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THE COFUN DEVELOPMENT FOR EVALUATING CONTAINMENT FAILURE FREQUENCY USING STRESS-STRENGTH ANALYSIS

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Abstract. When applied stress exceeds strength, containment failure may occur. The probability of leakage or rupture equals the probability that stress exceeds strength. Based on this stress-strength analysis, the COFUN code was developed to quantify ECF and LCF in the CET. The containment event tree (CET) models the progression of possible severe accident scenarios in the containment building. It is detailed to reflect various accident phenomena, progression, and damage. Each CET heading has a corresponding decomposition event tree (DET), which branches the PDS LD according to logical rules, with calculated branching probabilities. Each DET addresses key phenomena affecting these probabilities. Variables with probabilistic or discrete distributions are included due to phenomenological uncertainty. Containment failure frequencies for both ECF and LCF are evaluated using this analysis. Sensitivity analysis identifies variables impacting integrity, and statistical propagation with sampling is used to quantify uncertainty.

1 INTRODUCTION

The Level 1 Probabilistic Risk Assessment (PRA) evaluates the likelihood of core damage resulting from initiating events such as station blackout (SBO) or loss of coolant accidents (LOCA). Level 2 PRA extends this analysis to assess the potential release of fission products from the containment following core damage. Containment performance analysis identifies potential accident progressions within the containment, estimates the timing and nature of possible failures, and quantifies the likelihood and source term for each scenario. These insights are used to evaluate the effectiveness of plant design and containment systems in mitigating severe accident consequences, thereby supporting safety improvements.

For a reference plant, Level 2 PRA for internal events was conducted[1-2]. This included structural reliability analysis to calculate containment failure frequencies, along with sensitivity and uncertainty analyses for key probabilistic variables.

2 LEVEL 2 PRA EVENT TREES CONSTRUCTION

This chapter describes the level 2 PRA models for the reference reactor. The quantification, sensitivity study and uncertainty analysis are performed using COFUN the code [3].

2.1 The COFUN Code

The Containment Failure Probability and Uncertainty Analysis Program (COFUN) is a computational tool designed to automatically perform Level 2 PRA for nuclear power plants. For accident progression analysis, COFUN employs the concept of a small containment phenomenological event tree called CET(containment event tree), which provides a visual framework to trace individual accident progressions. And a large supporting event tree called DET(decomposed event tre) is utilized for detailed quantification.

To facilitate integrated Level 2 PRA, the code, COFUN provides a structured workspace that allows users to organize the various event trees required for systematic modeling of accident progression [4–7]. The COFUN code supports point value analysis, sensitivity, and uncertainty analyses. In uncertainty analysis, branches in the supporting event tree are categorized as either subjective probability branches or stochastic probability branches [8–10].

A subjective probability branch yields deterministic outcomes but is based on the analyst's degree of belief due to insufficient data or knowledge. In contrast, stochastic probability branches involve entirely random variables. Given the inherent uncertainty in both types of branches, uncertainty analysis is essential.

The COFUN provides various types of distributions for probability branches in the supporting event trees and performs uncertainty analysis by sampling these specified distributions. Moreover, the program can propagate uncertainty results from Level 1 PRA to Level 2 PRA. Specifically, the uncertainty results obtained at Level 1 are assigned to the final branch of each Plant Damage State (PDS), forming the basis for subsequent Level 2 uncertainty analysis [11].

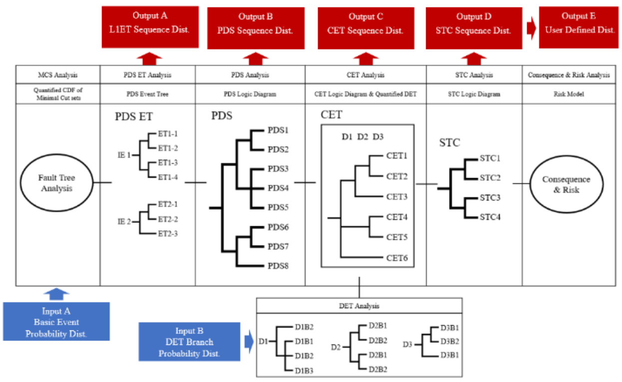

Figure 1: Overview of the level 2 PRA Process

2.2 Plant Damage State Logic Diagram

While the core damage accident sequences identified in Level 1 PRA include systems directly contributing to core damage, they do not consider systems associated with severe accident scenarios or the containment building's performance following core damage. To address this, the Level 1 PRA event tree is extended into a Plant Damage State Logic Diagram (PDS LD). This expanded model includes all systems critical to the analysis of containment behavior during severe accidents. The headings and branch elements of the PDS LD are illustrated in Figure 2. The last located WATER-IN heading accounts for the scrubbing effect in the event of a steam generator tube rupture (SGTR) accident.

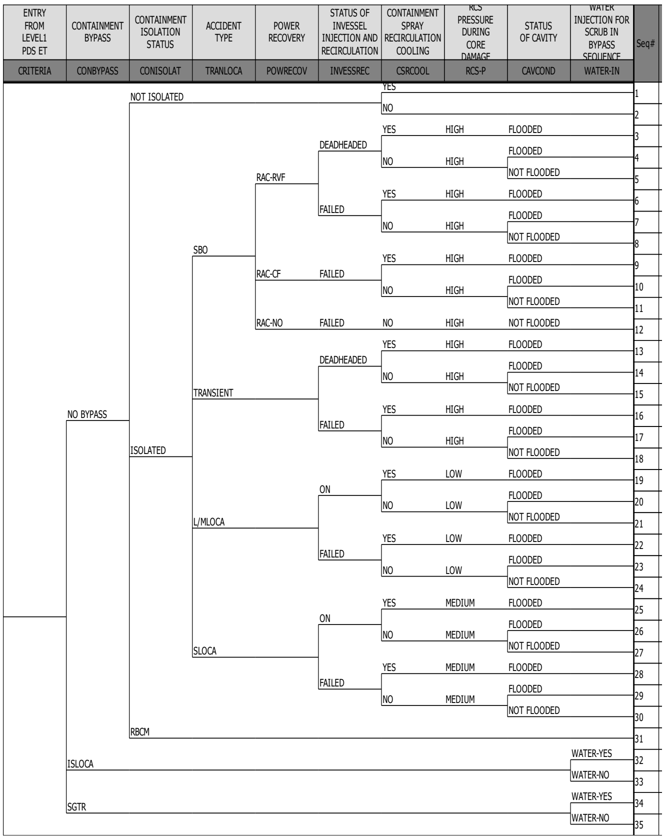

Figure 2: PDS LD of the reference reactor

2.3 Containment Event Tree

The Containment Event Tree (CET) is constructed to analyze key characteristics of the containment building during severe accident scenarios, including potential phenomena, containment status, and failure modes. Using Stress-Strength Analysis, COFUN quantifies both the Early Containment Failure (ECF) and Late Containment Failure (LCF) within the CET.

The CET must comprehensively model the progression of all plausible accident sequences within the containment structure. It should also be detailed enough to account for a wide range of severe accident phenomena, containment degradation pathways, and operator actions that influence the release of radioactive materials. The CET's headings and branches are presented in Figure 3.

For each heading in the CET, a corresponding Decomposition Event Tree (DET) is developed as a subtree to systematically categorize PDS LD based on logical criteria. The branching probabilities within the DETs related to ECF and LCF, are quantified using Stress-Strength Analysis. Each DET is designed to represent key phenomena that significantly influence the corresponding CET heading. An example of such a DET, 'RCS-FAIL', is shown in Figure 4.

3 STRESS-STRENGTH ANALYSIS

3.1 The Quantificatiuon Method

In nuclear power plants, containment failure occurs due to extreme events, such as pressure loads exceeding the containment’s yield strength. These pressures can result from slow over-pressurization or explosive phenomena like steam explosions, hydrogen burns, or direct containment heating.

Since containment failure is rare and each structure is unique, observational failure data is typically unavailable, making data collection ineffective. Therefore, structural reliability analysis relies on probabilistic models for both loads and resistance, using all available statistical information on materials and loads.

Containment should be designed so that its yield strength exceeds the pressure loads during accident progression. Both pressure loads (LOAD or DEMAND) and containment resistance (CAPACITY) must be treated as probabilistic variables due to inherent uncertainties.

Containment is considered to fail if LOAD exceeds CAPACITY—i.e., if the internal pressure surpasses the containment’s yield strength during a severe accident. Quantification involves convolution integrals, which can be complex depending on the distributions. A lognormal distribution is typically used for LOAD, and the containment failure (cf) probability can be expressed as $P_{cf} = \int_0^{\infty} \Pr(p_L = p)\left[\int_0^{p} \Pr(p_C = p')dp'\right] dp$.

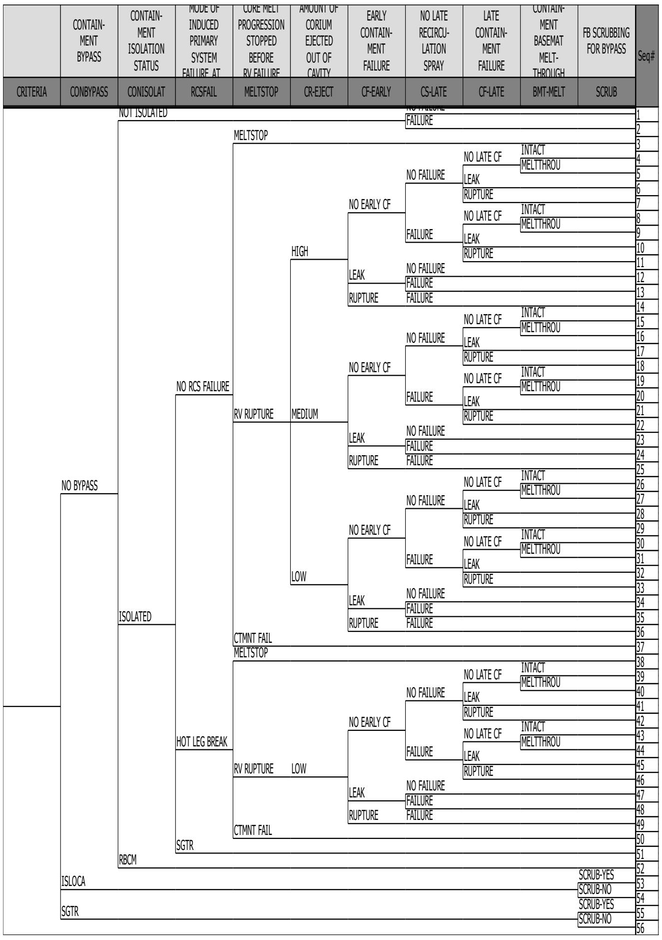

Figure 3: Containment event tree of the reference reactor

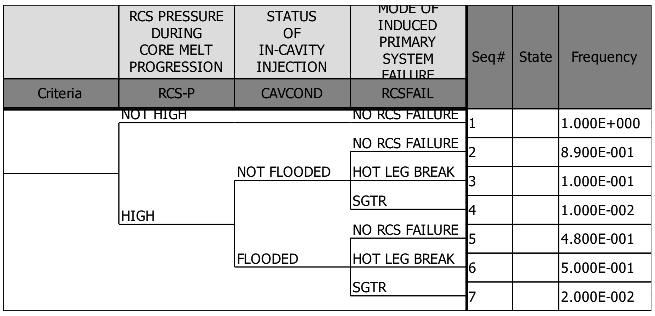

Figure 4: RCS-FAIL DET

3.2 The ECF and LCF Application

For the Early Containment Failure (ECF) DET, peak containment pressure during a Direct Containment Heating (DCH) scenario is derived using MELCOR and TCE codes—severe accident analysis tools developed by Sandia National Laboratories—and the corresponding failure probability is calculated. For the Late Containment Failure (LCF) DET, MELCOR is used to estimate peak pressure, and the containment failure probability is similarly evaluated [12].

3.3 Sourc Term Category

It is necessary to verify that the CET template accurately represents accident sequences from the initiating event to the release of fission products [13]. However, due to the similarity in release timing and quantity among many sequences, analyzing every pathway is unnecessary. Instead, CET endpoints are grouped into a limited number of Source Term Categories (STCs) based on the release characteristics of fission products. Figure 5 presents the Source Term Category Logic Diagram (STC LD).

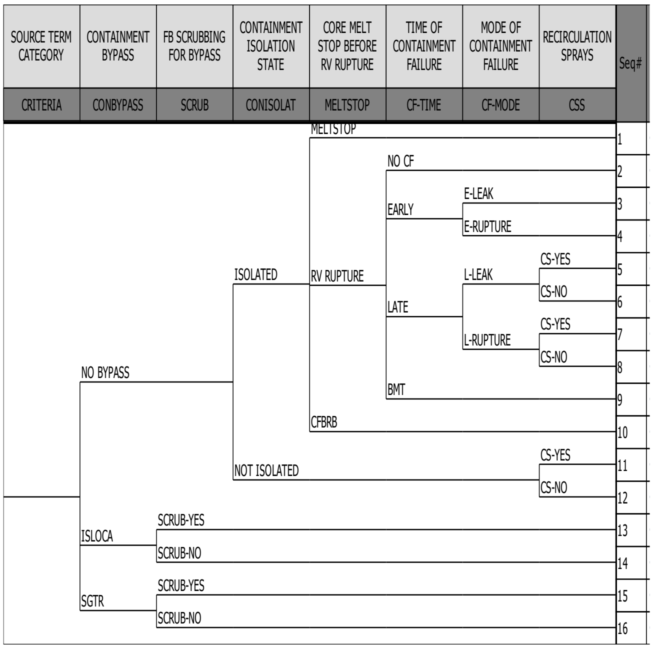

Figure 5: Source term category logic diagram of the reference reactor

4 RESULT AND DISCUSSION

4.1 Result of the Base Case

Based on the time from accident initiation to containment failure, scenarios are classified as Early Containment Failure (ECF) or Late Containment Failure (LCF). If LCF does not occur, basemat melt-through (BMT) is considered a potential outcome. Additionally, scenarios involving containment isolation failure and the leakage of radioactive materials bypassing the containment were evaluated. Seven cases are considered, including scenarios where containment integrity is maintained and where containment fails before the reactor vessel. Table 1 presents the failure probabilities and quantification results for different containment failure types. The failure frequency ratios for containment isolation failure (ISO FL), containment failure before reactor breach (CFBRB), and ECF are 0.1%, 7.3%, and under 0.1%, respectively. Contributions from BYPASS, BMT, and LCF are 11.0%, 0.3%, and under 11.3%, respectively.

The large early release frequency (LERF) is 11.1%, and Non-LERF is 18.9%.

Table 1: The results of containment failure frequency

Containment Failure Type	Frequency (/yr)	Percentage (%)
NO CF	3.42×10^{-06}	69.9
ECF	2.04×10^{-09}	<0.1
LCF	5.53×10^{-07}	11.3
BMT	1.56×10^{-08}	0.3
CFBRB	3.59×10^{-07}	7.3
ISO. FL.	2.57×10^{-09}	0.1
BYPASS	5.39×10^{-07}	11.0
Total	4.89×10^{-06}	100.0

4.2 Sensitivity Study

Uncertainties arise not only from severe accident phenomena but also from operator actions and system performance, significantly affecting the quantification of top events. Therefore, sensitivity analysis is conducted on decomposition event trees (DETs) that are expected to impact key outcomes such as the radiation source term. The effect of these uncertainties on probabilistic risk analysis results is evaluated, as shown in the results.

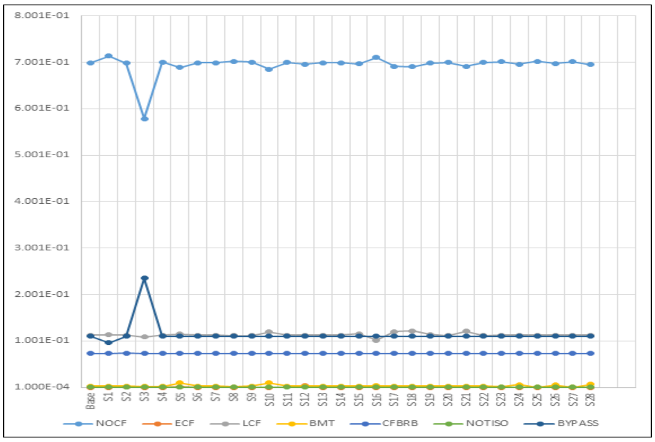

Figure 6: Result of the sensitivity study

Sensitivity analysis results indicate that Case S3 shows a notable deviation. In Cases S1, S2, and S3, input values were varied by modifying the branch probability of the RCS-FAIL DET in Figure 3. When the 'CAVCOND' branch in RCS-FAIL DET is set to NOT FLOODED, it is classified as Group A; when FLOODED, as Group B. Input values for each case are shown in Figure 6, and the analysis results are summarized in Table 3.

According to Table 3, in Case S3, the BYPASS frequency increased by 12.5%, while the NO CF frequency decreased by 12% compared to the BASE case. Table 2 confirms that in S3—where the TI-SGTR divergence probability was increased tenfold—the integrity of the containment is highly sensitive to the TI-SGTR branch probability in RCS-FAIL DET.

Table 2: Input data of the sensitivity study case for S1, S2 and S3

<table>
<tr><th rowspan="2">Branch Group</th><th rowspan="2">Branch</th><th colspan="4">Sensitivity Study Case</th></tr>
<tr><th>BASE</th><th>S1</th><th>S2</th><th>S3</th></tr>
<tr><td rowspan="3">A</td><td>NO RCS FAIL</td><td>0.89</td><td>1.0</td><td>0.29</td><td>0.8</td></tr>
<tr><td>HOT LEG BREAK</td><td>0.1</td><td>0.0</td><td>0.7</td><td>0.1</td></tr>
<tr><td>SGTR</td><td>0.01</td><td>0.0</td><td>0.01</td><td>0.1</td></tr>
<tr><td rowspan="3">B</td><td>NO RCS FAIL</td><td>0.48</td><td>1.0</td><td>0.03</td><td>0.3</td></tr>
<tr><td>HOT LEG BREAK</td><td>0.5</td><td>0.0</td><td>0.95</td><td>0.5</td></tr>
<tr><td>SGTR</td><td>0.02</td><td>0.0</td><td>0.02</td><td>0.2</td></tr>
</table>

Table 3: The result of the sensitivity study case for S1, S2 and S3

	NO CF	ECF	LCF	BMT	CFBRB	Iso. Fail	Bypass
Base	69.9	0.0	11.3	0.3	7.3	0.1	11.0
S1	71.3	0.0	11.3	0.3	7.3	0.1	9.6
S2	69.8	0.1	11.3	0.4	7.4	0.1	11.0
S3	57.9	0.0	10.9	0.3	7.3	0.1	23.5

4.3 Uncertainty Analysis

Uncertainties in containment performance analysis arise from the PDS LD and the branching probabilities used in CET quantification. PDS LD uncertainty is addressed using Minimal Cut Sets (MCS) and data uncertainties such as initiating event frequencies, equipment failure rates, and human error probabilities. During CET quantification, uncertainties exist in both classification branches and branch probabilities. Classification branch uncertainty is already reflected in PDS LD quantification, while branch probability uncertainty—particularly in DET top events—is addressed using Stress-Strength analysis. DETs for ECF and LCF are quantified accordingly, with representative probability distributions provided in Table 4. The COFUN code is used for uncertainty analysis, applying Monte Carlo sampling for uncertainty propagation. Starting from MCS, uncertainties are propagated through PDS LD, DET, CET, and STC, with results illustrated in Figures 7 and 8.

Table 4: Distribution data of the DETs

DET Identification	DET Branch	Point value	Distribution	Parameter
RCSFAIL-1:HOT LEG BREAK	HOT LEG BREAK	0.1	Lognormal	0.1|3
RCSFAIL-1:SGTR	SGTR	0.01	Lognormal	0.01|3
RCSFAIL-2:HOT LEG BREAK	HOT LEG BREAK	0.5	Lognormal	0.5|3

RCSFAIL-2:SGTR	SGTR	0.02	Lognormal	0.02|3
MELTSTOP-1:RV RUPTURE	RV RUPTURE	0.1	Lognormal	0.1|3
MELTSTOP-2:RV RUPTURE	RV RUPTURE	0.1	Lognormal	0.1|3
MELTSTOP-3:RV RUPTURE	RV RUPTURE	0.05	Lognormal	0.05|3
MELTSTOP-4:RV RUPTURE	RV RUPTURE	0.05	Lognormal	0.05|3
MELTSTOP-5:RV RUPTURE	RV RUPTURE	0.1	Lognormal	0.1|3
MELTSTOP-6:RV RUPTURE	RV RUPTURE	0.1	Lognormal	0.1|3
CR-EJECT-1:MEDIUM	MEDIUM	0.3	Lognormal	0.3|3
CR-EJECT-1:LOW	LOW	0.1	Lognormal	0.1|3
CR-EJECT-2:MEDIUM	MEDIUM	0.3	Lognormal	0.3|3
CR-EJECT-2:LOW	LOW	0.2	Lognormal	0.2|3
EVSE-1:YES	YES	0.005	Lognormal	0.005|3
EVSE-2:YES	YES	0.005	Lognormal	0.005|3
H2-MASS-1:LOW	LOW	0.5	Uniform	0|1
H2-MASS-2:LOW	LOW	0.5	Uniform	0|1
H2-MASS-3:LOW	LOW	0.5	Uniform	0|1
H2-MASS-4:LOW	LOW	0.5	Uniform	0|1
CS-DEBRIS-1:YES	YES	0.01	Lognormal	0.01|3
ERLY-BURN-1:BURN	BURN	0.5	Uniform	0|1
HMS-1:FAIL	FAIL	0.1507	Lognormal	0.1507|3
LATE-BURN-1:YES	YES	0.1	Lognormal	0.1|3
DB-DEPTH-1:SHALLOW	SHALLOW	0.1	Lognormal	0.1|3
DB-DEPTH-2:DEEP	DEEP	0.1	Lognormal	0.1|3
DB-DEPTH-3:DEEP	DEEP	0.2	Lognormal	0.2|3
EXVCOOL-1:NOT COOLED	NOT COOLED	0.1	Lognormal	0.1|3
EXVCOOL-2:NOT COOLED	NOT COOLED	0.5	Uniform	0|1
EXVCOOL-3:NOT COOLED	NOT COOLED	0.5	Uniform	0|1
BMT-MELT-1:MELTTHROU	MELTTHROU	0.05	Lognormal	0.05|3
BMT-MELT-2:MELTTHROU	MELTTHROU	0.05	Lognormal	0.05|3
BMT-MELT-3:MELTTHROU	MELTTHROU	0.1	Lognormal	0.1|3
BMT-MELT-4:MELTTHROU	MELTTHROU	0.05	Lognormal	0.05|3

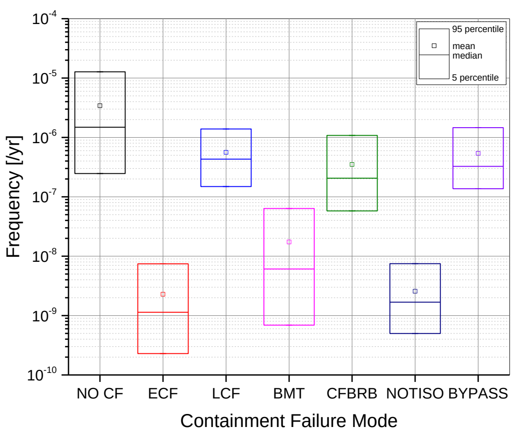

Figure 7: The result of uncertainty analysis for containment failure mode

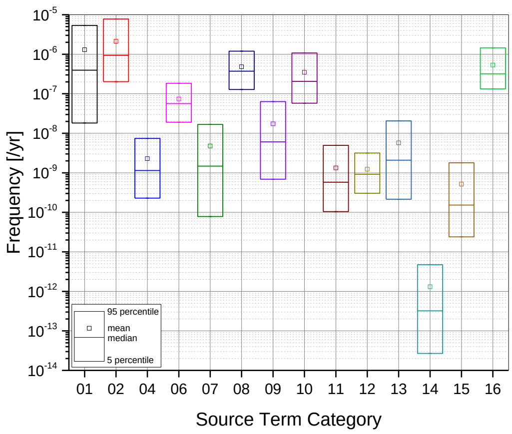

Figure 8: The result of uncertainty analysis for source term category

5 CONCLUSION

Containment failure occurs when applied stress(load) exceeds structural strength(capacity), and its probability is defined by this stress-strength relationship which is also known as a reliability physics model. Based on this method, the COFUN code was developed and quantify ECF and LCF occurrence probabilities in the CET. Level 2 PRA for internal events is performed using COFUN, while MELCOR handles source term analysis for a reference reactor. This study constructs and quantifies the PDS LD, CET, DET, and STC. Results show containment failure frequency ratios of 70% (No CF), 11% (LERF), and 18.9% (Non-LERF). Sensitivity analysis identifies TI-SGTR as a key contributor to containment integrity, and uncertainty analysis reveals BMT has the widest uncertainty range. Therefore, validation of the probability branch for TI-SGTR is essential in Level 2 PRA. These findings support enhancements in nuclear plant design and containment performance. This study focuses on internal events; future work should address external hazards such as earthquakes, fires, and tsunamis.

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