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Quantification of Failure Probabilities for the i-SMR Reactor Vessel Using MELCOR-Based Uncertainty Analysis

Sein Hong [a], Jin Hee Park [b], Jaehyun Cho [a*]
[a]*Chung-Ang Univ., 84 Heukseok-ro, Dongjak-gu, Seoul, 06974, Republic of Korea*
[b]*Korea Atomic Energy Research Institute, Daedeok-daero 989-111, Yuseong-gu, Daejeon, 34057, Republic of Korea*
[*]*Corresponding author: jcho@cau.ac.kr*

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1. Introduction

Level 2 Probabilistic Safety Assessment (PSA) evaluates severe accident progression and quantifies containment failure probability and large early release frequency (LERF). For the innovative small modular reactor (i-SMR) under development in Korea, certain severe accident mitigation features, such as external vessel cooling or dedicated backup spray systems, are not assumed in the current design; therefore, reactor vessel (RV) failure becomes a key safety metric that may directly challenge containment vessel (CV) integrity. Under this design philosophy, the RV integrity plays a critical role in preventing ex-vessel accident progression and subsequent CV loading. This study focuses on quantifying the conditional probability of RV failure for the i-SMR using a mechanistic severe accident analysis framework. By applying MELCOR-based uncertainty analysis within a Level 2 PSA context, uncertainties associated with core degradation, corium relocation, and vessel rupture phenomena are systematically propagated to estimate the RV failure probability, thereby providing a physically grounded and plant-specific assessment for advanced SMR safety assessment.

2. Methods and Results

To quantify the RV failure probability in a systematic manner, a structured analysis procedure was established within the Level 2 PSA framework. The overall approach consists of sequence clustering and selection, development of a plant-specific MELCOR model with identified uncertain parameters, sampling-based uncertainty analysis, and statistical estimation of RV failure probability.

2.1 Clustering of Core Damage Sequences and Selection of Representative Sequences

Among the 21 initiating events defined in Level 1 PSA, sequences with frequencies lower than 1.0E-12 per reactor-year were screened out, and anticipated transients without SCRAM (ATWS) were excluded from the scope of this study. The frequencies of the remaining initiating events and the corresponding numbers of core damage (CD) sequences considered in this study are summarized in Table I. As a result, a total of 34 CD sequences were selected for further analysis.

Table I: Initiating Event Frequencies and Number of CD Sequences Considered

Initiating Event	Frequency	# of Sequences
GTRN	-	5
LOCA-PSRV		4
SLOCA-INCV		4
LOOP-T		7
LOCA-ERV		2
LOMF		5
LOCA-PECCS		1
LOCA-SGTR		6

As an illustrative example, Fig. 1 presents the event tree for the GTRN initiating event, where the CD sequences analyzed in this study are highlighted. For each selected CD sequence, the success or failure status of relevant safety systems was identified to define the plant configuration for subsequent thermal-hydraulic analysis. Table II summarizes the system configurations for the GTRN CD sequences considered in the MARS-KS simulations. For each sequence, thermal-hydraulic behavior up to CD was analyzed using the MARS-KS code to characterize plant response under severe accident conditions.

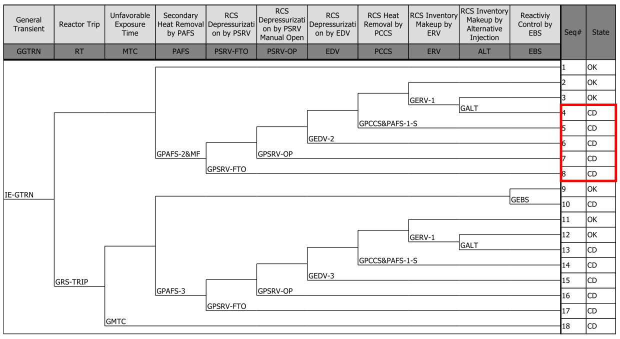

Fig. 1. Event Tree of GTRN and Analyzed CD Sequences

Table II: Safety System Configuration of GTRN CD Sequences for MARS-KS Analysis

Sequence	PAFS	PSRV	EDV	PCCS	ERV
Sequence #4	1/4	1/2 auto, 1/2 manual	2/3	2/2	0/2
Sequence #5	0/4	1/2 auto, 1/2 manual	2/3	2/2	0/2
Sequence #6	0/4	1/2 auto, 1/2 manual	0/3	0/2	0/2
Sequence #7	0/4	1/2	0/3	0/2	0/2
Sequence #8	0/4	0/2	0/3	0/2	0/2

To systematically identify accident sequences exhibiting similar dynamic behavior near CD conditions, a multivariate time-series clustering framework was developed based on six thermal-hydraulic variables: RV pressure, RV temperature, RV water level, CV pressure, CV temperature, and CV water level. These variables represent key indicators governing in-vessel thermal-hydraulic behavior and containment vessel response during severe accident progression. Because the accident sequences differ in temporal duration and sampling resolution, direct comparison of raw time-series data is not appropriate. Therefore, each time series was normalized to a common time scale and interpolated onto a unified temporal grid, enabling consistent shape-based comparison across sequences. To ensure that clustering reflects dynamic behavior rather than magnitude differences, each variable was standardized prior to analysis. The normalized multivariate time-series data were then integrated using feature-level (early) fusion by concatenating the synchronized variable sequences into a single high-dimensional representation for each sequence. Principal component Analysis (PCA) was applied to reduce dimensionality while preserving the dominant variance structure, and clustering was performed using the K-means algorithm in the PCA-transformed space. The optimal number of clusters was determined using the silhouette score. The resulting clustered dynamic profiles are shown in Fig. 2-4, which present the mean and standard deviation of RV and CV pressure, temperature, and water level for each cluster. Table III summarizes the CD sequences included in each cluster and identifies the representative sequence selected based on occurrence frequency. These representative sequences were subsequently used for the MELCOR-based uncertainty analysis. This clustering approach reduces redundant accident scenarios while preserving distinct dynamic characteristics relevant to RV failure evaluation.

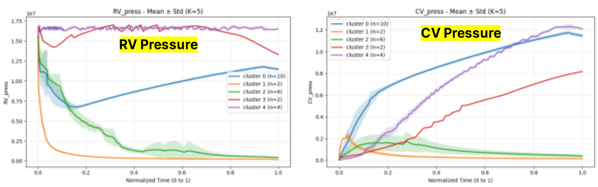

Fig. 2. Cluster-averaged RV and CV Pressure Time Histories (mean ± standard deviation)

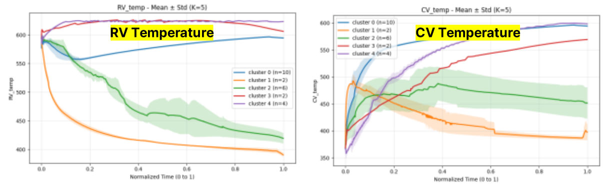

Fig. 3. Cluster-averaged RV and CV Temperature Time Histories (mean ± standard deviation)

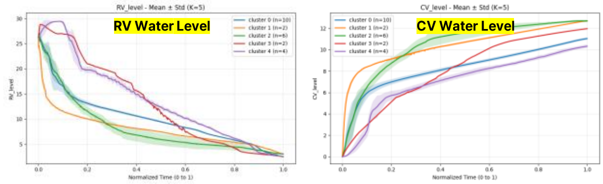

Fig. 4. Cluster-averaged RV and CV Water Level Time Histories (mean ± standard deviation)

Table III: CD Sequence Clustering Results and Selected Representative sequences

Cluster	Accident Sequence		Representative Sequence
Cluster 0	GTRN-seq5	GTRN-seq6	GTRN-seq5
	LOCA-PSRV-seq4	LOCA-PSRV-seq7	
	LOMF-seq5	LOMF-seq6	
	LOOP-seq8	LOOP-seq9	
	SGTR-seq7	SGTR-seq8	
Cluster 1	LOCA-PSRV-seq3	LOCA-PSRV-seq6	LOCA-PSRV-seq3
Cluster 2	GTRN-seq4	LOMF-seq4	SLOCA-INCV-seq3
	LOOP-seq7	SGTR-seq6	
	SLOCA-INCV-seq3	SLOCA-INCV-seq6	
Cluster 3	SLOCA-INCV-seq4	SLOCA-INCV-seq7	SLOCA-INCV-seq4
Cluster 4	GTRN-seq7	LOMF-seq7	GTRN-seq7
	LOOP-seq10	SGTR-seq9	

2.2 Development of MELCOR Input Model and Identification of Uncertain Parameters

A plant-specific MELCOR input model was developed to simulate severe accident progression of the i-SMR up to RV failure, capturing the sequential phenomena of core degradation, corium relocation to the lower plenum, and thermal loading of the vessel. The adopted nodalization scheme is shown in Fig. 5.

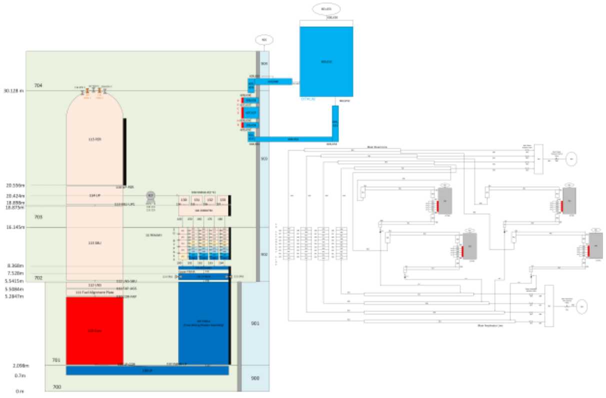

Fig. 5. MELCOR Nodalization for the i-SMR

In this study, RV failure was defined as the figure of merit (FOM) for the uncertainty analysis. Accordingly, uncertain input parameters were selected based on their physical relevance to the progression from core degradation to corium relocation and eventual vessel rupture. A total of eight MELCOR input parameters were identified as key contributors to uncertainty in RV failure prediction. For each parameter, the corresponding MELCOR record, probability distribution type, and uncertainty range were defined based on available literature and engineering judgment. The selected uncertain parameters and their statistical characteristics are summarized in Table IV.

Table IV: MELCOR Uncertain Input Parameters

Parameter	MELCOR Record	Distribution Type	Range
Oxidation kinetics	COR_OX	Discrete	0 / 1
Multiplier of ANS decay heat	SC3200	Uniform	0.9-1.1
Effective temperature at which the eutectic formed from ZrO2 and UO2 melts	-	Normal (Mean: 2479, σ: 83)	-
Zircaloy melt breakout temperature	SC1131	Scaled Beta (α: 3.83, β: 3)	2100-2540
Molten clad drainage rate	SC1141	Triangular (Mode: 0.2)	0.1-2.0
Radial molten debris relocation time constant	SC1020	Uniform	100-1000
Melt relocation heat transfer coefficient	COR_CHT / HFRZZR / HRFZZX / HFRZUOH	Triangular (Mode: 7500)	2000-22000
Failure temperature of lower head	COR_LHF	Triangular (Mode: 1273)	1000-1500

2.3 Sampling Strategy and Determination of Sample Size

A sampling-based uncertainty analysis was performed to propagate epistemic uncertainties in selected MELCOR input parameters to the FOM, defined in this study as RV failure. The uncertain parameters were assigned appropriate probability distributions based on literature data and engineering judgment, and random samples were generated assuming independent variation unless specific correlations were defined. A Monte Carlo-based sampling approach was adopted to evaluate the variability of severe accident progression up to the defined FOM.

The required sample size was determined using Wilks' formula to ensure a statistically meaningful tolerance limit for the RV failure probability. In this study, a third-order Wilks' formula was applied to obtain a one-sided upper tolerance limit with a 95% confidence level and 95% coverage probability. Based on this criterion, the minimum required sample size was calculated to be 124. Accordingly, 124 simulations were performed for each representative sequence to ensure that the estimated RV failure probability was statistically reliable.

3. Conclusions

This study presented a systematic framework for quantifying the RV failure probability of the i-SMR within a Level 2 PSA context. Instead of relying on legacy data or expert judgment, CD sequences were clustered into representative dynamic behavior groups, and selected sequences were analyzed using a plant-specific MELCOR model. Key severe accident parameters associated with RV failure – core degradation, corium relocation, and vessel rupture – were treated as uncertain variables, and a sampling-based uncertainty analysis was performed to propagate their effects on accident progression.

Through repeated simulations, the conditional probability of RV failure was statistically estimated. The results provide a physically grounded basis for determining branch probabilities associated with RV failure in Level 2 PSA with improved reliability. The proposed methodology enhances the physical realism and traceability of branch probability quantification and contributes to strengthening the technical robustness of severe accident evaluation for advanced SMR designs. Furthermore, this approach supports risk-informed safety assessment of innovative reactor concepts.

ACKNOWLEDGEMENT

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