Joint Application of Risk Oriented Accident Analysis Methodology and PSA Level 2 to Severe Accident Issues in Nordic BWR

Sergey Galushin[a], Lisa Ranlöf[b], Ola Bäckström[b], Yvonne Adolfsson[b], Dmitry Grishchenko[a], Pavel Kudinov[a], Anders Riber Marklund[b]
[a] Royal Institute of Technology (KTH), Stockholm, Sweden
[b] Lloyd's Register Consulting, Stockholm, Sweden

Abstract: A comprehensive and robust assessment of severe accident management effectiveness in preventing unacceptable releases is a challenge for a today's real life PSA. This is mainly due to the fact that major uncertainty is determined by the physical phenomena and timing of the events. The static PSA is built on choosing scenario parameters to describe the accident progression sequence and typically uses a limited number of simulations in the underlying deterministic analysis.
Risk Oriented Accident Analysis Methodology framework (ROAAM+) is being developed in order to enable consistent and comprehensive treatment of both epistemic and aleatory uncertainties. The framework is based on a set of deterministic models that describe different stages of the accident progression. The results are presented in terms of distributions of conditional containment failure probabilities for given combinations of the scenario parameters. This information is used for enhanced modeling in the PSA-L2. Specifically, it includes improved definitions of the sequences determined by the physical phenomena rather than stochastic failures of the equipment, improved knowledge of timing in sequences and estimation of probabilities determined by the uncertainties in the phenomena.
In this work we present an example of application of the dynamic approach in a large scale PSA model and show that the integration of the ROAAM+ results and the PSA model can potentially lead to a considerable change in PSA Level 2 analysis results.

Keywords: SEVERE ACCIDENT, NORDIC BWR, ROAAM, PSA L2.

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].
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 (see Figure 1).
To address this issue, the Risk Oriented Accident Analysis Methodology framework (ROAAM+) is being developed in order to enable consistent and comprehensive treatment of both epistemic and aleatory uncertainties [1,2]. The ROAAM integrates risk assessment (analysis) and risk management (modifications in the design, procedures, etc.) and marries probabilistic and deterministic approaches. The framework is based on a set of deterministic models that describe different stages of the accident progression (see Figure 2). Surrogate modeling approach is employed in order to increase computational efficiency for extensive sensitivity and uncertainty analysis. The results are presented in terms of distributions of conditional containment failure probabilities for given combinations of the scenario parameters. This information is used for enhanced modeling in the PSA-L2. Specifically, it includes improved definitions of the sequences determined by the physical phenomena rather than stochastic failures of the equipment, improved knowledge of timing in sequences and estimation of probabilities determined by the uncertainties in the phenomena.

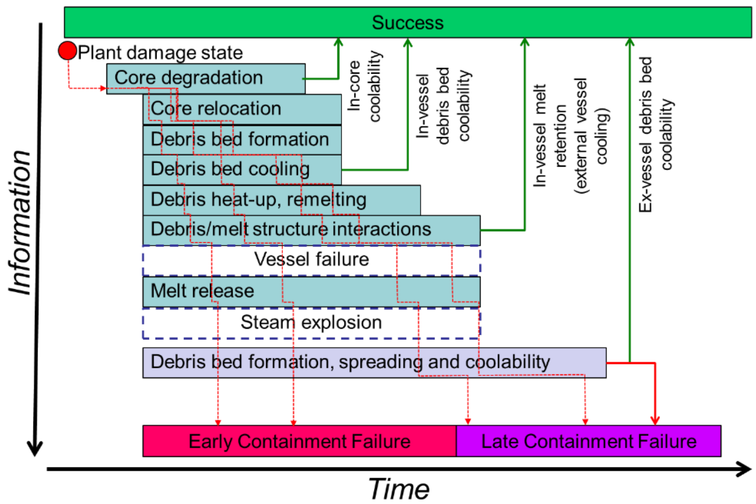


Figure 1. Severe accident progression in Nordic BWR [1].

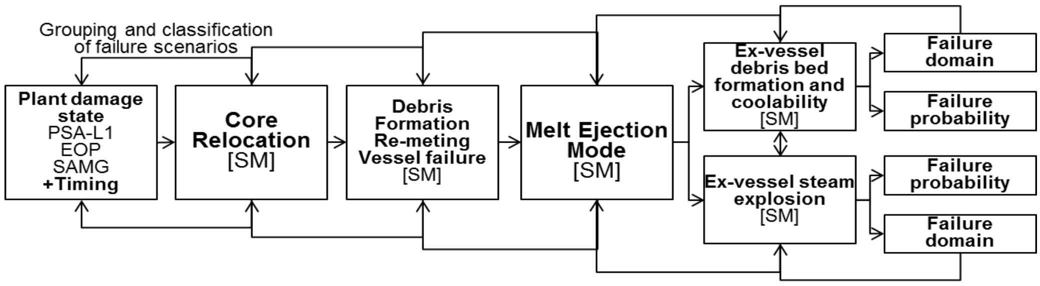


Figure 2. ROAAM+ framework for Nordic BWR [1].

In this work we present an example of application of the dynamic approach in a large scale PSA model and show that the integration of the ROAAM+ results and the PSA model can potentially lead to a considerable change in PSA Level 2 analysis results. The integrated approach also has the ability to give a more comprehensive estimation of the uncertainty compared to the standard approach and can improve current PSA scope and quality.

2. APPROACH

In this work a reference large scale PSA model was modified to consider the accident scenario parameters (such as depth of the water pool and the mass flow of corium at vessel melt through). The model was updated with regards to the containment event trees (CET) and scenario specific probabilities for the failure due to the containment phenomena.
This work aims at indicating the effect of taking the enhanced information about phenomena into account when calculating the large early release frequency for transients and CCI leading for these PDS. The analysis is performed for a few specific the plant damage states (PDS) corresponding to the core damage due to inadequate core coolant inventory makeup (initiating event is a transient or a CCI) at high pressure (HS2-TH1) and low pressure (HS2-TL4).
The first phase of an accident is studied in PSA L1 and the result is a number of sequences ending with either success or core damage. For those sequences ending with core damage the following accident progression is studied in PSA level 2 (L2). The enhanced information from a ROAAM+ is used in the PSA to (i) improve sequence definitions when phenomena can be relevant; (ii) estimation of probabilities for phenomena; (iii) improved knowledge of timing in sequences, which can be another base for improved realism in PSA quantification.
One example is how recovery of emergency cooling system (ECCS) and ADS might help to avoid more severe consequences. A successful recovery early in the sequence would allow the core to be arrested in the reactor pressure vessel (RPV) and hence provide the best possibility to limit the releases. The

human reliability analysis regarding recovery actions is based on the available time for the operator action.
In addition to a better representation of the sequences, it is important to improve our ability to estimate the probability that a certain phenomenon with risk significant consequences can occur.

2.1. Reference PSA Model

The reference case PSA model is a generic full scale PSA for Nordic BWR. In the reference PSA model the accident progression for PSA level 2 is modeled in a containment event tree, CET. In the CET there is no explicit modeling 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 modeling 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 feasibility study are:

- HS2-TH1. 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 still high (the automatic depressurization system, ADS, has failed).
- 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.

Furthermore, based on PSA L2&1 analysis the following assumptions has been made:

- After successful automatic opening of the LDW flooding system it is assumed that the LDW water level will be 7.8 m at reactor vessel melt through. This is according to MAAP calculations of HS2-TH1 and HS2-TL4 sequences.
- The probability for opening of LDW flooding is modelled in RiskSpectrum PSA Software.
- If automatic opening of LDW flooding fails it is assumed that the operators can manually take actions to fill the LDW. Possible actions are:
    - Manual opening of LDW flooding.
    - Manual start of the drywell spray system.
    - Manual start of the independent spray system.
- Successful manual start of LDW flooding is assumed to lead to shallow pool in LDW at reactor vessel melt through. The level for shallow pool is assumed to be 3.9 m. The probability for failure of manual flooding is assumed to be 0.1.

2.2. ROAAM Results

In ROAAM+ framework for Nordic BWR we use the concept of second-order probability in quantification of conditional containment failure probability. The need for the second-order probabilities comes from the realization of the nature of epistemic uncertainties in prediction of failure probability (i.e. partial probabilistic knowledge, [4]). Epistemic uncertain parameters in ROAAM+ framework are separated into two groups:

- Model deterministic parameters – complete probabilistic information (i.e. range and probability distribution).
- Model intangible parameters – incomplete or no probabilistic knowledge, one can only speculate regarding the possible range.

Since probabilities are designed to handle uncertainty, it would be logical to consider representing uncertain probabilities with probabilities. Thus, in order to assess the importance of the missing information about the distributions of intangible parameters we consider distributions as uncertain parameters. A space of possible probability distributions of the intangible parameters can be introduced. Each randomly selected set of distributions for the intangible parameters will result in a single value of failure probability P_f. Sampling in the space of the distributions for model intangible parameters will result in calculation of different possible values of P_f, including the bounding ones. A cumulative distribution function of $cdf(P_f)$ can be used to characterize confidence in prediction of P_f (see Figure 3).

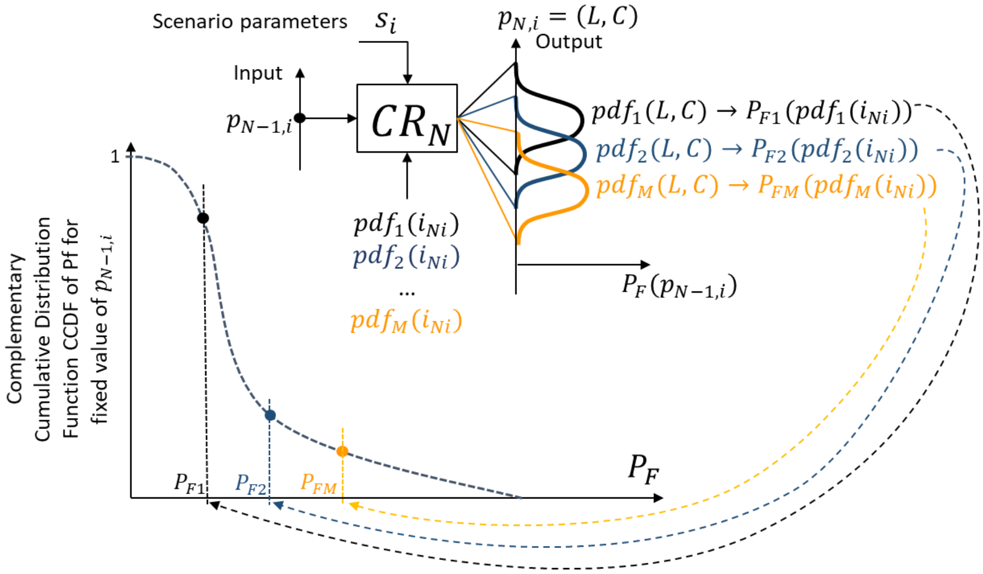


Figure 3. Treatment of model intangible parameters in ROAAM+ framework [2,3].

A distribution of probability of failure $pdf(P_{F_i})$ is obtained for every combination of accident scenario parameters (s_i) defined in ROAAM+ [1,2]. These distributions can be considered in decision support [3], regarding whether or not the risk associated with current SAM strategy is acceptable, such as presented in Figure 4 and 5, where decision criteria is based on Conditional Probability of Unacceptable Release (CPUR), and illustrated how $pdf(P_{Fi})$ (box and whiskers plots) can be used to judge regarding the impact of uncertainty on the SAM strategy effectiveness, and this information can be used in enhanced PSA model discussed in this paper.

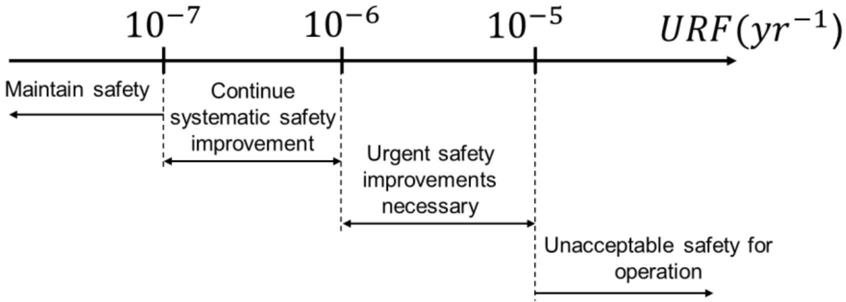


Figure 4. Unacceptable Release Frequency (URF(yr^{-1}) in classical ROAAM.

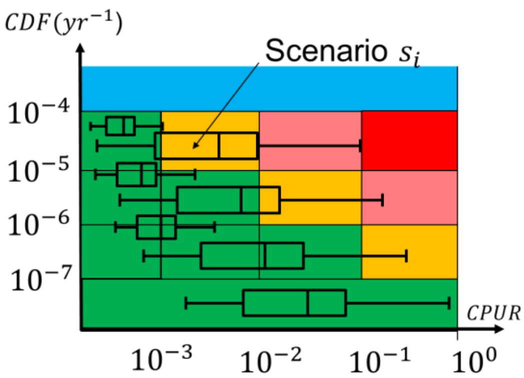
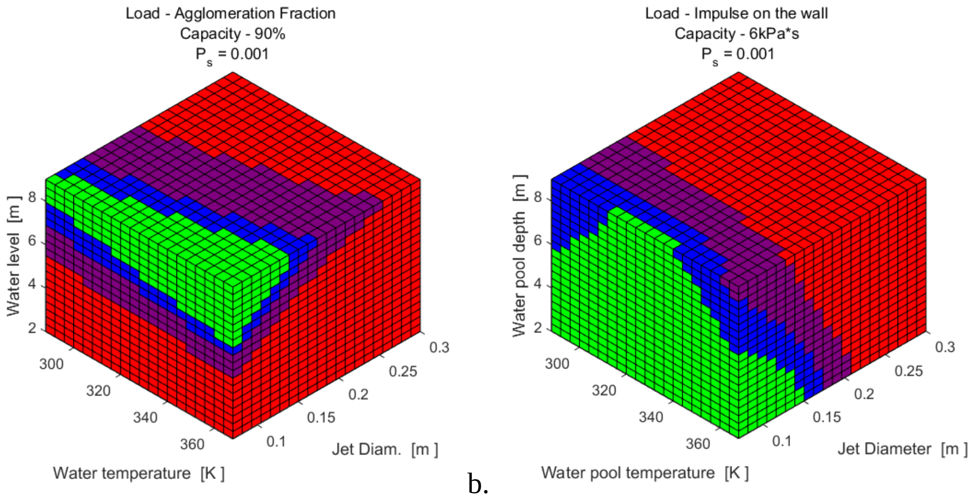


Figure 5. Decision support with ROAAM+, $pdf(P_{Fi})$ (box and whiskers plots) for scenario s_i. (green domain – remote and speculative sequences, orange domain – continue systematic safety

improvement, pale red domain – urgent safety improvements are necessary, red domain – unacceptable for safe operation).



a. b.

Figure 6. ROAAM+ Results for a) Ex-vessel Debris Agglomeration b) Ex-vessel Steam Explosion, as a function of accident scenarios s_i (Jet diameter (m), Lower drywell pool temperature (K), Lower drywell pool depth (m)).

Figure 6a,b – presents an example of the results of ROAAM+ for the analysis of containment failure probability P_{Fi} due to formation of non-coolable debris and ex-vessel steam explosion. Each cell in these 3-dimensional maps represent a single scenario s_i, where $pdf(P_{Fi})$ is color-coded according to the following criteria:

- CCDF $\{P_F \geq P_s\} \leq 0.05$ – Safe domain (Green).
- CCDF $\{P_F \geq P_s\} \in (0.05 - 0.5]$ – "Blue" subdomain – failure in less than half of the cases.
- CCDF $\{P_F \geq P_s\} \in (0.5 \ - 0.95]$ – "Purple" subdomain – failure in more than half of the cases.
- CCDF $\{P_F \geq P_s\} > 0.95$ – Failure domain (Red).

Furthermore, to illustrate an approach for use of ROAAM+ information in PSA analysis the following assumptions has been made regarding the lower drywell pool depth and melt release (jet diameter).

We assume that the lower drywell pool depth can be categorized into two categories: "Deep" if pool depth > 4m, or "Shallow" otherwise. Melt release from the vessel can be considered as: Dripping Mode (corresponds to IGT failure, Djet <75.e-3m), Medium release (ablated IGT, (75.e-3m <= Djet <150.e-3m), Large release (CRGT failure + ablated CRGT, Djet >= 150.e-3m). The temperature of the pool, according to PSA L1&2 analysis results is considered constant and equal 322K). Thus failure domain maps presented in Figure 6 can be represented by 6 modes (see Figure 7):

- Shallow pool + Dripping release (Mode 1).
- Shallow pool + Medium release (Mode 2).
- Shallow pool + Massive release (Mode 3).
- Deep pool + Dripping release (Mode 4).
- Deep pool + Large release (Mode 5).
- Deep pool + Medium release (Mode 6).

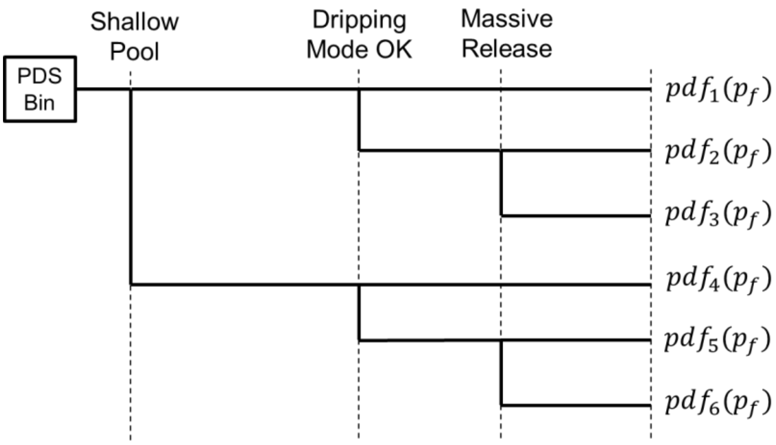


Figure 7. Refined CET with uncertainties

where each mode is characterized by a $pdf_n(P_F)$ e.g.

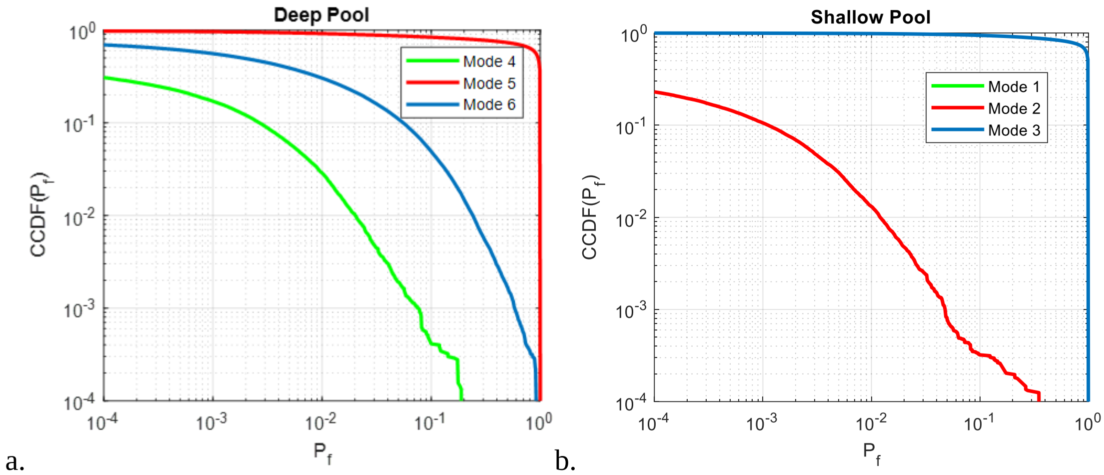


Figure 8. CCDF of P_F for ex-vessel steam explosion in different modes.

3. Results

3.1. Enhanced PSA Model

The reference large scale PSA model was modified to consider the depth of the water pool and the mass flow of corium at vessel melt through (parameters identified in ROAAM+ among the most influential [1,7,8,9]). The containment event trees for the plant damage states HS2-TH1 (high pressure) and HS2-TL4 (low pressure) are modified to consider the depth of the water pool in lower drywell (LDW) and the mass flow of corium at vessel melt through.

The water depth alternatives are:

i. Deep water pool in LDW;
ii. Shallow water pool in LDW;
iii. No water in LDW.

The melt flow alternatives correspond to the diameter of the melt jet:

i. Djet < 0.075 m – Dripping flow;
ii. 0.075 < Djet < 0.150 m – Medium Flow;
iii. Djet > 0.150 m – Large Flow.

For each combination of the water depth and melt flow there is a unique probability for steam explosion and non-coolable debris bed in LDW. This is explicitly modeled in the CET presented in Figure 9. ROAAM+ provides probability distributions $pdf_n(P_f)$ for containment damage by steam explosion and ex-vessel debris coolability given a certain combination of temperature, water depth and diameter of the melt jet.

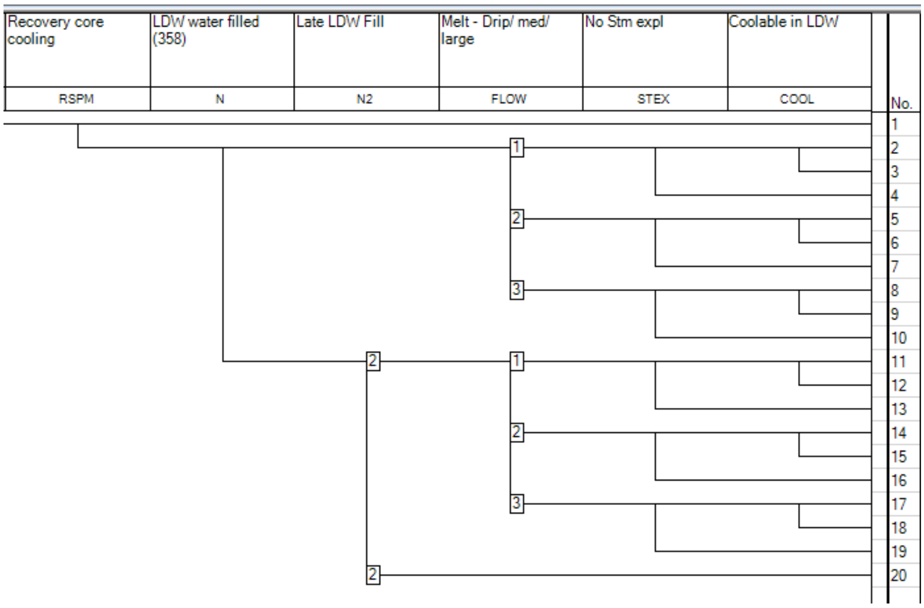


Figure 9. Containment Event Tree with explicit modeling of steam explosion and coolability [11].

For steam explosion the non-reinforced door (6 kPa*s) fragility was used. This gives the highest probabilities for the steam explosion damage of the containment structures. For ex-vessel debris coolability the 90 % agglomeration was used as a fragility limit, which might be an optimistic assumption.

If LDW flooding fails completely the following probabilities were assumed:

- Steam explosion 0.0;
- Debris bed not coolable 1.0.

At present there is no probability distribution for the different melt flow sizes so uniform distribution was used. The probabilities for steam explosion and non-coolability are calculated as the average value of different melt flows in each size respectively, given the depth and the temperature described above. This results in the following (see Table 1) probabilities for steam explosion and non-coolable debris bed in LDW.

Table 1: Expected Value of Conditional Containment Failure Probability in Enhanced PSA model.

Failure of containment due to	**Steam Expl.**	**Non. Coolable**
Deep pool, dripping flow	0	3.61E-02
Deep pool, medium flow	1.55E-02	2.83E-01
Deep pool, large flow	6.36E-01	8.52E-01
Shallow pool, dripping flow	0	1.0
Shallow pool, medium flow	3.60E-04	1.0
Shallow pool, large flow	3.78E-01	1.0

In the PSA model however, not the average values but the whole distributions of the containment damage probabilities obtained by the uncertainty analysis in ROAAM+ for each of the phenomenon were used. A non-standard interface, allowing the use of externally developed simulation data, was used in RiskSpectrum software to enable the uncertainty distribution for the phenomena to be consistently treated. It should be noted that the correlation between steam explosion and non-coolable debris bed has not been taken into account using the current data compilation from ROAAM+. This is due to the fact that the output from ROAAM+ and the current test case model built in RiskSpectrum has not fully taken all the data (the correlation) in consideration when the unacceptable release frequency is estimated. When performing an uncertainty analysis, samples were taken from the individual probability distributions for steam explosion and coolability, disregarding any correlation between the two.

All transients and CCIs leading to the plant damage states HS2-TH1 and HS2-TL4 were analyzed for all PSA Level 2 release categories. Release categories leading to the release frequencies over 0.1% of the core inventory of an 1800 MW BWR are grouped as non-acceptable. The normalized results for non-acceptable release per type of initiating event are shown in Figure 10 and Table 2. The results for the Loss of offsite power and non-acceptable release are set to 1.0 for the reference case and all the other results are divided by the same scaling factor.

The analysis shows that the non-acceptable release frequency is doubled in the enhanced model. The release frequency related to the release category "*Penetration of the LDW floor (basemat melt through)*" is shown in Table 3. The frequency approximately increases by a factor of 42 due to the increased probability for non-coolable debris bed. Note that basemat melt through is not grouped as a non-acceptable release. If this release category would be included the frequency for non-acceptable release would increase significantly.

The release frequency related to the release category "*Containment failure due to phenomena (always early and no DW spray is credited)*" is shown in Table 4. The frequency approximately increases with a factor of 4 due to the increased probability for steam explosion. The release frequency related to the release category "*Filtered release, Early opening, No DW spray*" decreases to 50 % of the reference case. The release frequency related to the remaining release categories changes only slightly between the reference model and the enhanced model.

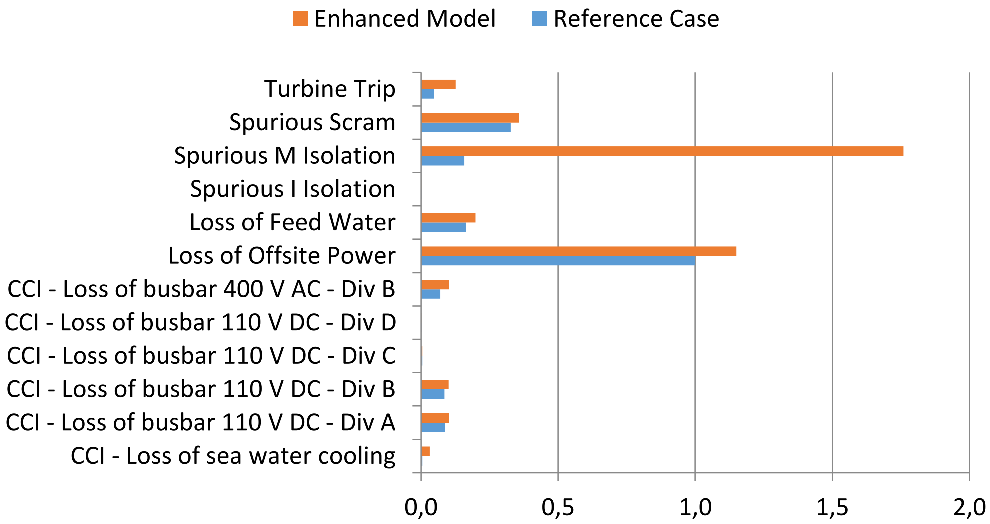


Figure 10. Comparison between the reference case and the modified model for non-acceptable release (normalized) [6].

Table 2: Comparison between the reference case and the modified model for non-acceptable release (normalized) [6].

Initiating event	Reference Case	Enhanced Model	Difference
CCI - Loss of sea water cooling	5.0E-03	3.2E-02	541%
CCI - Loss of busbar 110 V DC - Div A	8.7E-02	1.0E-01	19%
CCI - Loss of busbar 110 V DC - Div B	8.5E-02	1.0E-01	18%
CCI - Loss of busbar 110 V DC - Div C	4.7E-03	4.7E-03	0%
CCI - Loss of busbar 110 V DC - Div D	1.4E-03	1.2E-03	-16%
CCI - Loss of busbar 400 V AC - Div B	7.0E-02	1.0E-01	47%
Loss of Offsite Power	1.0E+00	1.2E+00	15%
Loss of Feed Water	1.7E-01	2.0E-01	21%
Spurious I Isolation	7.9E-04	1.7E-03	118%
Spurious M Isolation	1.6E-01	1.8E+00	1014%
Spurious Scram	3.3E-01	3.6E-01	9%
Turbine Trip	4.9E-02	1.3E-01	158%
Total result	**2.0E+00**	**3.9E+00**	**102%**

Table 3: Comparison between the reference case and the modified model for basemat melt through (normalized) [6].

Initiating event	Reference Case	Enhanced Model	Difference
CCI - Loss of sea water cooling		2.6E-02	
CCI - Loss of busbar 110 V DC - Div A	2.6E-03	1.7E-02	540%
CCI - Loss of busbar 110 V DC - Div B		1.5E-02	
CCI - Loss of busbar 110 V DC - Div C	2.7E-03	2.2E-03	-18%
CCI - Loss of busbar 110 V DC - Div D			
CCI - Loss of busbar 400 V AC - Div B		3.0E-02	
Loss of Offsite Power		1.4E-01	
Loss of Feed Water	7.0E-05	2.4E-02	33661%
Spurious I Isolation	7.0E-05	6.8E-04	874%
Spurious M Isolation	3.0E-02	1.6E+00	5256%
Spurious Scram	5.5E-03	4.2E-02	656%
Turbine Trip	5.3E-03	8.4E-02	1489%
Total result	**4.6E-02**	**2.0E+00**	**4179%**

Table 4: Comparison between the reference case and the modified model for containment failure due to phenomena (normalized) [6].

Initiating event	Reference Case	Enhanced Model	Difference
CCI - Loss of sea water cooling	4.8E-03	3.2E-02	561%
CCI - Loss of busbar 110 V DC - Div A	4.1E-03	2.1E-02	406%
CCI - Loss of busbar 110 V DC - Div B	1.8E-03	1.8E-02	873%
CCI - Loss of busbar 110 V DC - Div C	3.2E-03	3.2E-03	0%
CCI - Loss of busbar 110 V DC - Div D	2.4E-04	1.2E-4	-100%
CCI - Loss of busbar 400 V AC - Div B	1.5E-03	3.4E-02	2240%
Loss of Offsite Power	2.7E-01	4.2E-01	56%
Loss of Feed Water	6.6E-04	3.5E-02	5180%
Spurious I Isolation	6.6E-04	1.7E-03	150%
Spurious M Isolation	1.6E-01	1.8E+00	1035%
Spurious Scram	5.6E-02	8.8E-02	56%
Turbine Trip	4.4E-02	1.2E-01	178%
Total result	**5.4E-01**	**2.5E+00**	**370%**

The results of the uncertainty analysis for non-acceptable release are shown in Table 5. The results show that the uncertainty ranges from roughly half the point estimate frequency up to about 1.5 of the point estimate frequency. This is a reasonably narrow interval, which is positive – as the uncertainty is an important factor in PSA-L2. It could be relevant to further study the cases where the uncertainty range is greater – to understand if the uncertainty can be reduced.

Table 5: Uncertainty analysis for non-acceptable release (All the median values are normalized.) [6].

Initiating event	5%	median	95%
CCI - Loss of sea water cooling	56%	100%	158%
CCI - Loss of busbar 110 V DC - Div A	91%	100%	112%
CCI - Loss of busbar 110 V DC - Div B	91%	100%	112%
CCI - Loss of busbar 110 V DC - Div C	95%	100%	107%
CCI - Loss of busbar 110 V DC - Div D	100%	100%	100%
CCI - Loss of busbar 400 V AC - Div B	84%	100%	123%
Loss of Offsite Power	93%	100%	109%
Loss of Feed Water	91%	100%	112%
Spurious I Isolation	58%	100%	163%
Spurious M Isolation	54%	100%	161%
Spurious Scram	95%	100%	107%
Turbine Trip	66%	100%	147%

It can be noted that the initiating event group "Spurious M-isolation" (feed water lines) is much more affected by the enhanced modelling than the other initiating event types studied. To explain the reason for this, it can first be noted that the group of non-acceptable releases, for a BWR, to a relatively large extent contains so-called bypass sequences, in which closure (isolation) of the containment fails and the release path occurs through e.g. through open steam lines. Such sequences will not be affected by the ROAAM+ approach since they are not created by the studied containment rupture phenomena. M-isolation refers to a specific function of the reactor protection system, which initiates closing of isolation valves in the feed water lines. The effect of the initiating event is thereby at first sight similar to that of the loss of feed water transient. However, in the generic Nordic BWR plant design represented by the PSA model used in this study, M-isolation automatically activates another isolation function that initiates closure of the steam lines. This implies that for sequences starting with spurious M-isolation, bypass sequences through open steam lines are directly excluded (apart from cases with mechanical errors in the MSIVs) and this category of initiating events becomes the only category where the ROAAM+ methodology will influence all the resulting accident sequences. In contrast, e.g. the loss of feed water initiating event category has a relatively low frequency, which implies that together with the event probabilities prescribed by the ROAAM+ methodology, sequences affected by ROAAM+ to a large extent end up below the cut-off frequency of the PSA analysis, leaving almost only bypass sequences above it. In summary, loss of feed water sequences will in this model be minimally affected by ROAAM+ while the inverse is true for spurious M-isolation sequences, thereby creating a large difference between these seemingly similar initiating event families.

There is a number of assumptions and limitations in the implementation on the ROAAM+ results in the enhanced PSA model that influence the result which discussed in [6]. There is a need to make the feasibility study more realistic regarding some of the related parameters discussed in [6]. The quantitative results should, therefore, be seen as indicative.

4. CONCLUSIONS

A dynamic approach to PSA can significantly influence the analysis is several ways. The analysis shown in this paper is an example of a dynamic approach where the PSA is used as a basis to select important initiating events and sequences in the severe accident progression. These scenarios are then analyzed with a dynamic deterministic model yielding information about which parameters that are of high 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 uncertain parameters.

The integrated approach requires improvement in especially scenario definition, which practically leads to more plant damage states. The PDS should consider all necessary scenario parameters, that may affect the calculation of phenomena and hence consider also the system availability normally represented within CETs.

The implementation of the dynamic approach in this study in a large scale PSA model shows that the integration of the ROAAM+ results and the PSA model is not only feasible, but could potentially lead to a considerable change of the frequency for non-acceptable release. The results show that the parameters indicated by the dynamic approach as being of high importance to the results are indeed of high importance to the quantitative results. It also emphasizes the need to distinguish between different probabilities of phenomena depending on different scenario, physical and intangible parameters.

The integrated approach will also 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 intangible and physical parameters, and of course scenario parameters).

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References

1. Pavel Kudinov, Sergey Galushin, Dmitry Grishchenko, Sergey Yakush, Simone Basso, Alexander Konovalenko, Mikhail Davydov, "Application of Integrated Deterministic-Probabilistic Safety Analysis to Assessment of Severe Accident Management Effectiveness in Nordic BWRs," The 17th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-17) Paper: 21590, Qujiang Int'l Conference Center, Xi'an, China, September 3-8, 2017.
2. P. Kudinov, S. Galushin, W. Villanueva, S. Yakush, D. Grishenko, Development of Risk Oriented Approach for Assessment of Severe Accident Management Effectiveness in Nordic BWR, PSAM12, Hawaii, USA, June (2014).
3. S. Galushin, D. Grishchenko, P. Kudinov, "Risk Analysis Framework for Severe Accident Mitigation Strategy in Nordic BWR: An Approach to Communication and Decision Making", 2017 International Topical Meeting on Probabilistic Safety Assessment and Analysis (PSA 2017), Pittsburgh, PA, September 24-29, (2017).
4. S.O. Hansson, "An Overview of Decision Theory", SKN Report 41, (1990).
5. H. Tuomisto, "ROAAM Methodology and its Application to the SAM Strategy Development at the Loviisa Plant" *APRI 6 – Seminarium Svåra haverier & Haverihantering,* Lejondal, Sweden, 22-23 January (2009)
6. Kudinov P., Galushin S., Grishchenko D., Yakush S., Adolfsson Y., Ranlöf L., Bäckström O., Enerholm A., "Scenarios and Phenomena Affecting Risk of Containment Failure and Release Characteristics," SPARC Project, NKS-395 22 Aug 2017, 2017.
7. Grishchenko D., Galushin S., Basso S., Kudinov P., "Failure Domain Analysis and Uncertainty Quantification using Surrogate Models for Steam Explosion in Nordic type BWRs", 17th International Topical Meeting on Nuclear Reactor Thermal Hydraulics, Xi'an, Shaanxi, China, Sept. 3-8, (2017).
8. Grishchenko D., Galushin S., Basso S., Kudinov P., "Application of TEXAS-V Surrogate Model to Assessment of the Containment Failure Risk Due to Steam Explosion in a Nordic type BWR", Gyeongju, South Korea, October 9-13, (2016).
9. P. Kudinov, S. Galushin, M. Davydov, "Analysis of the Risk of Formation of Agglomerated Debris in Nordic BWRs", NUTHOS11, Gyeongju, South Korea, October 9-13, (2016).