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Application of a Level 2 PSA methodology in the event of loss of coolant circulation accident in the spent fuel pool of a vSMR

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1. Introduction

The Probabilistic Safety Assessment (PSA) is a method of quantitative safety and risk assessment. In nuclear industry, the PSA are performed for three different levels. The Level 2 PSA addresses the phenomenological and physical events that can occur during and after core melt. The end states of the event tree provide significant insights on accident prevention and mitigation, pointing to measures with great potential to improve the design and operation of Small Modular Reactors (SMR).

SMR are smaller than conventional reactors and are designed to produce electricity of up to 300 MW(e).. Very small modular reactors (vSMR) (about 10 to 50 MWe) are a type of SMR. This paper describes a model under development with to quantify the risk during a loss of coolant circulation accident in the spent fuel pool (SFP) of a generic pressurized water reactor (PWR) - type vSMR with a 10 MW(e) in Low Power and Shutdown (LPS) operating mode, as part of a Level 2 PSA.

2. Methodology

The methodology applied in the development of PSA Level 2, presented in detail in [1] is based on the recommendations of the International Atomic Energy Agency (IAEA) [2]. The Table I summarizes the main steps.

Table I: Main steps of the PSA Level 2 methodology [1]

Steps	Description
1	Selection of initiating events (IE) to be considered in the analysis;
2	Grouping of possible accidents (Definition of plant damage states, PDS)
3	Accident progression analysis and probability of event trees creation
4	Release category (RC) identification
5	Source Terms (ST) analyses
6	Calculation of the Large Early Release Frequency (LERF).

The reference vSMR is a two-loop pressurized water reactor (PWR) with an electrical capacity of 10

MWe, the core consist of 21 fuel elements with uranium oxide (UO2) fuel rods enriched at about 5%. The SFP will have a storage capacity of 378 fuel elements of the 17 × 17 fuel type, enough for thirty years of operation. The description and the initial steady-state conditions is presented in previous papers [3] [4].

Plant configuration changes significantly during the refueling process, these changes are due to fuel movement, changes in water level, maintenance and other activities. The Phase III of the Low Power and Shutdown Mode of Operation is 20 days long with one reactor core located in the SFP. The plant damage states (PDS) “Loss of Coolant Circulation” in the SFP (Fase III) appears in second highest frequency (the highest contribution to the CDF of the plant, Table II) and correspond to 11.44% of the contribution of the total CDF.

Table II - CDF in the Level 1 PSA for the reference NPP [1][5]

Operational Mode	Initiating Events and PDS		CDF (/yr)	Percentage of Total CDF (%)
Full Power	Internal Events		7.62E-06	3.43%
	External Events		1.69E-04	76.23%
Low Power and Shutdown	Internal Events	Loss of coolant circulation in the SFP (Phase II and IV)	2.54E-06	1.14%
		Loss of coolant circulation in the SFP (Phase III)	2.54E-05	11.44%
		Loss of coolant circulation in the reator core (Phase I and V)	1.44E-05	6.47%
		Loss of coolant circulation in the reator core (Phase II and IV)	2.85E-06	1.28%
Total			2.22E-04	100.00%

The next step is the analysis of the PDS progression and the creation of event trees, the PDS “Loss of Coolant Circulation” in the SFP (Fase III) was chosen for analysis.

3. Event Tree

The accident progression of the PDS involve the following frontline systems: a) the primary fuel pool cooling system (SRA), used when the fuel is present in the SFP and in the transfer channel, and; b) the heating, ventilating and air conditioning system (HVAC) and subsystems, used to maintain ambient conditions within acceptable limits of temperature and humidity, control contamination and monitor the release of air from the controlled areas and contain any radioactivity that might be released in the event of an accident. More details of the systems can be found in [6].

The progression of the accident was evaluated through event tree, Figure 1, using CAFTA [7]. The generic failure rates used are based on a review and combination of previous industry generic data and data from published plants' PSA. A detailed fault tree model was developed for each of the systems identified as necessary during the course of the accident. The quantification was performed using PraQuant software [8].

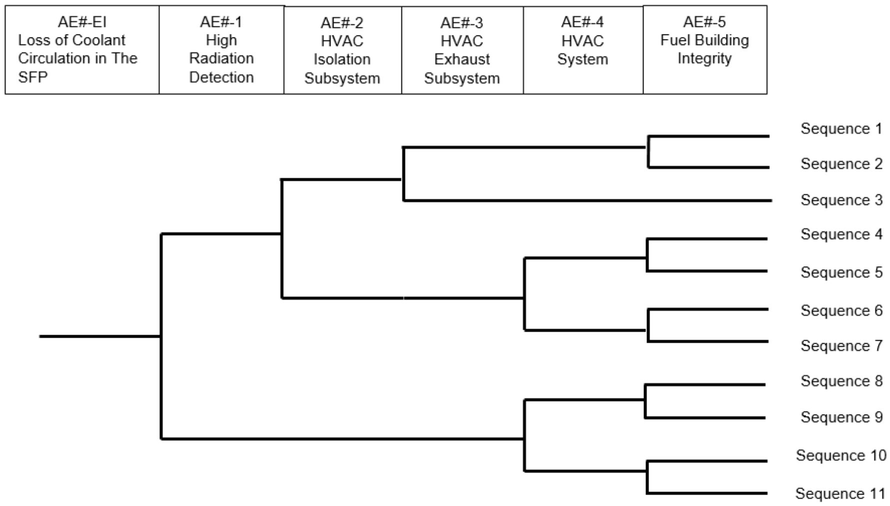

Figure 1: Event Tree

4. Deterministic Analysis

The Sequence 4, 5, 6 and 7 have a frequency of approximately E-09 with uncontrolled release to the atmosphere. To continue the analysis and define the specific radiological release of these sequences, the initial events of these sequences were simulated using the MELCOR [9]. The validation of the models applied in this paper is restricted to the validation performed by the authors of MELCOR [10]. The thermal-hydraulic diagram of the SFP nodalization is shown in Figure 2. The sequence simulated was:

1) AE#-1 High Radiation Detection – Success
2) AE#-2 HVAC Isolation Subsytem – Success
3) AE#-3 HVAC Exhaust Subsystem – Failure

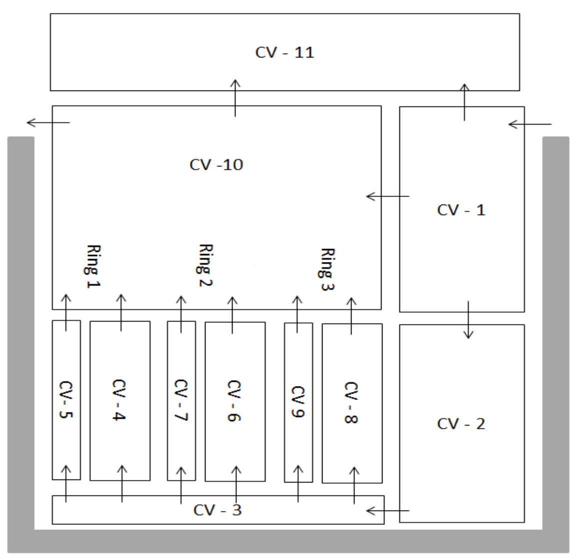

Figure 2: Thermal-hydraulic nodalization of the SFP in MELCOR [11]

The core start to be uncovered at about 83 hours after the loss of coolant circulation and the SFP runs out of liquid water at about 97.22 hours. The maximum cladding temperature reaches 1700 K after the core is uncovered, The fuel claddings are partially melted. The generation of hydrogen starts at 87 hours, the total cumulative amount of hydrogen produced by the zirconium oxidation and the steel oxidation during the progress of the accident is 10.7 kg, the mass of hydrogen in the atmosphere of the building is 8.2 kg.

5. Preliminary Results and Discussion

The results are preliminary considering the methodology is under development. With the result of the simulation and an analysis of the fuel building structure, the fault trees corresponding to the AE # -4 HVAC System and AE # -5 Fuel Building Integrity will be updated, considering the integrity of the components and hydrogen combustion, making the analysis more realistic to determine the release categories.

The Shapiro diagram [12] is used to determine whether the mixture′s composition in the building atmosphere during the accident is flammable. Figure 3 shows that the composition of the mixture during the accident is outside the flammability region due to the large concentration of vapor from the SFP.

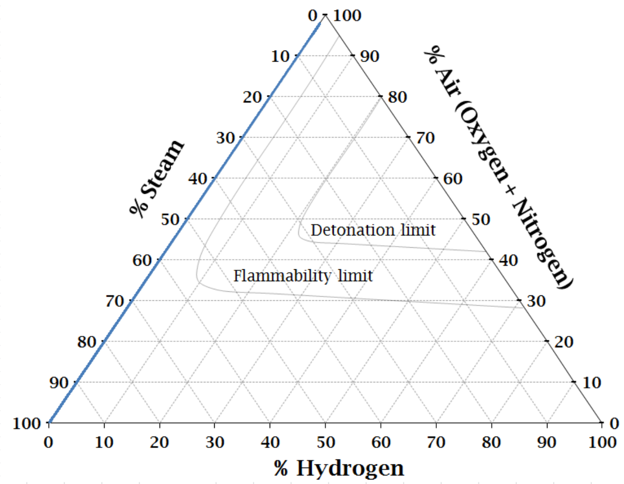

Figure 3: Hydrogen risk in containment for TLOFW.

6. Conclusions

So far, the methodology has proved to be satisfactory and the analysis is being carried out without major complications. In the future, step 3 will be concluded and steps 4 to 6 will be performed. Subsequently, with the experience acquired in this study, the others Low Power and Shutdown PDSs will also be analyzed using the same methodology.

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