,73E-06
SUP Phase 3	2,22E-06	1,18E-06	2,86E-08	1,05E-07	4,81E-06	3,63E-06

Figure 1: Comparison of Total CDF Profiles with Phases 2 and 3 Addressed vs. Phase 1 Addressed

Due to significant reduction of baseline total CDF for internal fire (≈ 24%) and internal initiating events (≈ 21%), then moderate reduction of baseline total CDF for seismic events (≈ 13%), and floods (≈ 10%), and negligible baseline total CDF changes from HELB events (≈ 3%) and other external events (≈ 2%), a noticeable change in fractional contribution of initiators to total CDF posterior to Phase 3 is observed. Therefore, the largest total CDF contributor posterior to Phase 3 are seismic events with contribution of around 40% (before Phase 2 it was ≈ 24%), and other external events became the second largest with contribution of around 30% (before Phase 2 it was ≈ 10%).

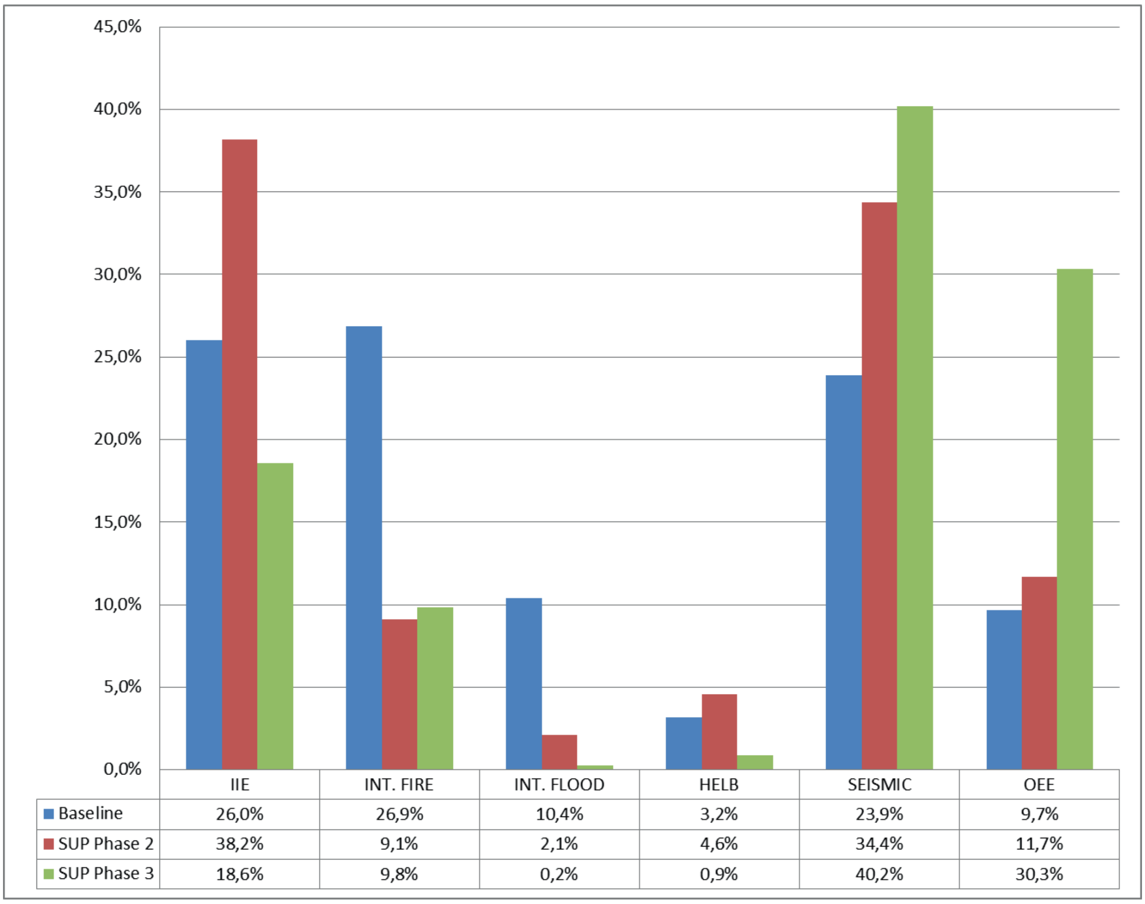


Figure 2: Fractional Contribution of Initiators to Total CDF (SUP Phase 2 and 3 Addressed vs. Phase 1 Addressed)

An additional step was taken to get an insight and evaluate contribution of various events within the category of other external events. There were changes observed in contribution to CDF for two OEE subcategories:

(1) *External flooding*. An upgrade of flooding protection of NSSS island is planned for implementation in Phase 2. New design would ensure flood protection even in case of Sava River bank failure. The required margin level is 40 cm (upper edge of the existing river bank plus 40 cm yields protection to 157.53 m above sea level). This measure reduces CDF due to external flooding for factor of 5, consequently reducing external flooding contribution to total CDF due to OEE from 22% (before Phase 2) to 6% (after Phase 3).

(2) *Aircraft accidents*. Any civil adaptation to existing BB1 building or BB2 building planned for construction in Phase 3 would be designed and reinforced to be capable to sustain events related to external large commercial aircraft impact on the plant (dynamic forcing function equivalent to B.5.b U.S. NRC requirement or European Utility Requirements for LWR NPP and large fires) and would assure functionality of equipment located inside these buildings during and after DEC. Hence, a reduction of CDF for factor of 2 due to the aircraft accident is estimated, causing a minor change in aircraft accidents contribution to total CDF due to OEE from 4.4% (before Phase 2) to 2.8% (after Phase 3).

These two measures have very low impact on the reduction of the baseline total CDF of 1.93%. However, the largest contributor to the CDF due to OEE remains severe winds (tornados), whose contribution is shifted from 55% to 69% due to implementation of measures described in two paragraphs above.

3.2.2 Most Dominant Core Damage Sequences Posterior to SUP Phase 3

An additional analysis of the core damage sequences due to all initiating events covered by the scope of NEK at-power PSA model was performed. The core damage sequences with frequency above 2E-7 /rcryr for the plant status with Phase 3 implemented are listed in Table 4. There are 12 core damage sequences with frequency higher than or about 2E-7 per rcryr. Their absolute contribution is 1.0E-5 /rcryr, which makes approximately 85% of total CDF.

Table 4 Most Dominant Initiators Contributing to the Total CDF Posterior to SUP Phase 3

	Initiating Event Description	**Abs. Contribution to Total CDF [/rcryr]**	**Rel. Contribution to Total CDF [%]**
1	High winds – tornado strikes (induced station blackout)	2.50E-6	20.9%
2	Seismically induced transient without reactor trip	1.50E-6	12.5%
3	Earthquakes with PGA over 1.1*g*	1.27E-6	10.6%
4	Fire in the main control room	1.08E-6	9.0%
5	Steam line break due to IIE	9.18E-7	7.7%
6	Seismically induced station blackout	9.15E-7	7.6%
7	Seismically induced loss of offsite power	5.97E-7	5.0%
8	RPV failure due to IIE	3.42E-7	2.9%
9	SG tube rupture due to IIE	3.32E-7	2.8%
10	Seismically induced liquefaction (induced loss of ESW system)	2.80E-7	2.3%
11	Seismically induced loss of ESW system	2.04E-7	1.7%
12	External flooding	2.00E-7	1.7%
	...	...	...
	Total	***1.20E-5***	***100.0%***

There are 4 core damage sequences, with frequency above the value of 1E-6 /rcryr. They contribute roughly 53% to the total CDF. They stem from the following initiator categories:

a) *Severe winds – tornados (induced station blackout)* (contribution of 20.9% to total CDF)
It is possible to reduce the risk due to tornados by installation of shielding for coolers and air suction of Diesel Generator #3 against the missiles, which may be generated by tornadoes.

b) *Seismically induced transient without reactor trip* (contribution of 12.5% to total CDF)
It is not possible to reduce the risk from seismically induced transient without reactor trip. The analysis has shown that more than 98% of the seismic hazard comes from the area within the radius of 25 km from the plant. Thus, given the seismically triggered reactor trip protection would be successful for the earthquakes distanced from NEK for more than 28 km, it can be concluded that seismically triggered reactor trip protection is in most cases insufficient (too slow, considering the seismic waves propagation and time needed to successfully trip the reactor) and, therefore, implementation would not significantly improve the plant's seismic safety.

c) *Earthquakes with PGA over 1.1g* (contribution of 10.6% to total CDF)
It is not possible to reduce the risk from the earthquake with PGA over 1.1*g*. The frequency of earthquakes above 1.1*g* PGA is a natural characteristic of the plant site, evaluated by the seismic risk studies for NEK.

d) *Fire in the Main Control Room (MCR)* (contribution of 9.0% to total CDF)
A more detailed analysis should evaluate the residual risk due to fire upon construction of ECR. In this analysis the fire risk in the MCR was reduced for 90% (from 1.08E-5 to 1.08E-6) due to construction of ECR.

4 CONCLUSION

In this paper a Level 1 PSA was performed on NEK at-power PSA model in order to evaluate the impact of plant modernization and upgrades, Phases 2 and 3 of the NEK Safety Upgrade Program, on the total core damage frequency. Analysis was also used to identify the most dominant measures for reduction of total CDF.

Total CDF posterior to the implementation of Phase 3 was estimated at 1.20E-5 /rcryr, which represents a reduction of 63% (a reduction factor of 2.7) as compared to the total CDF obtained posterior to the Phase 2. The measures planned for implementation in Phase 3 with the most significant contribution to reduction of baseline total CDF are (1) installation of additional pumps for primary and secondary injection (A-SI and A-AF) and associated tanks (A-BWT and A-CYT) (21%), and (2) construction of in interconnections between BB1 and BB2 buildings and interconnections between BB2 building and NSSS island (13%).

Cumulative reduction of total CDF after Phase 3, when compared to the baseline total CDF (4.69E-5 /rcryr), is about 3.5E-5 /rcryr, which represent a significant reduction of total CDF for 75% (reduction factor of nearly 4). Upon a completion of Phase 3 the most dominant reduction of baseline total CDF for 24% is observed for internal fire events. This is primarily due to installation of the ECR as part of BB1 project in Phase 2 (21%). The second largest contribution comes from internal initiating events (reduction of baseline total CDF for 21%). It all comes from the modifications in Phase 2, primarily due to installation A-AF and A-SI pumps and associated tanks (A-CYT and A-BWT) in BB2 building. The third largest contribution is from seismic events (reduction of baseline total CDF for 14%). This comes from the modifications in Phase 3, due to construction of an interconnections between BB1 and BB2 buildings and interconnections between BB2 building and NSSS island.

A noticeable change in fractional contribution of initiators' groups to total CDF posterior to Phase 3 is observed. The largest total CDF contributor posterior to Phase 3 are seismic events with contribution of around 40% (before Phase 2 this was ≈ 24%), and other external events became the second largest with contribution of around 30% (before Phase 2 this was ≈ 10%).

As for the other external events, two measures, an upgrade of flooding protection of NSSS island (external flooding) and design change during civil adaptation of existing BB1 building or design of BB2 building capable to sustain events related to external large commercial aircraft impact (aircraft accidents), have very low impact on reduction of the baseline total CDF of 1.93%. High winds category remains the largest contributor to the CDF due to OEE, and its fractional contribution to OEE is shifted from 55% to 69%.

To conclude, a continuous two decade trend of total CDF reduction is present at NEK and is foreseen to be lowered even more in mid-term by additional safety measures and plant modernization defined by the scope of NEK SUP.

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