

Determining LOCA Initiator Categories for a PSA of a New Reactor Design

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ABSTRACT

When developing Probabilistic Safety Assessment (PSA) for a new reactor design, initiators are often categorized on the basis of generic criteria used for the existing nuclear power plants such as large Pressurized Water Reactors (PWRs). However, when a new NPP design differs significantly from conventional large PWRs, the absence of plant-specific deterministic justification for the adopted LOCA break size thresholds (large, medium, small) introduces uncertainty in accident sequence modelling and success criteria definition as well as potential incorrect definition of the initiating events frequencies.

A deterministic, design-specific LOCA categorization should be developed with the objective of establishing break size ranges that reflect the actual response and functional requirements for the specific plant design, thereby improving the technical credibility of the PSA event tree and fault tree models.

In the following paper, a LOCA break size categorization methodology will be shown, along with dedicated best-estimate deterministic simulation models qualified for the specific NPP design in order to substantiate key assumptions and obtained results.

The methodology comprises two main steps: (1) definition of LOCA categories in terms of required system functions for accident mitigation, including short-term core cooling and long-term recirculation; and (2) performance of selected plant-specific thermal-hydraulic analyses to determine representative break size ranges and associated phenomena that can be mitigated by the same combinations of systems.

This categorization process follows internationally recognized PSA guidance and standards, including IAEA SSG-3, ASME PRA Standard, IAEA TECDOC-1804, and NRC Regulatory Guide 1.200, which emphasize grouping of initiating events based on similarity of accident progression, plant response, and success criteria of mitigating systems.

The resulting LOCA categorization, spanning from Double-Ended Guillotine Break (DEGB) to Very Small LOCA, provides a technically justified basis for identifying the systems required for mitigation of each break size range and for subsequent defining of consistent success criteria within the PSA framework. This methodology aims at enhancing the reliability of accident sequence analysis and supports risk-informed evaluation of plant safety.

Keywords: PSA, LOCA initiator categories, system success criteria, event tree modeling

1 INTRODUCTION

Categorization of LOCA initiators in PSA for traditional Nuclear Power Plants (NPPs) is frequently based on generic rules and conventions. For example, lower end for the large LOCA was in many PWR NPP PSAs set to 6" and lower end for the medium LOCA to 2" (e.g., [1]). As percentages of the break flow area relative to that of the main pipe, these correspond respectively, considering a typical PWR Cold Leg diameter, to about 4% and 0.5%. These can be considered typical ranges for large and medium LOCAs in large PWR PSAs.

The above mentioned approach to LOCA categorization is acceptable at the stage of design at which a preliminary PSA can be performed. It enables the first, indicative, assessment of risk.

On the other hand, for new NPP design, the primary system as well as the Emergency Core Cooling System (ECCS) can be different (with regard to scale and features / functions) than in the large PWRs. Design bases and design parameters are different, components and pipes are much smaller, etc.

Identification of systems relevant for mitigation of particular LOCA category, for the purpose of PSA event tree and fault tree modeling, needs to be based on a design-specific analysis, so that the event trees would reflect appropriate logic of accident sequences and system success criteria.

The approach to LOCA categorization presented in this paper relies on comprehensive set of supporting thermal-hydraulic analyses, in order to identify the LOCA break ranges which require or can be bounded by the same combinations of plant systems to perform safety injection and recirculation of the coolant following a LOCA initiator.

To establish generic bases for categorization of LOCA categories in a new NPP PSA, provided below are main points extracted from four internationally well-known PSA guides and normative documents: IAEA SSG-3, [2], ASME PRA Standard, [3], IAEA TECDOC-1804, [4], and NRC's Regulatory Guide 1.200, [5].

IAEA SSG-3 [2], in its paragraph 5.31, states that the set of LOCAs identified should be categorized and grouped in accordance with the success criteria of the Systems, Structures and Components (SSCs) that need to be operated to prevent core damage. For traditional PWRs, LOCAs are usually categorized as large, medium or small, mainly on the basis of the response needed from the coolant injection systems to mitigate LOCA. Depending on the plant design, as stated, a different set of equipment may be needed to provide protection from very small loss of coolant accidents.

In paragraph 5.34 (dealing with subject of grouping of initiating events), it is stated that initiating events should be arranged in groups in which all of the following properties of the initiating events are the same (or very similar):

- The accident progression following the initiating event;
- The success criteria for the credited systems;
- The effect of the initiating event on the availability and operation of credited systems, including the presence of conditions for signals that will actuate protection actions or block actuation of systems;
- The response expected from operating personnel.

In paragraph 5.35, it is stated that the success criteria for the credited systems used for a specific group of initiating events should be the most stringent criteria for all the individual events within the group.

In its "supporting requirement" IE-A2, the ASME PRA Standard [3] states that LOCA initiators should be differentiated using a defined rationale and that examples of LOCA types include:

- Small LOCAs. Examples: reactor coolant pump seal LOCAs, small pipe breaks;
- Medium LOCAs. Examples: stuck open safety or relief valves;
- Large LOCAs. Examples: inadvertent ADS, component ruptures;

- Excessive LOCAs (LOCAs that cannot be mitigated by any combination of engineered systems). Example: reactor pressure vessel rupture;
- LOCAs Outside Containment (category relevant for BWRs, in principle).

In supporting requirement IE-B2, the Standard requires that a structured, systematic process for grouping of initiating events is used. This is followed, under IB-3, by the requirement that initiating events are grouped only when the following can be ensured:

- Events can be considered similar in terms of plant response, success criteria, timing, and the effect on the operability and performance of operators and relevant mitigating systems; or
- Events can be subsumed into a group and bounded by the worst case impacts within the group.

However, with regard to the second bullet, there is a caution not to subsume initiators into a group unless:

- The impacts are comparable to or less than those of the remaining events in that group; and
- It is demonstrated that such grouping does not impact significant accident sequences.

In the supporting requirement IE-B4, it is clarified that grouped separately from other initiating event categories should be those categories with different plant response (i.e., those with different success rate criteria) impacts or those that could have more severe radionuclide release potential (e.g., Large Early Release Frequency (LERF)). This includes such initiators as excessive LOCA, interfacing systems LOCA, steam generator tube ruptures, and un-isolated breaks outside containment.

IAEA-TECDOC-1804 [4] describes the attributes which a Level 1 PSA should have in order to achieve technical adequacy. It was developed with the purpose to provide structured methodological assistance for Level 1 PSA quality reviews. It can be considered that the idea behind its development was to generalize the above-mentioned ASME PRA Standard in a way that it may be applied to the reactor types other than Light Water Reactors (LWRs), for use in IAEA member states worldwide.

Under the “general attribute” IE-C01, it is stated that a structured, systematic process for grouping of the initiating events is used. Under the associated rationale, it is clarified that grouping can be performed only based on similarity of the accident progression and success criteria of all the events included in the group.

General attribute IE-C02 is defined as: “Grouping of initiating events is justified”. Under the associated comments, it is recommended that Initiating events are grouped in a single group only when the following can be assured:

- Events have the same safe and unsafe end states and lead to a similar accident progression in terms of the plant response, success criteria, timing, and the effect on the operability of relevant mitigating systems and operators’ performance; or
- Events can be subsumed into a group and bounded by the worst case impacts within the group.

It is further clarified that those initiating events that have significantly different environmental impact or could have a more severe radionuclide release potential are grouped separately from other initiating event categories.

The NRC Regulatory Guide 1.200 [6] basically endorses, with certain exceptions and clarifications, the use of the aforementioned ASME PRA Standard as one possible approach to demonstrate conformance with regulatory positions concerning the acceptability of a PSA for risk-informed applications.

With regard to the above mentioned requirements from the Standard, no objections are given. In section C.1.2.1, it is stated that, with regard to initiating event analysis, an understanding of the nature of the events is performed such that a grouping of the events, with the groups defined by similarity of system and plant responses (based on the success criteria), may be performed to manage the large number of potential events that can challenge the plant.

2 METHODOLOGY FOR LOCA CATEGORIZATION

Given the wide spectrum of possible break sizes and locations, a structured categorization methodology is required to systematically evaluate plant response, identify the necessary mitigation systems, and define consistent success criteria.

The present methodology establishes a stepwise approach to categorize LOCA scenarios based on thermal-hydraulic behaviour and safety system actions, ensuring consistency between deterministic safety analysis and supporting evaluations. This approach combines system-level functional analysis with progressively refined thermal-hydraulic simulations to define bounding conditions, success sequences, and category thresholds. The methodology comprises two main steps: (1) definition of LOCA categories in terms of required system functions for accident mitigation, including short-term core cooling and long-term recirculation; and (2) performance of selected plant-specific thermal-hydraulic analyses to determine representative break size ranges and associated phenomena that can be mitigated by the same combinations of systems. The former step is composed of a number of tasks, which in the following have been divided into further steps for better clarity, leading to a total of six distinct steps:

1. Identification of Mitigation Systems;
2. Preliminary Mapping of Mitigation Systems;
3. Definition of Core Damage Criterion and Identification of Preliminary Success Sequences;
4. Identification of Minimum Requirements for a Bounding Scenario;
5. Reformulation of the Success Sequence for each LOCA Category;
6. Thermal-Hydraulic Analyses with Decreasing Break Size.

The above steps are presented in more details in the following subsections.

2.1 Identification of Mitigation Systems

The first step consists of identifying all systems that contribute to the mitigation of a LOCA. These include both active and passive safety systems credited in accident conditions. Typical systems considered are:

- Emergency Core Cooling System (ECCS), including High-Pressure and Low-Pressure Injection Systems (HPIS/LPIS);
- Residual Heat Removal (RHR) system;
- Accumulators;
- Emergency Feedwater (EFW) system;
- Reactor Make-Up System.

For each system, the functional role in core cooling and inventory control and the actuation criteria and timing should be known.

2.2 Preliminary Mapping of Mitigation Systems into LOCA Categories

A functional mapping is developed to associate mitigation systems to specific LOCA scenarios and phases of the accident. This mapping provides:

- Association between break size/location and system action;
- Identification of overlapping or redundant safety functions (if present);
- Pressure conditions at which each system is expected to operate.

The mapping is typically structured as a matrix correlating LOCA categories and available mitigation systems

This step helps the identification of systems that are essential for preventing core damage under different conditions.

2.3 Definition of Core Damage Criteria

A clear and measurable criterion for core damage is defined to assess the success or failure of mitigation strategies. This criterion is generally based on thermal limits, such as:

- Maximum cladding temperature (e.g., about 1473 K for zirconium alloys);
- Onset of significant fuel degradation or oxidation;
- Core uncover duration exceeding acceptable limits.

The selected criterion must be consistent with regulatory requirements and safety analysis practices.

2.4 Identification of Minimum Requirements for a Bounding Scenario

Thermal-hydraulic analyses are initially performed for bounding scenarios, typically corresponding to the most severe break size and most penalizing location within each LOCA category. In this context, the selection of available mitigation systems is carried out conservatively; for example, only a single train of each injection system may be assumed to be available, rather than the full redundancy foreseen in the design. These systems' combinations are defined here as Minimum Requirements. This approach ensures that the capability of the plant to cope with the examined accidental scenario is demonstrated under limiting conditions, relying on the minimum set of mitigation systems necessary to prevent core damage.

These bounding analyses therefore represent a baseline hypothesis for accident mitigation and provide a conservative reference for following analyses. In addition, they allow for the evaluation of the thermal-hydraulic response of the primary system, including the evolution of pressure and temperature, phase distribution, and core cooling conditions. Additionally, these bounding analyses allow the analyst to identify the characteristic actuation timing of the mitigating systems under limiting conditions and to assess whether revisions to the system design or actuation logic may be required. This may include, for instance, adjustments to setpoints, delays, or system capacities in order to ensure timely and effective response during the most demanding accident scenarios.

2.5 Formulation of the Success Sequences for Each LOCA Category

Based on the results of the initial TH analyses, the success sequences for each LOCA category are developed. Success sequences should include:

- Identification of systems (or combinations of systems) required for the mitigation of the accident scenario;
- Identification of the setpoints for each mitigation system;
- Identification of the actuation sequence of the mitigation systems;
- Identification of alternative mitigation paths.

This step ensures that the success criteria are realistic and consistent with system performance.

2.6 Thermal-Hydraulic Analyses with Decreasing Break Size

A systematic series of TH simulations is performed with progressively decreasing break sizes. The objective is to identify the transition points between LOCA categories. As an example, starting from DEGB-LOCA, at a certain point the available safety system will no longer be sufficient to mitigate the accident scenario and will lead to the identification of the minimum break size threshold for core damage for that category. At that stage, the LOCA category will change (e.g. IB-LOCA), and new safety systems will be introduced to mitigate this subsequent category.

This iterative process allows for a refinement of categorization thresholds giving also an improved understanding of mitigation systems' operation and effectiveness across all the LOCA spectrum.

3 RESULTS FOR A SMALL PWR DESIGN APPLICATION

Following the methodology described in the previous section, here is presented its application to a new small PWR design. The objective is to demonstrate how the proposed LOCA categorization approach can be systematically implemented in practice, highlighting the transition from the methodological framework to concrete results.

According to the considered reactor design, systems relevant for mitigation of LOCA initiators, considering the overall range of breaks, are identified as follows:

- High Pressure Injection System (HPIS);
- Low Pressure Injection System (LPIS);
- Accumulators;
- Residual Heat Removal System (RHR);
- Emergency Feedwater System (EFW);
- Make-Up System.

A preliminary mapping of this mitigation systems into a certain LOCA category is presented in Table 3.1.

Table 3.1: Mitigation Systems for each LOCA Break Size.

LOCA Category	Mitigation Systems
LB-LOCA	LPIS ACC RHR
IB-LOCA	HPIS (and LPIS if needed) ACC RHR
SB-LOCA	HPIS ACC EFW RHR
VS-LOCA	EFW Make-Up System

Keeping in mind that the definition of core damage is intended to represent significant fuel relocation due to fracture or melting providing a conservative estimate for this condition, the following criteria has been selected:

- A temperature in excess of 2200 degrees Fahrenheit (about 1478 Kelvin) is reached in any node of the core as defined by a best-estimate thermal-hydraulic calculation.

This condition is based on the ASME PRA Standard requirement SC-A2, [3].

The Double-Ended Guillotine Break (DEGB) LOCA in the Cold Leg (CL) has been identified as a bounding scenario. Even if this consideration appears reasonably conservative, a series of deterministic analyses for different break locations has been conducted in order to substantiate this assumption. After this, a simulation considering the availability of only one train for each injecting systems (LPIS and Accumulator) has been conducted. This is made in order to identify Minimum Requirements, i.e., minimum systems capable of mitigating this bounding accident scenario. This demonstrate that, in DEGB-LOCA, a single injecting train has been sufficient to prevent overcoming the core damage threshold and assuring the reach of Long Term Cooling (LTC) conditions through RHR recirculation.

Considering the pressure setpoints for actuation of the previously mentioned systems, success sequence for each LOCA category has been developed as shown in Table 3.2.

Table 3.2: Success Sequences.

LOCA Category	Success Sequence
LB-LOCA	Reactor Trip
	Start of successful injection from Accumulator;
	Start of successful injection from LPIS;
	at high containment sump level (or low ECCS tank level) start of successful recirculation from the sump through RHR.
IB-LOCA	Reactor Trip
	Start of successful injection from HPIS;
	Start of successful injection from Accumulators;
	Start of successful injection from LPIS;
	<ul style="list-style-type: none"> The need for LPIS should be investigated through deterministic analyses: range of IB LOCA may need to be subdivided into a range with LPIS needed and a range in which it is not needed.
at high containment sump level (or low HPIS/LPIS tank level) start of successful recirculation from the Sump through RHR.	
SB-LOCA	Reactor Trip
	Start of successful injection from HPIS;
	successful start and operation of EFW with steam relief path;
	Start of successful injection from Accumulators;
	at high containment sump level (or low HPIS/LPIS tank level) start of successful recirculation from the Sump through RHR.
VS-LOCA	Reactor Trip
	Make-Up System activation. <ul style="list-style-type: none"> Upper bound for VSLOCA is set in correspondence of a loss of coolant that can be compensated by the normal capacity of the Make-Up system. Lower bound is then established as the largest break which would still not cause an automatic reactor trip (leakage).

In the following, simulations results are shown for different break size, considering three figures of merit:

- Cladding temperature of the highest power fuel assembly, at three different relevant heights, normalized respect the critical damage temperature (1473 K): to show if cladding temperatures have overcome the core damage threshold before the depletion of the ECCS tank and of the available accumulator;
- Normalized residual mass of the tank feeding HPIS/LPIS: tank residual mass lower than 4% of the total is sufficient to allow for RHR recirculation from the sump, and so to reach LTC;
- Normalized accumulator water level: as a conservative assumption the depletion of one accumulator has been considered a necessary condition to reach LTC.

For the sake of brevity, only the results obtained for break transition from LB-LOCA to IB-LOCA are shown. In each described cases, break size is indicated as a percentage of the total flow area of the pipe in which break is happening.

As shown in Figure 3.1, cladding temperatures in the highest power fuel assembly reach the 1473 K threshold defined for the core damage in the case of 1% break. Conversely, for the 2%

break, only an initial dryout peak is observed during the early phase of system depressurization. Then the action of the accumulator allows for an effective quenching of the cladding leading to a rapid reduction in temperature. The LPIS then maintains the cladding temperatures at relatively low value for the rest of the transient.

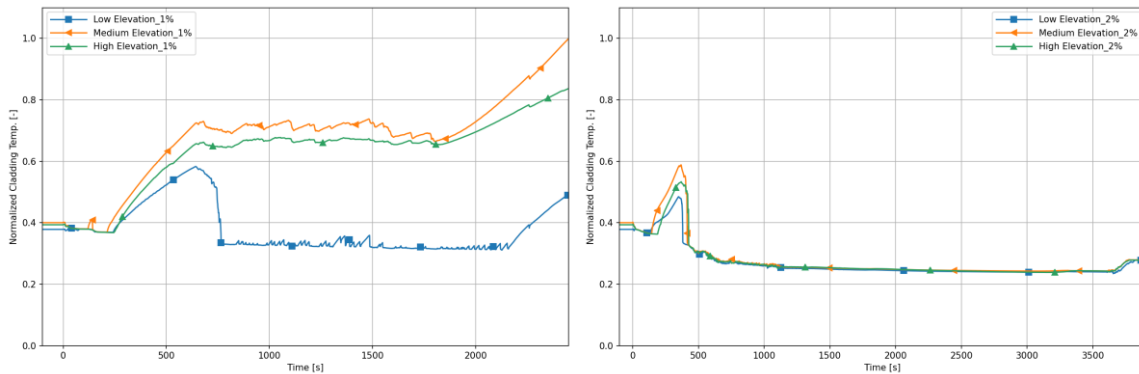


Figure 3.1: Cladding Temperatures for 1% (left) and 2% (right) Break Sizes.

Core damage temperature is reached in the 1% break case due to the slower depressurization associated with the smaller break size. Under these conditions, the system pressure remains sufficiently high to delay and reduce the effectiveness of LPIS injection. As a consequence, the coolant flow provided is insufficient to prevent the progressive temperature escalation that ultimately leads to the critical threshold being exceeded. This behaviour is further supported by the residual coolant mass observed in the ECCS tank (Figure 3.2).

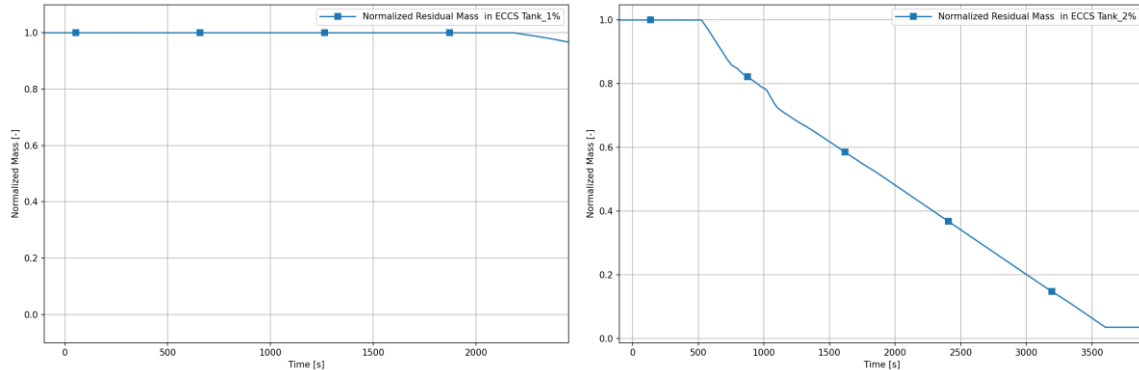


Figure 3.2: Residual Mass in the ECCS Tank for 1% (left) and 2% (right) Break Sizes.

Additionally, Figure 3.3 shows that the accumulator inventory decreases more gradually in the reduced break size case. This trend highlights not only the delayed initiation of injection but also the extended duration over which the accumulator contributes to core cooling, ultimately identifying the time at which the injection process is completed.

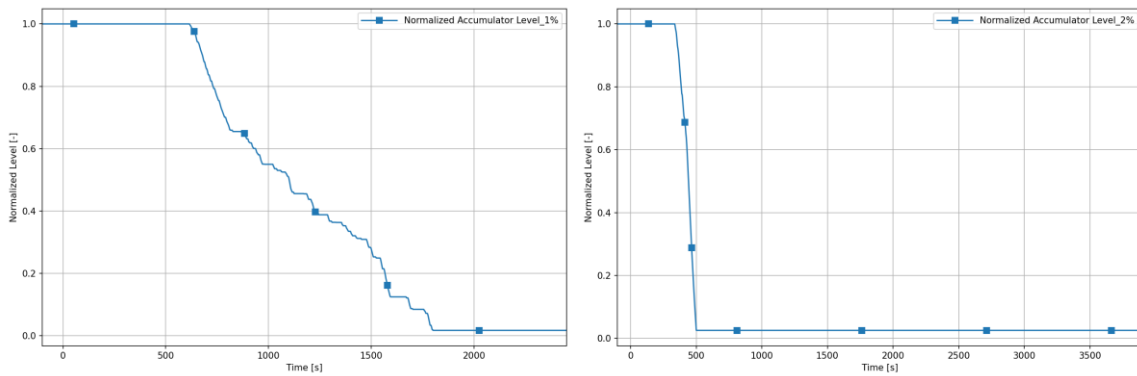


Figure 3.3: Water Level in the Accumulator for 1% (left) and 2% (right) Break Sizes.

For the 0.5% break scenario, the cladding temperature of the highest-power fuel assembly reaches the 1473 K threshold significantly later—by more than 8000 seconds—compared to the 1% break case (Figure 3.4). In this configuration, the HPIS is able to operate, and its performance is effective for these rates of depressurization.

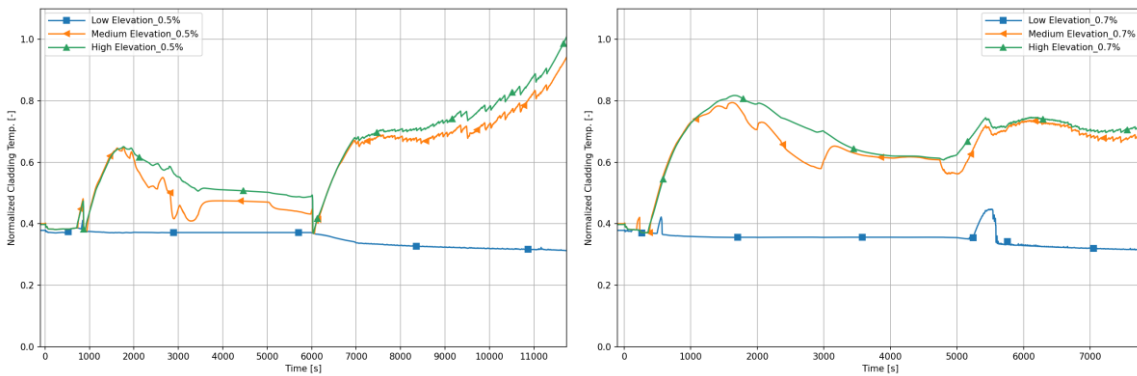


Figure 3.4: Cladding Temperatures for 0.5% (left) and 0.7% (right) Break Sizes.

This is confirmed by Figure 3.5, which shows that, for both the 0.5% and 0.7% break cases, the ECCS tank inventory is fully depleted, indicating sustained and successful coolant injection by the system.

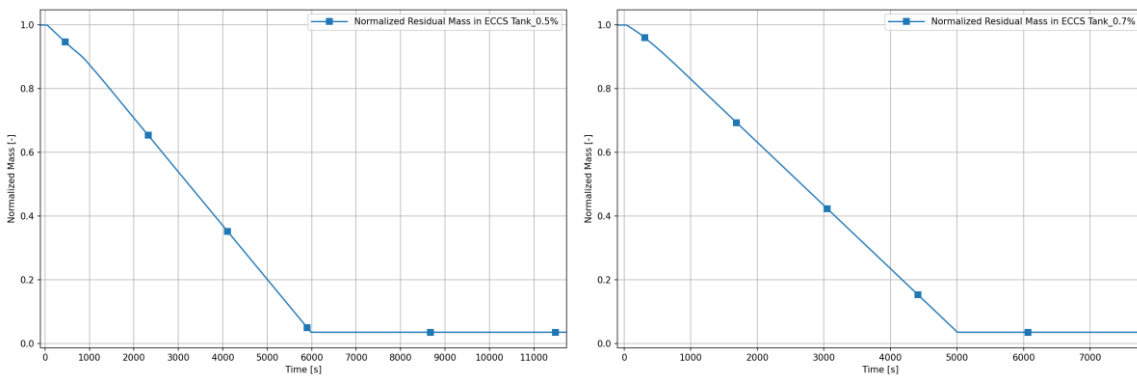


Figure 3.5: Residual Mass in the ECCS Tank for 0.5% (left) and 0.7% (right) Break Sizes.

Despite the proper functioning of the HPIS, core damage is ultimately not prevented in the 0.5% break case. The key limiting factor is the slow injection by the accumulator. Due to the break size, system depressurization proceeds slowly, as does the decrease of the accumulator level (Figure 3.6). As a result, accumulator inject slowly the coolant, without preventing an increase in cladding

temperature. This delay in the activation of a high-flow, passive injection mechanism critically reduces the overall effectiveness of the emergency core cooling strategy, ultimately leading to the attainment of the core damage temperature threshold.

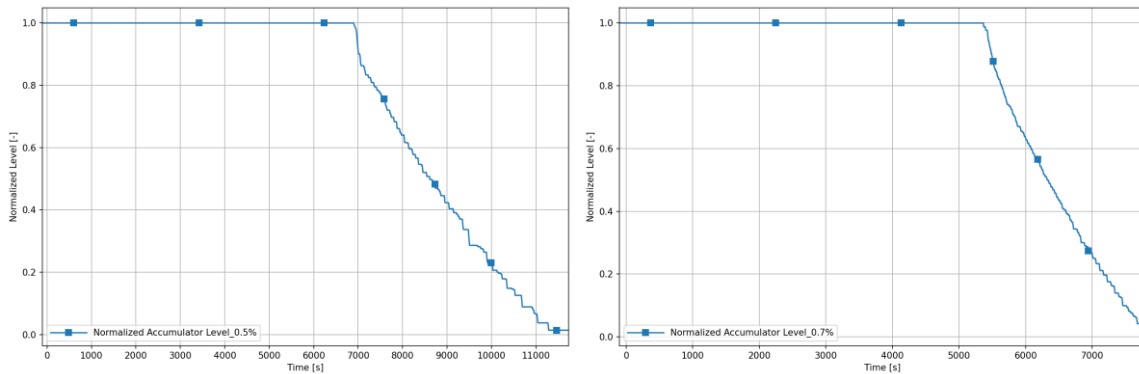


Figure 3.6: Water Level in the Accumulator for 0.5% (left) and 0.7% (right) Break Sizes.

The simulations therefore led to the conclusion that the break sizes corresponding to the transition between LB and IB, and between IB and SB, are equal to 1% and 0.5%, respectively, of the total flow area of the pipe in which the break occurs.

Further deterministic analyses allow the determination of the remaining threshold break sizes, in compliance with the identified success sequences.

4 CONCLUSION

The work presented in this paper shows how the adoption of a design-specific approach to LOCA categorization with support of deterministic analyses can overcome the limitations associated with the direct application of generic break size thresholds derived from conventional large PWR that, in some cases, wouldn't be applicable to smaller size NPP.

By linking LOCA categories to required safety functions and supporting this classification with best-estimate thermal-hydraulic analyses, obtained results can give a realistic identification of accident progression, system performance, and success criteria.

The application of the methodology confirms that plant-specific phenomena, system interactions, and design features can lead to substantially different break size thresholds compared to traditional classifications. Consequently, the derived LOCA categories provide a more consistent and technically justified basis for defining initiating events, modelling event trees and fault trees, and assigning success criteria within the PSA framework.

Furthermore, alignment with internationally recognized standards and guidance ensures that the proposed approach remains compatible with established regulatory expectations while improving the credibility and traceability of modelling assumptions.

5 REFERENCES

- [1] Poloski, J.P. et al., Rates of Initiating Events at U.S. Nuclear Power Plants: 1987 – 1995, NUREG/CR-5750, Idaho National Engineering and Environmental Laboratory, Prepared for U.S. Nuclear Regulatory Commission, February 1999.

- [2] IAEA Safety Standards, No. SSG-3, “Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants”, Revision 1, International Atomic Energy Agency, Vienna, 2024.
- [3] ANS/ASME RA-S-1.1-2024, “Standard for Level 1 / Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications”, The American Society of Mechanical Engineers, 2024
- [4] IAEA TECDOC Series, IAEA-TECDOC-1804, “Attributes of Full Scope Level 1 Probabilistic Safety Assessment (PSA) for Applications in Nuclear Power Plants”, International Atomic Energy Agency, Vienna, 2016.
- [5] NRC Regulatory guide 1.200 Revision 3, “Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities”, U.S. Nuclear Regulatory Commission, 2020
- [6] Regulatory Guide 1.200, Revision 3, Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities, U.S. Nuclear Regulatory Commission, 2020.