Treatment of Phenomenological Uncertainties in Level 2 PSA for Nordic BWR Using Risk Oriented Accident Analysis Methodology

Sergey Galushin[a], Anders Riber Marklund[b], Anders Olsson[c], Ola Bäckström[d], Dmitry Grishchenko[e], Pavel Kudinov[f]

[a]Vysus Group, Stockholm, Sweden, sergey.galushin@vysusgroup.com
[b]Vysus Group, Stockholm, Sweden, anders.ribermarklund@vysusgroup.com
[c]Vysus Group, Stockholm, Sweden, anders.olsson@vysusgroup.com
[d]Lloyd's Register, Stockholm, Sweden, ola.backstrom@lr.org
[e]Royal Institute of Technology (KTH), Stockholm, Sweden, dmitrygr@kth.se
[f]Royal Institute of Technology (KTH), Stockholm, Sweden, pkudinov@kth.se

Abstract: A comprehensive and robust assessment of phenomenological uncertainties is a challenge for the current real-life PSA L2 applications, since such uncertainty is majorly driven by physical phenomena and timing of events. Typically, the static PSA models are built on a pre-determined set of scenario parameters to describe the accident progression sequence and use a limited number of simulations in the underlying deterministic analysis to evaluate the consequences.

The Risk Oriented Accident Analysis Methodology (ROAAM+) has been developed to enable consistent and comprehensive treatment of both epistemic and aleatory sources of uncertainty in risk quantification. The framework is comprised of a set of deterministic models that simulate different stages of the accident progression, and a probabilistic platform that performs quantification of the uncertainty in conditional containment failure probability. This information is used for enhanced modeling in the PSA-L2 for improved definition of sequences, where information from the ROAAM is used to refine PSA model resolution regarding risk important accident scenario parameters, that can be modelled within the PSA.

This work presents an example of application of the dynamic approach in a large-scale PSA model and demonstrate the integration of the ROAAM+ results in the PSA model.

1. INTRODUCTION

Severe accident management in Nordic Boiling Water Reactors (BWR) relies on ex-vessel core debris coolability. In case of core melt and vessel failure, melt is poured into a deep pool of water located under the reactor (lower dry well (LDW)). The melt is expected to fragment, quench, and form a debris bed, coolable by natural circulation of water. Success of the strategy is contingent upon melt release conditions from the vessel which determine (i) properties and thus coolability of the bed, (ii) potential for energetic steam explosions. If decay heat cannot be removed from the debris bed, the debris can re-melt and attack containment basemat. Strong steam explosion can damage containment structures. Melt release conditions are recognized as the major source of uncertainty in quantification of the risk of containment failure in Nordic BWRs [1][2][9].

While conceptually simple, the strategy involves complex interactions between (i) stochastic scenarios of time dependent accident progressions, and (ii) deterministic phenomena, which make ex-vessel debris coolability and steam explosion issues intractable for separate probabilistic or deterministic analysis in Nordic BWR design.

A comprehensive and robust assessment of phenomenological uncertainties is a challenge for the current real-life PSA L2 applications, since such uncertainty is majorly driven by physical phenomena and timing of events. Typically, the static PSA models are built on a pre-determined set of scenario parameters to describe the accident progression sequence and use a limited number of simulations in the underlying deterministic analysis to evaluate the consequences. Furthermore, in case of a Nordic-type

BWR, there is no explicit modelling of phenomena in the PSA model. Instead, there is a function event where all phenomena are treated in a common fault tree with predefined probabilities of phenomena damaging the containment, that are based on (i) expert judgement, (ii) do not consider neither epistemic nor aleatory sources of uncertainty.

The Risk Oriented Accident Analysis Methodology (ROAAM+) has been developed to enable consistent and comprehensive treatment of both epistemic and aleatory sources of uncertainty in risk quantification [1][2][9]. The framework is comprised of a set of deterministic models that simulate different stages of the accident progression, and a probabilistic platform that performs quantification of the uncertainty in conditional containment probability. The results of ROAAM analysis are presented in form of distributions of conditional containment failure probabilities for given combinations of scenario parameters that can be considered within the PSA model.

The goal of this work is to use the ROAAM+ generated information for enhanced modelling in the PSA-L2. Specifically, it can be used for improved definition of the sequences to be modelled in PSA, where information from ROAAM can be used to refine PSA model resolution regarding risk important accident scenario parameters. Define respective values of probabilities of phenomena damaging the containment in different sequences given the current state of the art knowledge in epistemic uncertain parameters.

Not being the main focus of this work, the proposed approaches can provide insights on the effect of possible design modification on PSA results, taking into account different sources of uncertainties.

2. APPROACH

2.1. ROAAM Framework for Nordic BWR

This section gives a brief overview of the surrogate models used in ROAAM+ framework of Nordic BWR.

Melt Ejection Surrogate Model (MEM SM)

Melt ejection mode surrogate model (MEM SM) [3] is based on the uncertainty analysis results of vessel failure mode and melt release conditions in Nordic BWR [4][5] predicted by MELCOR code.

The MEM SM is built for the unmitigated SBO scenario with depressurization. Accident sequences initiated by the loss of offsite power has the highest contribution to the core damage frequency (CDF) according to PSA L1 analysis, and among the most challenging scenarios for boiling water reactors (BWRs), as illustrated at Fukushima-Daiichi accident.

It is important to note that the MEM SM predicts melt release conditions for four splinter scenarios represented by the combinations of the fraction of failed penetrations (*EIGT*) and the mode of debris ejection from the vessel (*IDEJ*), with respective surrogate models are considered in the analysis.

More details on the MEM SM, the list of input and output parameters can be found in [3].

Steam Explosion Surrogate Model (SEIM SM)

Steam explosion surrogate model (SEIM SM) has been developed for the assessment of the risk of containment failure due to steam explosion in Nordic-type BWRs. More details about SM development and implementation, data base of TEXAS-V code solutions can be found in [6].

Ex-vessel Debris Coolability Surrogate Model (DECO SM)

Analysis of ex-vessel debris coolability in ROAAM+ framework for Nordic BWR is subdivided into 3 subtasks: i) analysis of the effect of melt release conditions on jet breakup, droplets cooling and solidification and agglomeration of melt droplets – treated in Debris Agglomeration analysis (VAPEX SM and Agglomeration mode and Agglomeration SM (AGG SM) [7]; ii) ex-vessel debris spreading and coolability – threated in DECOSIM code (DECO SM) [12][13].

2.2. PSA Model of Nordic BWR

The reference PSA model is a generic full-scale PSA for a Nordic BWR. In the reference PSA model, the accident progression for PSA level 2 is modelled in a containment event tree, CET. In the CET there is no explicit modelling of phenomena. Instead, there is a function event where all the phenomena are treated in a common fault tree.

The probabilities for steam explosion resulting in containment failure are:

- 1E-3 - for low pressure melt through.
- 3E-3 - for high pressure melt through.

These values are always applied even if the lower drywell (LDW) flooding system fails. The rationale for this modelling is that no positive credit should be taken for system failures. Furthermore, there may be enough water for a steam explosion to occur but not enough to avoid melt through of the penetrations in the LDW floor. The probabilities for melt through of the penetrations in the LDW floor are:

- 1E-3 - for successful LDW flooding.
- 1.0 - for failure of the LDW flooding system.

The studied PDS in this work was HS2-TL4. This is a plant damage state where the initiating event is a transient or a CCI, core cooling has failed, and the reactor vessel pressure is low.

3. RESULTS

3.1. ROAAM Analysis Results

The ROAAM+ analysis results (see [10]) showed that the risk of containment failure due to ex-vessel steam explosion and ex-vessel debris coolability is mostly affected by the uncertainty in the water pool depth, which can be reflected in enhanced PSA modelling. Other parameters that have high contribution to the results (e.g., debris slope angle, velocity of falling debris, heat transfer coefficients between debris and IGTs, etc.) are epistemic modelling parameters and cannot be considered in ET/FT analysis. On the other hand, such parameters as (IDEJ1/IDEJ0) mode of debris ejection from the vessel are considered as phenomenological splinters, i.e., phenomenological scenarios where relevant epistemic uncertainties are beyond the reach of any reasonably verifiable quantification. These splinters will be treated in PSA analysis in the same manner as in the ROAAM+ framework.

Furthermore, water depth at “deep pool” is related to system functionality and can be calculated with deterministic models. If the LDW flooding system works, the LDW water level will be about 7-8 m. Thus, for the “deep pool” analysis cases in ROAAM+ framework MELCOR code predictions of the pool depth were used [4]. The water depth for shallow pool conditions is much more uncertain since this completely depends on the sequence, therefore in current implementation it was considered as intangible parameter on the specified range.

3.2. PSA Analysis Results

The reference PSA model containment event trees for the plant damage state HS2-TL4 was modified to consider the depth of the water pool in lower drywell (LDW) and respective ex-vessel phenomena, such as ex-vessel debris coolability (COOL) and ex-vessel steam explosion (STEX).

The water depth alternatives are (i) Deep water pool in LDW; (ii) Shallow water pool in LDW; (iii) No water in LDW.

Figure 1 shows an example of the sequences with explicit modelling of ex-vessel steam explosion and debris coolability (dashed red). For each end state in the CET there is a specific probability distribution generated with ROAAM+ framework for Nordic BWR [10].

Figure 1. Containment event tree sequences in enhanced PSA model of Nordic BWR with explicit modelling of containment phenomena (dashed red).

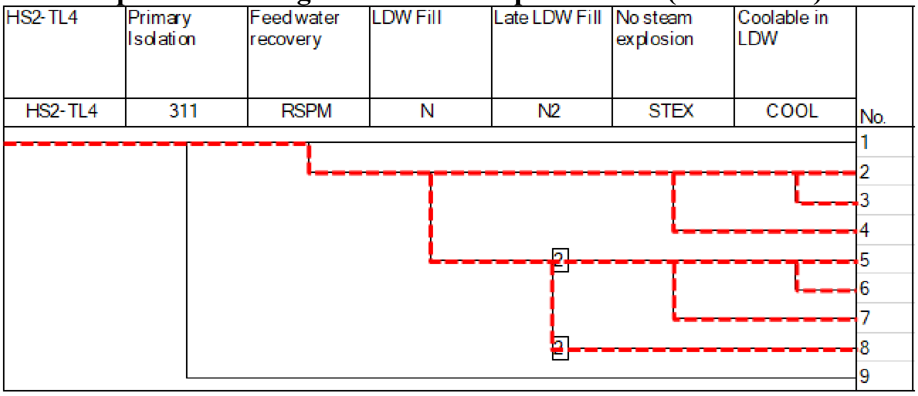


Figure 2. Treatment of containment phenomena in reference PSA model of Nordic BWR.

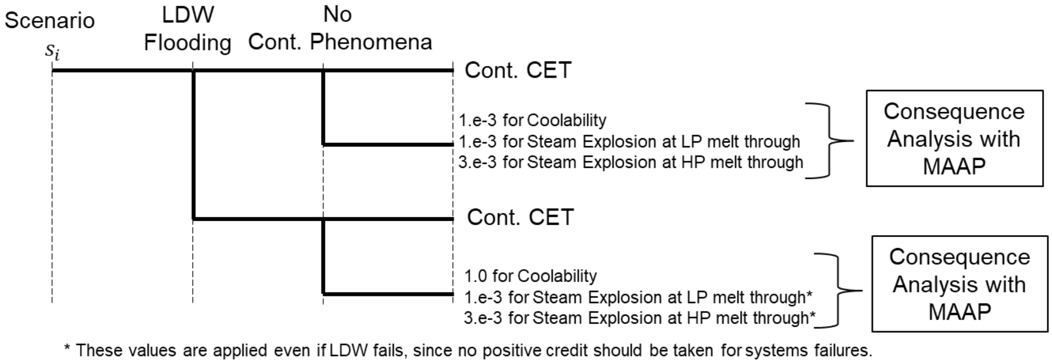


Table 1. RiskSpectum PSA simulation matrix.

Description	Debris ejection mode (splinter)	
	EIGT100-IDEJ1	EIGT100-IDEJ0
Deep pool (M)[a], Steam explosion load vs. fragility (Non-reinforced door).	CASEID_001D Mean: 1.0	CASEID_001D: Mean: 1.236e-2
Deep pool (M)[a], Steam explosion load vs. fragility (Reinforced door).	CASEID_002D Mean: 2.697e-1	CASEID_002D Mean: 0.0
Deep pool (M)[a], Debris non-coolable if max. temperature exceed 1700K (SS melting temperature).	CASEID_003D Mean: 6.047e-1	CASEID_003D Mean: 8.547e-3
Shallow pool (1-4m)[b], Steam explosion load vs. fragility (Non-reinforced door).	CASEID_001S Mean: 9.98e-1	CASEID_001S Mean: 1.647e-3
Shallow pool (1-4m)[b], Steam explosion load vs. fragility (Reinforced door).	CASEID_002S Mean: 7.144-e4	CASEID_002S Mean: 0.0
Shallow pool (1-4m)[b], Debris non-coolable if max. temperature exceed 1700K (SS melting temperature).	CASEID_003S Mean: 1.0	CASEID_003S Mean: 5.312e-1

Table 1 show the summary of the results of ROAAM+ analysis (see [10] for more details) and to be used in RiskSpectrum PSA analysis using enhanced model. Note that two fragility limits will be used for ex-vessel steam explosion, in order to evaluate the effect of design modifications on PSA analysis results:

- Non-reinforced hatch door, that can withstand 6kPa*s explosion impulses (original design).
- Reinforced hatch door, that can withstand 50kPa*s explosion impulses (modified design).

[a]. LDW water pool depth (XPW) for "Deep pool" was predicted by MELCOR code (MEM SM).
[b]. LDW water pool depth (XPW) for "Shallow pool" was considered as intangible parameter on the range from [1-4m].

In case of ex-vessel debris coolability – remelting of metallic debris (debris bed temperature exceeding stainless steel melting temperature (1700K)) was considered as a failure criterion.

3.3. Comparison

All transients and CCIs leading to the plant damage state HS2-TL4 (core damage due to inadequate coolant inventory make-up) were analysed for "non-contained release" group (include release categories leading to release frequencies over 0.1% of the core inventory of an 1800 MW BWR, including basemat melt-through sequences and bypass sequences).

The normalized results for medium values of non-contained release frequency per type of initiating event are shown in Figure 3 and 4. The normalization is done with respect to the frequency for non-contained release due to Loss of offsite power. The results show that the non-contained release frequency in most of the cases:

(i) increases in the enhanced model in case of EIGT100-IDEJ1, and
(ii) decreases in case of EIGT100-IDEJ0.

Figure 3. Comparison of normalized non-contained release frequencies between the reference and the enhanced models for EIGT100-IDEJ1 scenario with non-reinforced hatch door.

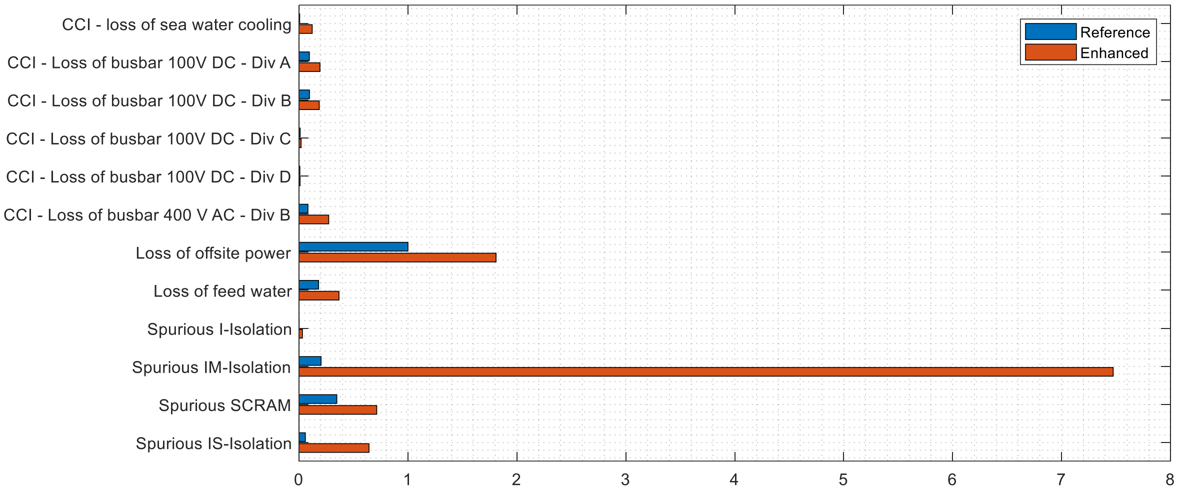
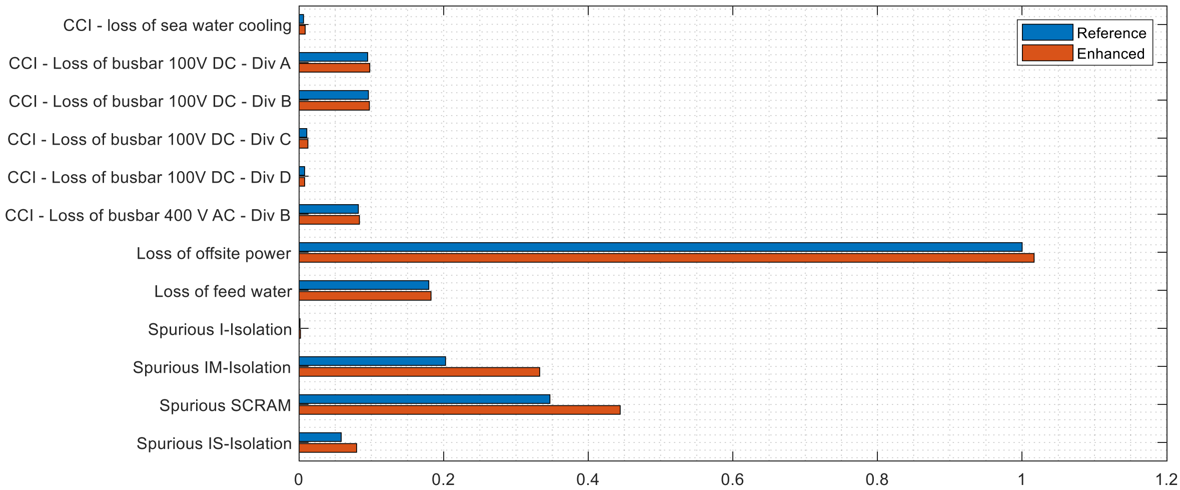


Figure 4. Comparison of normalized non-contained release frequencies between the reference and the enhanced models for EIGT100-IDEJ0 scenario with non-reinforced hatch door.



This behavior can be explained by the effect of the mode of debris ejection from the vessel on melt release conditions predicted by MELCOR code (MEM SM). It is clear from these results the high sensitivity of the non-contained release frequency to the mode of debris ejection from the vessel (see [4][5]).

Figure 5. Uncertainty analysis results of normalized non-contained release frequencies using enhanced model for EIGT100-IDEJ1 scenario with non-reinforced hatch door.

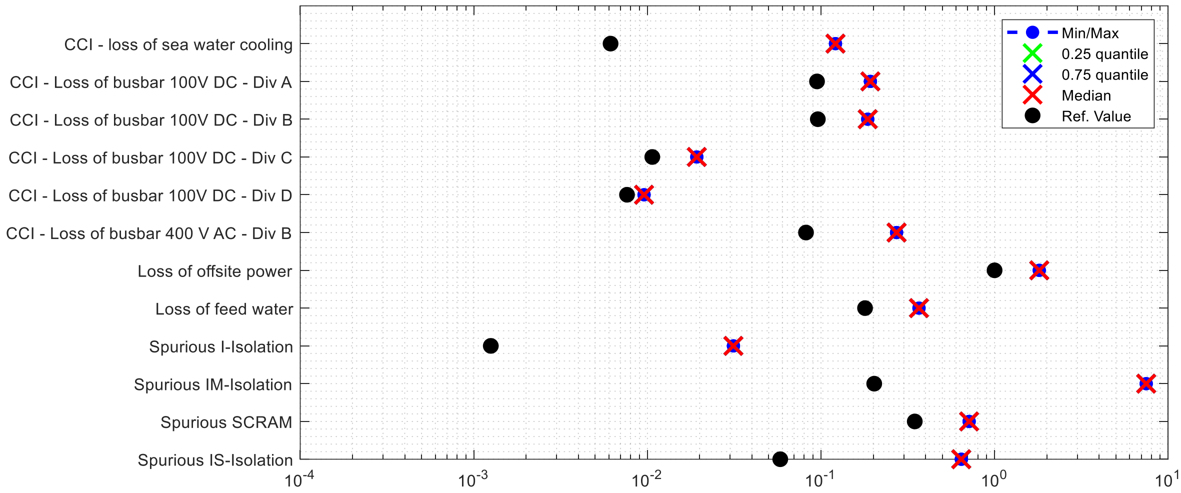


Figure 6. Uncertainty analysis results of normalized non-contained release frequencies using enhanced model for EIGT100-IDEJ0 scenario with non-reinforced hatch door.

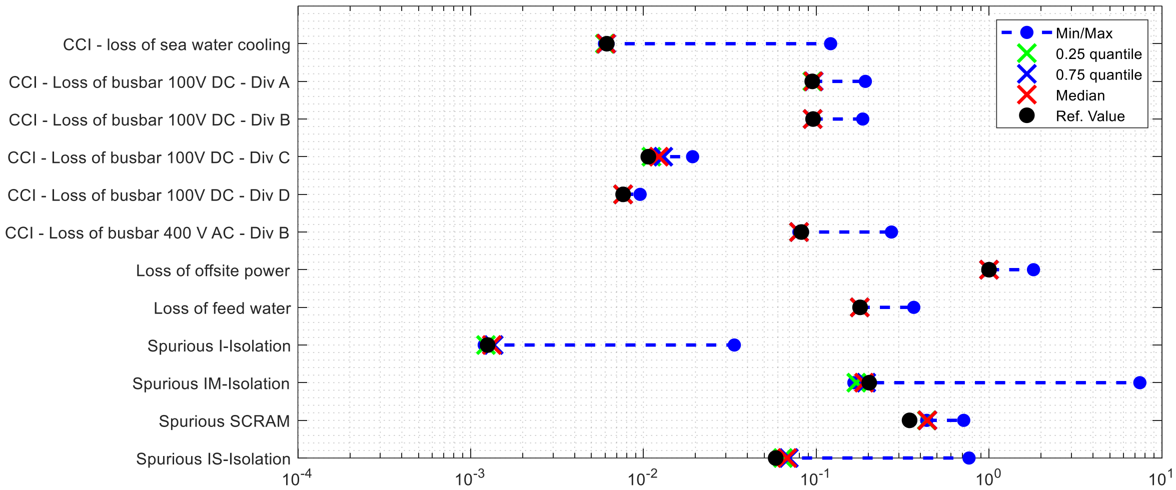


Figure 5 and 6 show the results of uncertainty analysis using ROAAM+ generated values of probability of containment failure due to ex-vessel debris coolability and steam explosion (see [10]). The results show that the uncertainty in the results depends on the mode of debris ejection from the vessel. For example, in case of IDEJ1 (solid debris ejection – OFF) the resultant distributions are concentrated around its expected values, which mean that the results of PSA analysis will not be significantly affected by the probability distributions for ex-vessel debris coolability, with exception to some scenarios with shallow pool conditions.

In case of IDEJ0 (Figure 6) the resultant distributions are very narrow; however, the right tails of the distributions can span over several orders of magnitude for some accident scenarios.

Furthermore, figures 5 and 6 show the values of normalized non-contained release frequencies obtained with reference PSA model of Nordic BWR (marked with black "•"). The results indicate that in case of IDEJ1 (Figure 5), the reference values lie outside the range of respective distributions generated with enhanced model.

In case of IDEJ0 (Figure 6) the reference values give very good estimates of non-contained release frequencies (e.g., judging by 0.25, 0.5, 0.75 quantiles of the distributions) in all initiating event groups.

CONCLUSIONS

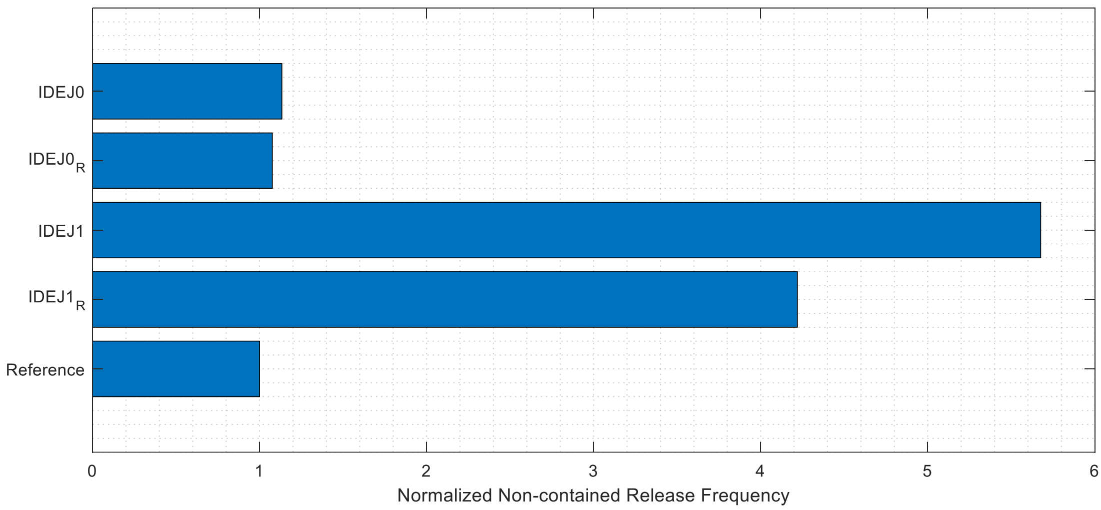
This paper presents an example of application of dynamic approached (such as ROAAM+) in PSA. In this approach the PSA was used as a basis to select important initiating events and sequences in the severe accident progression. These scenarios are then analyzed with the ROAAM+ tool, yielding information about which parameters that are of the highest importance for the development of the accident progression. The results from the deterministic analysis are used in the PSA to improve sequence definition as well as improve the estimation of phenomena depending on the sequence and the varied parameters. Furthermore, ROAAM+ framework provides an assessment of the effect of epistemic (knowledge) uncertainty on the results employing "knowledge-based treatment" of epistemic uncertain parameters, i.e., no probability distributions of epistemic uncertain parameters are assumed if there is no available knowledge about them.

In particular, probability of containment failure due to ex-vessel steam explosion and ex-vessel debris coolability strongly depends on debris ejection mode from the vessel (Solid debris ejection option – IDEJ1 vs IDEJ0). In case of solid debris ejection – off (IDEJ1) containment failure due to ex-vessel phenomena cannot be considered as physically unreasonable (for both threats). In case of solid debris ejection – on (IDEJ0), containment failure due to ex-vessel steam explosion can be considered as physically unreasonable only in case of modified design (with reinforced hatch door)). On the other hand, containment hatch door reinforcement does not affect the probability of containment failure due to basemat melt-through and this threat cannot be considered as physically unreasonable in case of IDEJ0.

The ROAAM+ analysis results show that the probability of phenomena damaging the containment significantly depend on the depth of the pool in the lower drywell, e.g., coolability increases with the depth of the pool, however opposite is true for steam explosion (i.e., higher energetics in the deep pool), this information was used in enhanced PSA modelling.

The results obtained with the enhanced PSA model suggest that the non-contained release frequency depends on the mode of debris ejection from the vessel (IDEJ). In case of IDEJ1, it results in ~5.6 times larger value of non-contained release frequency when compared to the results obtained with reference model. Reinforcement of the hatch door in case of IDEJ1 results in reduction of the non-contained release frequency from ~5.6 to 4.2 times the value obtained with the reference PSA model. In case of IDEJ0 (both reinforced and non-reinforced design) enhanced PSA modelling results show relatively low increase of non-contained release frequency, from ~8 to 13% for non-reinforced and reinforced hatch door respectively.

Figure 7: Expected value of Normalized Non-contained Release Frequency.



Overall results show that the values of probabilities of phenomena damaging the containment used in the reference PSA model can be underestimated, judged by the respective values predicted by ROAAM+ framework. On the other hand, if it can be demonstrated that the vessel LH failure will be limited to

IGTs failure and ablation of the opening will be limited, then the reference PSA model values of probabilities of phenomena damaging the containment can be considered as valid.

Present results show the dominant effect of the mode of debris ejection (IDEJ) on the results. However, given current state of knowledge about these phenomena, it should be treated as “phenomenological splinters” scenarios in PSA, that is, it should be demonstrated that non-contained release frequency is below regulatory requirements for all splinter scenarios considered.

It should be noted that current modelling approaches used in MELCOR code, for prediction of penetrations failure and melt and debris ejection, might be over-simplified in some aspects and lack necessary validation database. Furthermore, resent evidences from the Fukushima Daiichi Unit 2 and 3 primary containment vessel investigation [11], provided evidences that challenge ability of existing severe accident analysis tools to adequately predict transition of SA progression from in-vessel to ex-vessel phases in BWRs. Therefore, the results presented in this report should be considered as bounding estimates.

The approach has demonstrated that the vision, to develop the sequence from core melting, and to understand what are the important factors, is possible to meet. The integrated approach will have the ability to give a more comprehensive estimation of the uncertainty compared to the standard approach. The uncertainty related to phenomena will consider the interdependency between phenomena (all the way back to relevant deterministic and intangible, boundary conditions and scenario parameters.

ACKNOWLEDGMENT

The work is supported by the Swedish Nuclear Radiation Protection Authority (SSM), Swedish Power Companies, Nordic Nuclear Safety Research (NKS), Swiss Federal Nuclear Safety Inspectorate (ENSI) under the APRI-MSWI program at the Royal Institute of Technology (KTH), Stockholm, Sweden.

REFERENCES

[1] Galushin, S. Development of Risk Oriented Accident Analysis Methodology for Assessment of Effectiveness of Severe Accident Management Strategy in Nordic BWR, Royal Institute of Technology (KTH), urn:nbn:se:kth:diva-242353, (2019).

[2] S. Galushin, D. Grishchenko, P. Kudinov, “Implementation of Framework for Assessment of Severe Accident Management Effectiveness in Nordic BWR”, Reliability Engineering & System Safety, Vol 203, Article 107049, https://doi.org/10.1016/j.ress.2020.107049, November (2020).

[3] Galushin, S., Grishchenko, D., Kudinov, P. Surrogate Model Development for Prediction of Vessel Failure Mode and Melt Release Conditions in Nordic BWR based on MELCOR code. ICONE-27, 27th International Conference on Nuclear Engineering, Tsukuba, Ibaraki, Japan, May 19-24, (2019).

[4] Galushin, S., Kudinov, P. Uncertainty Analysis of Vessel Failure Mode and Melt Release in Station Blackout Scenario in Nordic BWR using MELCOR code. 18th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-18), Portland, OR, USA, August 18-22, (2019).

[5] Galushin, S., Kudinov, P. Sensitivity and Uncertainty Analysis of the Vessel Lower Head Failure Mode and Melt Release Conditions in Nordic BWR using MELCOR Code. Annals of Nuclear Energy, 135. 10.1016/j.anucene.2019.106976, (2020).

[6] Grishchenko, D., Basso, S., Kudinov, P. Development of a surrogate model for analysis of ex-vessel steam explosion in Nordic type BWRs. Nuclear Engineering and Design, Volume 310, 15 December 2016, Pages 311-327, (2016).

[7] Kudinov, P., Galushin, S., Davydov, M. Analysis of the Risk of Formation of Agglomerated Debris in Nordic BWRs. NUTHOS11, Gyeongju, South Korea, October 9-13, (2016).

[8] Kudinov, P., Galushin, S., Grishchenko, D., Yakush, S. Development of Risk Oriented Accident Analysis Methodology for Assessment of Ex-vessel Severe Accident Management Effectiveness. The 18th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-18), (2019).

[9] Kudinov, P., Galushin, S., Yakush, S., Villanueva, W., Phung, V. A., Grishchenko, D., Dinh, N. A framework for assessment of severe accident management effectiveness in Nordic BWR plants. PSAM 2014 - Probabilistic Safety Assessment and Management, June, (2014).
[10] NKS. Scenarios and Phenomena Affecting Risk of Containment Failure and Release Characteristics, NKS-428, ISBN 978-87-7893-518-2, (2019).
[11] TEPCO. 3-D Rendering of Images obtained during the Fukushima Daiichi Nuclear Power Station Unit 3 Primary Containment Vessel (PCV) Internal Investigation, (2018).
[12] Yakush, S. E., Konovalenko, A., Basso, S., Kudinov, P. Validation of DECOSIM Code Against Experiments on Particle Spreading by Two-Phase Flows in Water Pool. NUTHOS-11: The 11th International Topical Meeting on Nuclear Reactor Thermal Hydraulics, Operation and Safety, Gyeongju, Korea, October 9-13, (2016).
[13] Yakush, S. E., Kudinov, P. Analysis of the Risk of Formation of Agglomerated Debris in Nordic BWRs. NUTHOS11, Gyeongju, South Korea, October 9-13, (2016).