Transactions of the Korean Nuclear Society Autumn Meeting
Changwon, Korea, October 24-25, 2024

Uncertainty Analysis for Cs-137 Quantification for Risk Safety Goal Assessment

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***Keywords :** release mass uncertainty, source term category, Level 2 PSA, Cs-137, MAAP5

1. Introduction

The probability of a severe accident in a nuclear power plant is extremely low, but if it occurs, it can have a significant impact on the surrounding area. Therefore, it is essential to conduct a 'safety assessment' that realistically analyzes and predicts the frequency and consequences of such accidents. However, in Level 2 of Probabilistic Safety Assessment (PSA), thermal-hydraulic calculations and risk evaluations have been conducted very conservatively due to the high uncertainty of the severe accidents.

According to the Nuclear Safety and Security Commission's regulations on the "Detailed Standards for Accident Management Scope and Evaluation of Accident Management Capability," the 'Risk Assessment' conducted through PSA must be suitable for comprehensive evaluation of the accidents risk in power reactor facilities. For example, one of the key objectives in PSA is to ensure that the sum of accidents frequency resulting in Cs-137 releases exceeding 100 TBq remains below E-06 per year.

To reduce the conservatism of Level 2 PSA, it is essential to identify and quantify the uncertainty factors inherent in the analysis of severe accident phenomena. This paper analyzes the uncertainty of Cs-137 release, which is one of the key results of Level 2 PSA and a criterion for regulatory compliance in risk assessment, for representative accident scenarios specific to each Source Term Categories (STC) of OPR-1000.

Furthermore, this paper emphasizes the necessity for uncertainty analysis of release frequency and ultimately aims to establish a framework for performing efficient PSA for future advanced reactors and their specific characteristics.

2. Methods and Results

Uncertainty analysis is conducted for quantitative assessment of the impact of input values and the model on specific analysis results. This study follows the methodology of five steps for the uncertainty analysis using MAAP5 Code, as a Monte-Carlo method. KAERI (Korea Atomic Energy Research Institute) Research Report (Ref. 4) suggested this framework in 2020, and some modification with automation process were added in this case study.

First, input files of MAAP5 for representative accident scenarios by STC are prepared. Second, the parameters are selected, and their distributions are determined as objective of uncertainty analysis. Third, values for the uncertainty variables are sampled to create MAAP5 input files. Fourth, the sampled values are applied to the input files, and the MAAP5 simulations are run. Finally, the results are collected and organized for further analysis.

2.1 MAAP5 Input files for representative accident scenarios of each STC

The representative accident scenarios for each STC are determined based on a combination of Level 2 PSA analysis results (Fig. 1). The representative OPR accident scenarios are presented in the KAERI Research Report (Table I).

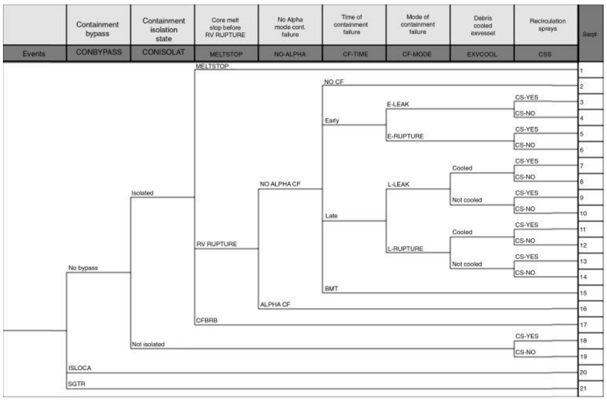

Fig. 1. Containment event tree for OPR1000 (from Ref. 5)

Numerous accident scenarios are grouped into 21 STCs. Through MAAP5 simulations, these can be further categorized into the Late Release group and the Early Release group, which includes SGTR, based on the timing of the release.

Table I: Representative accident scenarios and frequencies by OPR-STC-01 to OPR-STC-21

#STC	원자로 건물 손상모드	원자로 건물 손상 특성	방출군 빈도	대표시나리오 빈도	대표 시나리오
1	NOCF		5.19E-07	1.52E-07	MLOCA3-CET3
2	NOCF		1.20E-06	2.17E-07	SBOR38-CET4
3	ECF	LEAK, CS-YES	1.09E-08	5.29E-09	SBOR38-CET18
4	ECF	LEAK, CS-NO	8.35E-09	4.78E-09	TLOCCW4-CET20
5	ECF	RUPTURE, CS-YES	1.81E-08	8.51E-09	SBOR38-CET22
6	ECF	RUPTURE, CS-NO	1.44E-08	6.66E-09	TLOCCW4-CET24
7	LCF	LEAK, cooled, CS-YES			
8	LCF	LEAK, cooled, CS-NO	1.15E-07	8.39E-08	TLOCCW4-CET12
9	LCF	LEAK, not cooled, CS-YES	1.60E-12	3.48E-13	SBOR40-CET78
10	LCF	LEAK, not cooled, CS-NO	1.25E-07	3.13E-08	TLOCCW4-CET38
11	LCF	RUPTURE, cooled, CS-YES			
12	LCF	RUPTURE, cooled, CS-NO	1.28E-07	9.75E-08	TLOCCW4-CET13
13	LCF	RUPTURE, not cooled, CS-YES	5.34E-13	1.16E-13	SBOR40-CET79
14	LCF	RUPTURE, not cooled, CS-NO	1.30E-07	3.64E-08	TLOCCW4-CET39
15	BMT		1.80E-08	3.81E-09	LODCA16-CET59
16	ECF	ALPHA	4.88E-09	8.62E-10	LODCA16-CET70
17	CFBRB		4.15E-07	1.75E-07	SLOCA2-CET98
18	NOISO	CS-YES	2.69E-09	1.01E-09	LSSB-OUT55-CET1
19	NOISO	CS-NO	1.08E-09	2.37E-10	SBOR45-CET2
20	BYPASS	ISLOCA	1.01E-08	1.01E-08	ISLOCA1-CET99
21	BYPASS	SGTR	2.37E-07	1.15E-07	SGTR17-CET100

As shown in the table I, the information that needs to be included in the MAAP5 input consists of the accident progression from the PDS ET and CET sequences. The report also suggests factors that require the analyst's assumptions or expert judgments.

2.2 Selection of Parameters and Determination of Distributions

The information provided for each release category in the results of Level 2 PSA is expressed in terms of the magnitude of the radiological source term (release fraction) over the duration of the release. Consequently, all parameters related to severe accident progression and radiological source term behavior can potentially influence these results. However, practically speaking, it is not feasible to include all variables from the model and control sections of the MAAP parameter file in the uncertainty analysis.

Therefore, along with analyst's assumptions and expert judgments, key parameters related to severe accident progression and source term behavior, as suggested by Fauske & Associates, were included in the analysis. These parameters encompass uncertainties of core melt progression, lower containment accident progression, molten core material-concrete interactions, and the dynamics of fission products and aerosols, as well as uncertainties in the accident sequences themselves.

For analysis parameters where optimal, minimum, and maximum values are known, a triangular distribution is applied, with the probabilities of the minimum and maximum values set to zero. This applies to most of the parameters in the analysis. In contrast, if there is no meaningful optimal value or if the value is not well-defined, a uniform distribution is assumed. For cases involving specific model selections, a discrete probability distribution is assumed. (Fig. 2)

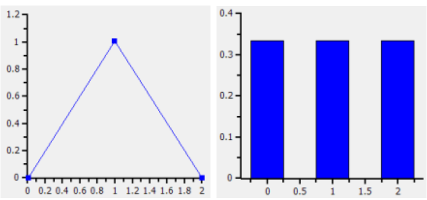

Fig. 2. (a), (b). Examples of triangular and discrete parameter distribution

2.3 Sampling of Parameters

Using the parameters and distribution information selected in the previous step, the parameter values are sampled. In this study, 124 sets of parameter samples were generated using MOSAIQUE (Module for SAmpling Input and QUantifying Estimator) software.

The number of parameter sets follow the Wilks' Method, which ensures a statistical confidence interval while minimizing the sample size. With a minimal sample size, this method can achieve a 95% confidence interval. For instance, if the 124 output values sampled in this study are considered as a sample under Wilks' Method, the third largest value among them would represent the 95% confidence point.

In other words, the third largest value in each of these outputs of STCs can be considered the 95% confidence point, representing the most probable final release amount for each STC. This will be revisited in the results summary stage.

2.4 Application of Sampled Values and Run MAAP5

In this study, uncertainty analysis was conducted on two representative scenarios, one from late release and one from early release, out of the 21 STCs. STC-07 and STC-11 were excluded as they did not have associated frequency values. To select the scenarios, MAAP5 simulations were performed using default parameter, and the results were plotted for comparison (Fig. 3).

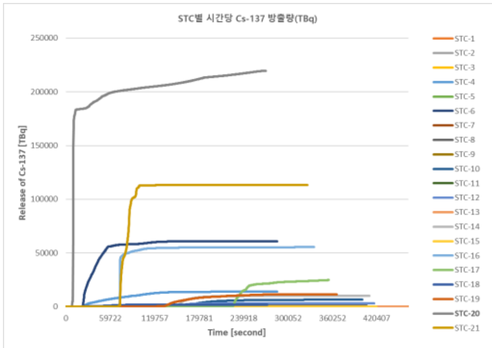

Fig. 3. Cs-137 release mass [TBq] by time [sec] of 21 STCs with default parameters

The scenario with the most rapid and significant release, STC-20, was selected as the early release case. The representative accident scenario for STC-20 is 'ISLOCA (Interfacing System Loss-of-Coolant Accident),' an accident in which primary system coolant is directly lost outside the containment through the low-pressure boundary of the reactor coolant system. Next, to select the late release case, the MAAP5 simulation results of STC-02 and STC-17 were compared (Table II).

Table II: STC Comparison for representative selection with default parameters

	LATE Release Case		EARLY Release Case	
	STC-02*	STC-17	STC-20*	STC-21
Cntm Damage Mode	NOCF	CFBRB	BYPASS	BYPASS
Damage Characteristic	-	-	ISLOCA	SGTR
Representative Scenario(RS)	SBOR38-CET4	SLOCA2-CET98	ISLOCA1-CET99	SGTR17-CET100
	High Frequency			*High Frequency*
STC Frequency	1.20E-06	4.15E-07	1.01E-08	2.37E-07
RS Frequency	2.17E-07	1.75E-07	1.01E-08	1.15E-07
		Later Release	*Earlier Release*	
First Release Time[sec]	108351.3517	222672.7038	8103.37722	70225.23384
First Release Mass[TBq]	2.55E-02	1.786743851	4158.636794	182.8501319
		Release More	*Release More*	
Run Time[sec]	366067.7803	355381.9997	269658.361	326600.7586
Final Release Mass[TBq]	53.29686874	24798.46461	219476.4584	113392.7106

STC-02 represents NOCF (No Containment Failure), while STC-17 represents CFBRB (Containment Failure

Before Vessel Breach). Among these, the scenario that releases Cs-137 later and in larger quantities is STC-17. However, STC-17 has a less realistic scenario, as it assumes that human operators would take no action for 2-3 days, allowing containment pressure to rise, with the situation of core cooling is possible during this time. Considering this, STC-02 was selected as the late release case, as it has the highest frequency among the late release scenarios.

On the other hand, the simulation performed with default parameters for STC-02 resulted in a release amount of approximately 53 TBq, which falls significantly short of the 100 TBq, which is the threshold standard for regulatory adequacy assessment. Therefore, an initial analysis using 40 samples was conducted. The results showed that a substantial number of cases exceeded 100 TBq, with average value 119.93 TBq, confirming that this scenario was suitable for further analysis. To easily apply the sampled values to each input file and to facilitate the management of output files, Python code programs were utilized.

3. Conclusions

3.1 Results

Fig. 4 and Fig. 5 show the results of the uncertainty analysis conducted on Cs-137 release for STC-02, the representative case of a late release accident. The maximum final release amount was 366.05 TBq, while the minimum was 20.31 TBq. In this analysis, Cs-137 release mass of 67 cases (over 124 cases) exceeded 100 TBq which is the threshold standard for regulatory adequacy assessment.

Additionally, the third-largest release value (indicated by the pink dot with thinner dotted line) was 292.73 TBq. This represents the 95% confidence point, meaning there is a 95% probability that the Cs-137 release amount will fall within the top 95% percentile in the event of this accident.

The next two graphs (Fig. 6, 7) show the results of the uncertainty analysis conducted on Cs-137 release for STC-17, the representative case of an early release accident. The maximum final release amount was 322,575.24 TBq, the minimum was 183,103.52 TBq, and the third-largest release value was 312,474 TBq. The ISLOCA corresponds to a Large Early Release accident, with all Cs-137 release result amount values significantly exceeding 100 TBq.

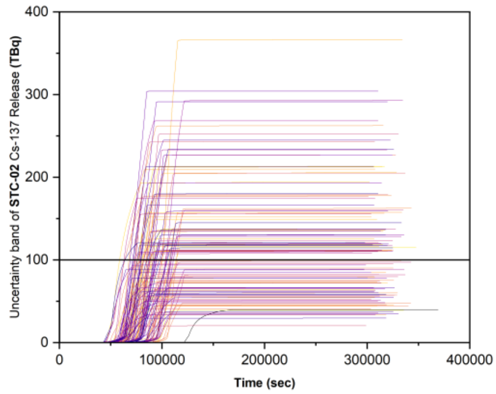

Fig. 4. Cs-137 release mass [TBq] by time [sec] of STC-02 (NOCF), for 124 parameter samples

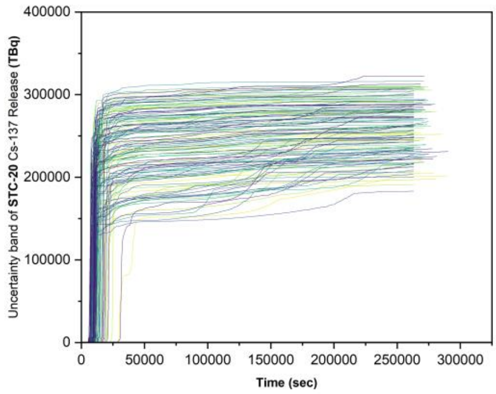

Fig. 6. Cs-137 release mass [TBq] by time [sec] of STC-20 (ISLOCA), for 124 parameter samples

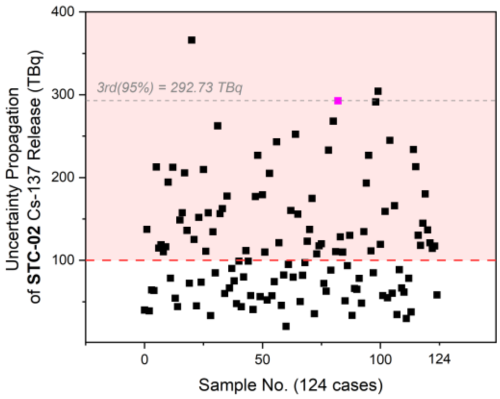

Fig. 5. Distribution of STC-02(NOCF) Cs-137 release mass.

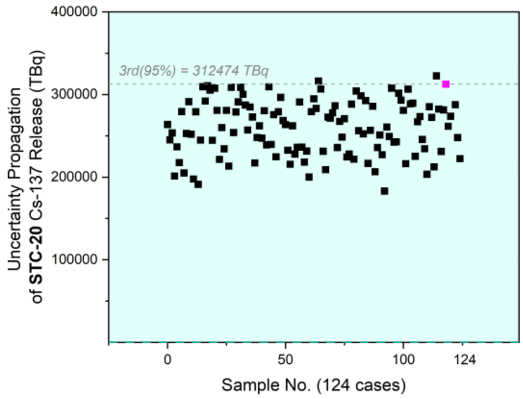

Fig. 7. Distribution of STC-20(ISLOCA) Cs-137 release mass

3.2 Discussion

To reduce the conservatism of Level 2 PSA, the uncertainty analysis of severe accident phenomena was conducted. In this study, the uncertainty analysis focused on the release amount of Cs-137, an important source term in the events of an accident, by selecting representative STCs for the OPR-1000 reactor. This analysis uses MAAP5 program as a Monte-Carlo Method. Also, this paper uses Wilks' Method to determine the confidence intervals (95/95).

Considering the 100 TBq Cs-137 release criteria established by the Nuclear Safety and Security Commission's regulations, the uncertainty analysis of the ISLOCA Cs-137 release, as shown in Fig. 7, revealed that the release amount exceeded the standard by more than 1,000 times. Therefore, it is essential to apply the uncertainty analysis framework demonstrated in this study to other STCs and incorporate the findings into revisions aimed at reducing excessive conservatism.

While ensuring compliance with the current risk safety goals when Cs-137 release exceeds 100 TBq, this approach should complement existing conservative and fragmented cases by reevaluating with the uncertainty concerns. It will help establish Cs-137 release criteria that account for a wider range of scenarios and frequencies.

During the processing input and output files of MAAP5, additional software such as MOSAIQUE and Python were utilized to establish an automated sampling and workflow. This automation enables a streamlined and efficient uncertainty analysis following the established framework.

3.3 Further work & Application

Although this study conducted the analysis on the representative scenarios for the OPR-1000 STCs, it is also necessary to the radiological source term for STCs developed through Level 2 PSA for other reactor types.

The uncertainty factors influencing severe accident phenomena are not limited to Cs-137 release mass. For example, frequency of Cs-137 releases uncertainty analysis also should be concerned. And combining with the previous mass uncertainty analysis will help us to develop more comprehensive risk index.

Concerns have been raised about the domestic model of Level 2 PSA. For instance, the probabilities of severe accident phenomena in domestic nuclear power plant PSA models heavily rely on the results presented in NUREG-4551. Additionally, the lack of consistency among the Level 2 PSA models for different domestic nuclear power plants hinders effective specific risk assessment for these plants.

As further work, we will conduct an analysis for Cs-137 release frequency to address these issues, using study data on expert judgment. The new methodology will be developed through this analysis and will help reduce the conservatism of Level 2 PSA enhancing its practical applicability. Moreover, this uncertainty analysis process could be applied to a range of reactors, from newly designed advanced reactors to various types of small modular reactors (SMR), improving the efficiency of safety assessment in the future.

Acknowledgements

This research was supported by a grant from Korea Research Institute of Ships and Ocean engineering Endowment Project of “Study on Concept Design of SMR-powered Ship” funded by Ministry of Oceans and Fisheries (PES5122).

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