CREEP RUPTURE PROBABILITY EVALUATION FOR RCS COMPONENTS IN SEVERE ACCIDENTS; AN IDPSA APPROACH

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Accident progression event tree (APET) is among the tools for the quantification of large early release frequency in level-2 PSA of nuclear power plants. The top events in APET structure represent phenomenological questions associated with the plant response to postulated severe accidents. Evaluation of the probability of each phenomenon is therefore of major importance to level-2 PSA. Here we propose an IDPSA approach for evaluation of the probability of creep rupture of major RCS components in high pressure sequences of severe accidents in PWR type NPPs. The proposed approach uses the obtained results from deterministic calculation of severe accident scenarios (e.g. by MELCOR code). The next step identifies 2 categories of uncertainties inherent in the level-2 PSA model, namely epistemic and aleatory uncertainties. Then it calculates the probability of creep rupture through an uncertainty analysis taking into account both categories of the uncertainty. A code is developed in MATLAB to integrate the deterministic and probabilistic calculations. The obtained results are two-fold in which it represents the deterministic characteristics of the plant in station black out accident as well as the probabilistic estimation of the creep rupture for major RCS components. The final results are presented as so called fragility curves where the cumulative probability of creep rupture is estimated as a function of damage function.

I. INTRODUCTION

Integrated deterministic and probabilistic safety assessment (IDPSA) is a daily challenge to the safety community which identifies the progression of accident scenarios in complex dynamic systems like nuclear power plants (NPPs). It takes into account interactions of complex systems, the physical phenomena, human actions and safety systems. Here, we employ an IDPSA approach for evaluation of the creep rupture probability of major RCS components in high pressure sequences of severe accidents. Level 2 probabilistic safety assessment (PSA) deals with phenomenological uncertainties by integrating the deterministic severe accident analysis (e.g. MELCOR code calculations) with accident progression analysis. Figure 1 presents accident progression event tree (APET) of the plant under study. The main objective for accident progression analysis in support of level-2 PSA is to understand the plant response to severe accident. Concretely, the goal of such analysis is two-fold for each severe accident sequence:

- Whether temperature induced SGTR is possible or not.
- Whether core damage could be arrested without vessel breach.

Uncertainties of the second task are captured through the probabilistic modeling of the severe accident management strategies in the PSA model. In fact this is a PSA related task rather than being a deterministic issue. The severe accident phenomena of interest are primarily those that have the potential for an early large release of radioactive materials to the environment through early containment failure or bypass. Therefore, one of the critical severe accident phenomena is creep rupture as it may affect the relative timing of major RCS component failures such as surge line, hot leg, and steam generator (SG) tubes. If the SG tubes fail first, bypass of the containment will happen. However, if the surge line or hot leg fails first, it will lead to rapid depressurization of the primary system and preclude SG tube rupture.

As a matter of fact, the other top events are not in the scope of the current paper since their probabilistic modeling is associated with system performance and/or human action which is accounted for through employment of fault tree approach. Here the main focus is on the quantification of uncertainties related to creep rupture of steam generator tube (as RCS component) in the harsh severe accident environment. The significance of this phenomenon is due to the fact that once it occurs it opens a direct path for the radionuclides to release into the environment. This is called containment bypass and is one of the main concerns regarding the safety of PWRs.

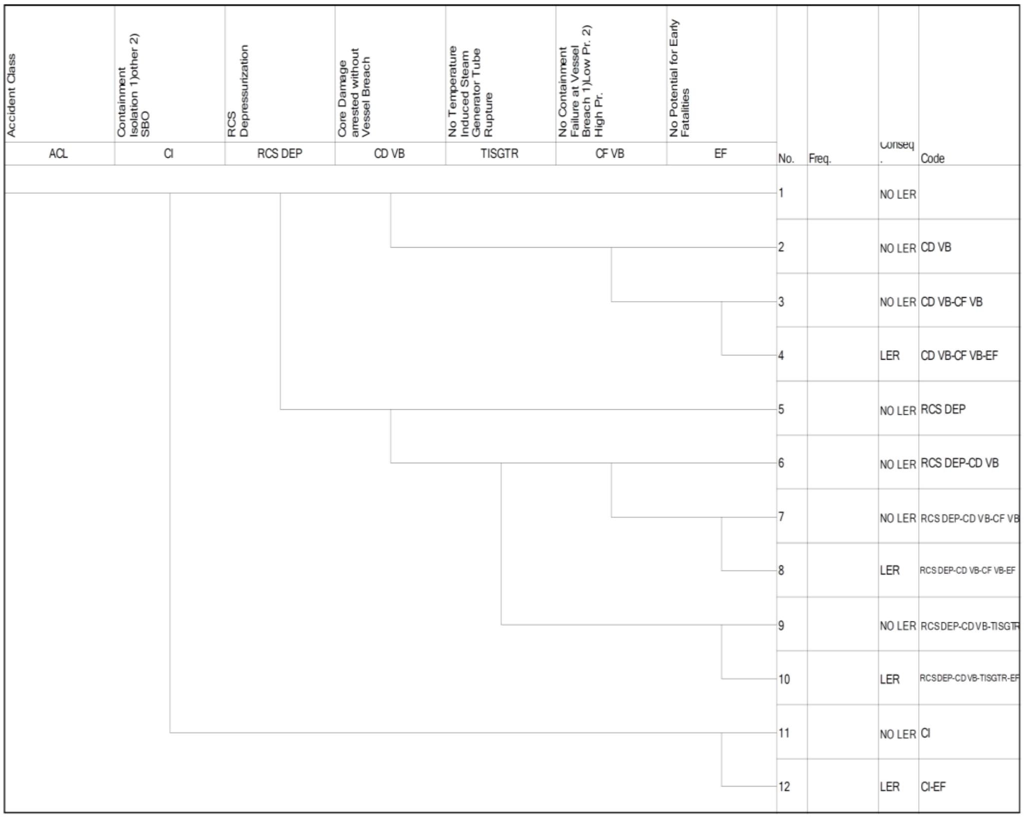

Figure 1: Accident Progression Event Tree

The paper organization is as follows. Section II explains the phenomenological uncertainties of level-2 PSA. Section III provides a Monte-Carlo algorithm for the uncertainty evaluation by taking into account aleatory as well as epistemic sources of uncertainty. Details of deterministic calculations by MELCOR code are discussed in section IV. The obtained results are given in section VI and finally section VII concludes the article.

II. Accident Progression Uncertainties and Level-2 PSA

Uncertainties are classified as aleatory and epistemic uncertainties. Aleatory uncertainties are associated with random or stochastic phenomena. Epistemic uncertainties are related to, or involving, knowledge, also called "state-of-knowledge uncertainty". Epistemic uncertainties are in different level of modeling [1]:

- Parameter Uncertainty - results from lack of knowledge about correct inputs to models
- Model Uncertainty - perfect models cannot be constructed
- Completeness Uncertainty - whether or not all scenarios, phenomena relationship are considered.

Issue of mixing aleatory and epistemic uncertainties has been posed in several technical meeting in problems where both kinds of uncertainties contribute to the total uncertainty of code calculation [2]. Through distinction, we know which part of uncertainty is removable or at least reducible and which part due to aleatory uncertainties is irreducible. Reference [3] indicates that it is important to distinguish between the two types of uncertainty, not only because it can impact the answer being given to a decision maker, and hence have an impact on the decision outcomes, but also because it is essential to truly understand the nature of the model of the world that is being incorporated in the PRA.

Central to any level-2 PSA is the containment event tree (CET) or Accident Progression Event Tree (APET) which displays the characteristics of the severe accident progression impacting the fission-product source term to the environment. CET is a time-line of accident progression and represents the sequence of events that could lead to failure of the containment pressure boundary and fission product release to the environment. It is a Probabilistic model in which:

- represents uncertainties in ability to predict accident progression
- Particular assumptions regarding each uncertainty lead to different conclusions regarding plant response to the sequence

Branch point probabilities typically are not based on statistical analysis of "data", instead they reflect confidence that one outcome is more likely to be correct than its alternative in the form of questions on whether some physical phenomena occur (see Figure 2). A number of physical parameters determine which of the two complementary answers is the right answer; however this answer is inevitably uncertain due to uncertainties inherent in the parameters, models and boundary conditions.

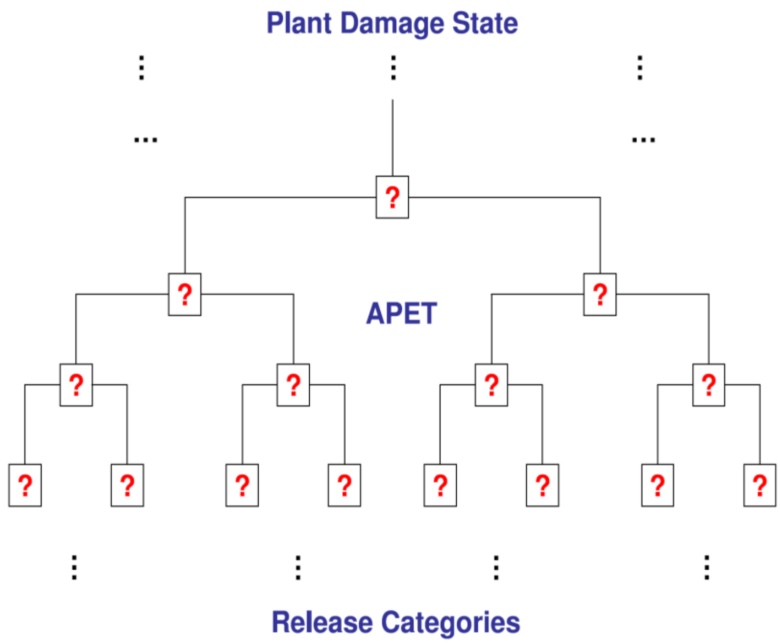

Figure 2: Nodes in APET Structure representing phenomenological questions [3]

In level-2 PSA, the common approach for the uncertainty quantification is to characterize the parameter's distribution through statistical analysis on the obtained results of the calculating tools like MELCOR and MAAP. In this process, randomness is represented by probability distribution functions in which the epistemic uncertainty is translated to our knowledge about the parameters of the distribution.

III. Monte Carlo algorithm for uncertainty evaluation; treatment of aleatory and epistemic uncertainty

A MATLAB code is developed here for the uncertainty evaluation accounting for both aleatory and epistemic uncertainties by using two nested Monte-Carlo loops. The flowchart of Figure 3 (introduced by [3] for treatment of Level-2 PSA uncertainties) is the basis for the development of the programming algorithm. In the outer loop (symbolized by dashed arrows representing epistemic loop), random draws of the model parameters θ are performed. In the inner loop (aleatory loop), characteristic parameters X are randomly sampled from their joint probability distribution function $p(X|\theta)$, which is defined by the θ values drawn in the outer loop. The outer loop is passed through n_{out} times, and each cycle of the outer loop is followed by n_{in} cycles of the inner loop.

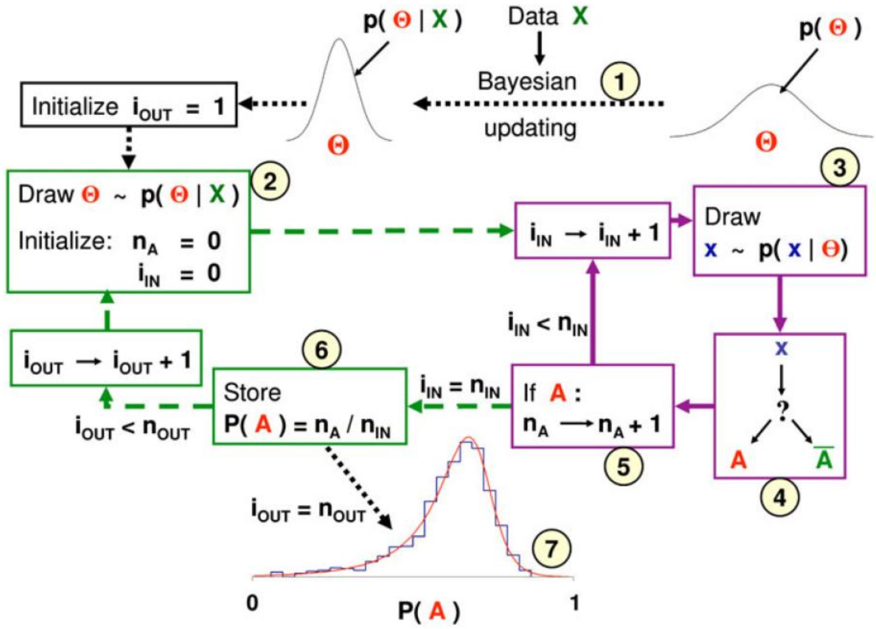

Figure 3: A generic Monte Carlo algorithm for drawing random samples [3]

For modeling of the creep rupture in major RCS components, Uncertainties of physical parameters, are represented by epistemic uncertainty (e.g. the Larson-Miller Parameters); however, uncertainties in accident progression parameters fall into the category of “aleatory uncertainty”. These parameters are obtained through deterministic analysis of the plant response in the course of accident (by e.g. MELCOR code) which is directly affected the following factors [4] as discussed previously in the Larson-Miller model:

- Temperature profile of the pipe,
- Mechanical stress on the pipe wall due to pressure difference,
- Time when risk of partial failure ends, e.g. RPV failure.

Table 1 describes steps 1 through 7 of the flowchart of Figure 3 and elaborates each step.

Table 1: Steps for implementation of the Monte Carlo algorithm

Step	Details of the step
Initialization	Selection of distributions for the epistemic parameters. Drawing a set of parameters according to the epistemic distributions.
aleatory loop	Selection of distributions for the aleatory parameters. Draw a set of parameters according to the aleatory distributions. Calculate a single value of the creep rupture time with the drawn epistemic and aleatory parameters. Draw a new set of aleatory parameters to finally end up with a distribution of the rupture time. By comparing this distribution with the time, when risk of passive rupture ends (e.g. RPV-failure) a single value for the branch probability is gained.
epistemic loop	Select another set of epistemic parameters. Within the (repeated) aleatory loop a new branch probability is calculated, as described in steps 3 to 7. Repeating these two loops several times one finally ends-up with a distribution of the branch probability that can be used for error propagation in the APET.

For implementation of the Monte-Carlo algorithm the distribution of the aleatory parameters are obtained through the thermo-hydraulics analysis by MELCOR code described in details in the following section.

IV. MELCOR modeling of SBO

MELCOR prediction of the primary system's pressure is provided in Figure 4. Initially, the pressure decreases as the heat removal from the steam generator exceeds the decay heat power. The RCS pressure drops until it reaches an equilibrium state with the secondary pressure (i.e., ~8 MPa). However, once the steam generators dry out at ~0.5 hr., the RCS inventory boils which results in RCS pressurization. According to Figure 4, the RCS pressure rises until it reaches the pressurizer relief valve set point.

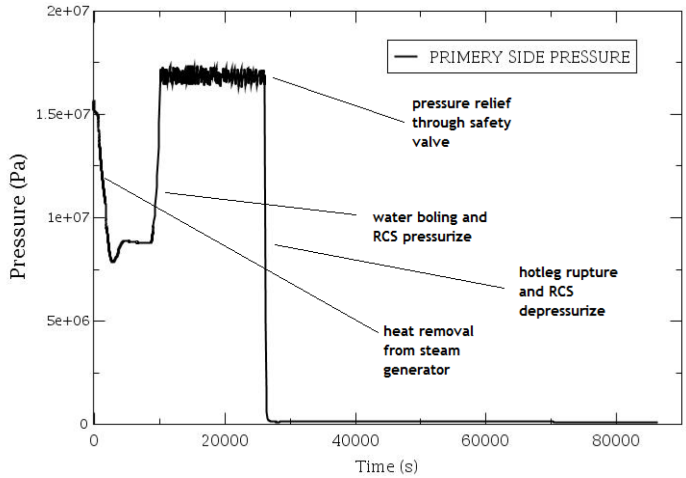

Figure 4: pressure history

Figure 5 shows the MELCOR prediction of the heat structure temperatures in the hot-leg nozzle, surge line, and SG average tube.

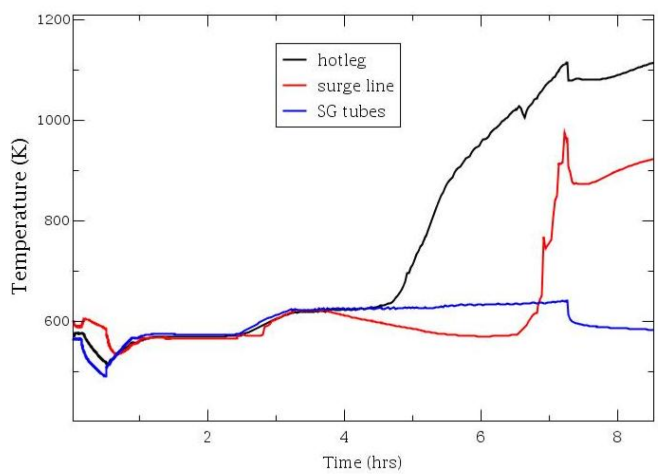

Figure 5: heat structure temperature

As seen in Figure 6, the creep rapture parameter behavior for the surge line, SG tubes, and the hot leg shows that for the case analyzed, the hot leg reaches the critical value R = 1 prior to the surge line or steam generator tubes and would fail first.

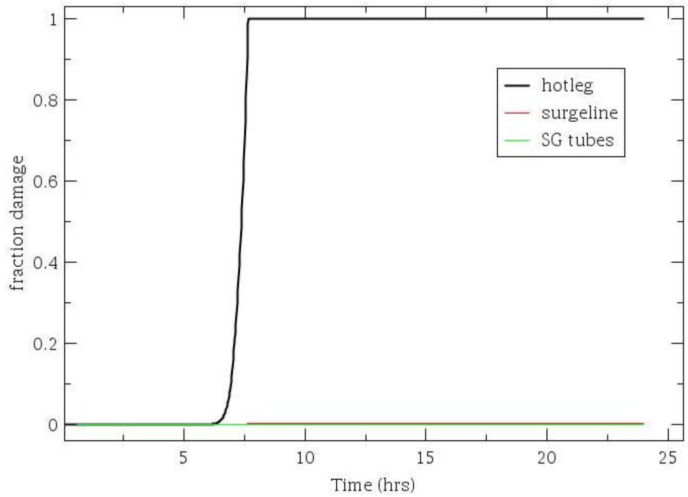

Figure 6: damage function

As shown in Figure 4, due to the combination of high pressure and temperature in the RCS, the hot leg fails due to a thermo mechanical creep rupture at 7.65 hr (Figure 6). The RCS pressure follows decreasing trend after the hot leg rupture, eventually approaching the containment pressure. The subsequent rapid depressurization leads to accumulators discharge, which refloods the core.

V. Integration of deterministic and probabilistic results

Within the Larson-Miller model (described in section 2.1) the criterion for the creep rupture of RCS components is determined by the temperature T(t) and the stress as functions of time $\sigma(t)$, by the material parameter vector β, and by the time to reactor pressure vessel failure (t_{VF}); see Figure 7.

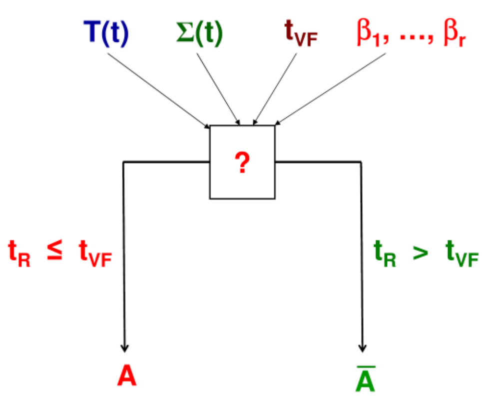

Figure 7: RCS component creep rupture parameters and criterion [3]

The criterion for hot leg creep failure is defined as the occurrence of hot leg failure before vessel breach due to melt erosion. For the steam generator tube the criterion is set to meet the following condition:

- rupture time is less than time for vessel breach $t_R \leq t_{VF}$
- rupture occurs before the time for hot leg piping failure $t_R \leq t_{HL}$

Table 2: List of uncertain input parameters & their distribution for SG tubes

Parameter	Mean	Standard Deviation	Distribution
a	-11333	0.01*mean	Normal
b	43333	0.01*mean	Normal
c	-15	0.01*mean	Normal
T (R)	[975,2340]		Uniform
σ (ksi)	[7.25,16.82]		Uniform
T_{VF} (hr)	[6.5,11.64]		Uniform

Material properties, on the other hand experience their highest probability of occurrence for their central values with a normal distribution around their mean value.

VI. Concluding Remarks

The quantification process is performed by developing a MATLAB code which pursues the algorithm discussed in previous section. The input to this code is Larson-Miller parameters along with their uncertainties. The code returns the calculated probability of creep rupture failure of the simulated RCS component. We implemented the methodology on "hot leg piping" as well as "steam generator tubes".

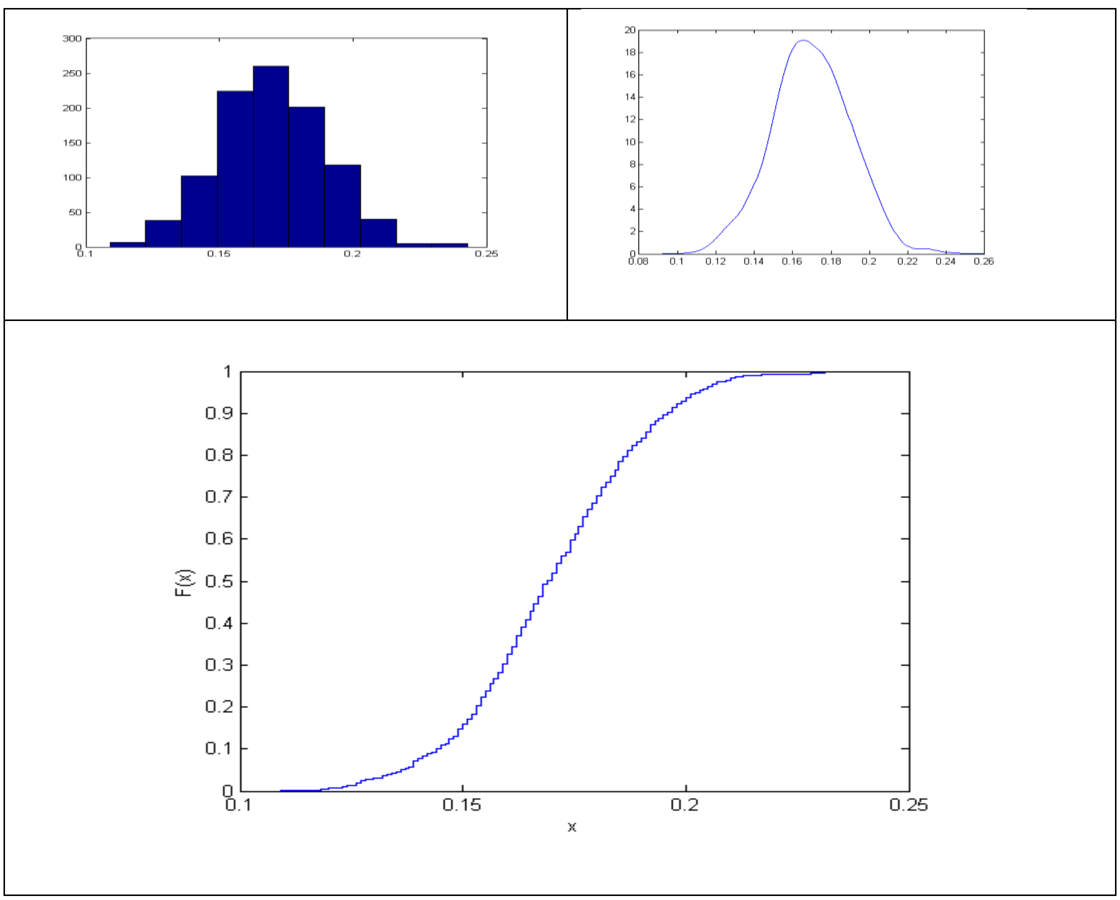

Figure 8: Monte Carlo generated uncertainty distribution for steam generator tube failure

Now sufficient information is available to calculate the probability of creep rupture and its distribution from the deterministic calculation. A MC-calculation with 1000 iterations in the outer loop and 1000 iterations in the inner loop is performed, for a total of 106 iteration steps, creating a sufficiently large sample size to approximate the PDF. The histogram, PDF resulting from the 1000 point values and its cumulative probability are shown for steam generator tubes in Figure 8. The horizontal axis in these figures represent the probability of failure of the component meeting the criterion mentioned earlier, while the vertical axis stands for their frequencies.

As a final note, the conditional probabilities of the creep rupture failure on the SG tubes and the hot-leg were calculated as 0.17 and 0.75, respectively. It is needed to incorporate this uncertainty into the event tree model to be able to quantify the confidence interval for the final L2PSA results.

REFERENCES

[1] M. Pourgol-Mohammad and SM Hosseini, "Comprehensive Uncertainty Assessment Methodology for Probabilistic Risk Assessment (PRA)," in *Proceedings of PSA2011 Conference*, Wilmington, 2011.

[2] Eduard Hofer, "When to separate uncertainties and when not to separate," *Reliability Engineering and System Safety*, vol. 54, pp. 113-118, 1996.

[3] A. Hoefer, "Uncertainty Treatment for Level-2 Probabilistic Safety Analysis," *Nuclear Science and Engineering*, vol. 166, pp. 202–217, 2010.

[4] OECD, "best-practices guidelines for level 2 PSA development and application," IRSN-PSN/RES/SAG 2013-0177 2013.