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

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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.
- ✓ Article 9 (Risk Assessment) of the Nuclear Safety and Security Commission's "Detailed Standards for Accident Management Scope and Evaluation of Accident Management Capability" regulation specifies that the results of **Probabilistic Safety Assessment** (PSA) must be utilized to enhance the prevention and mitigation capabilities of severe accidents. Among the regulatory targets to be applied in PSA is the requirement that "the total frequency of accidents resulting in the **release of more than 100 TBq of Cs-137 must be less than 1E-6 per year**."
- ✓ According to current operational situation, this target follows existing standards and has adopted a conservative approach. In this study, uncertainty analysis was conducted to complement the **Risk Assessment framework**.

2 Methods

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. Numerous accident scenarios of are grouped into 21 STCs in the case of OPR-1000 reactor type. 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.

2) Selection of Parameters and Determination of Distributions

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.

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 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. 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.

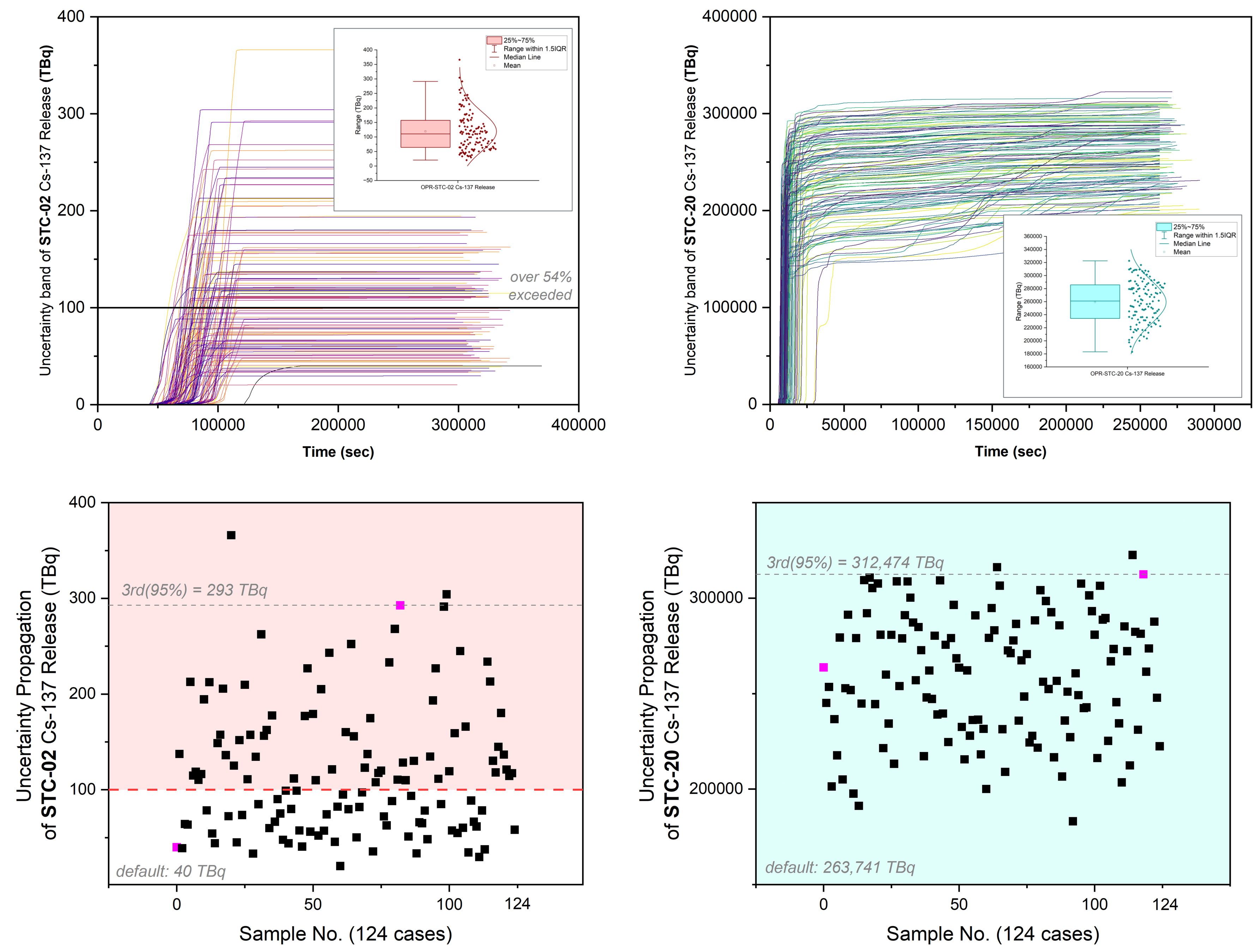
Table 1. STCs Comparison for representative case selection with default parameters

	LATE Release Case		EARLY Release Case	
	STC-02*	STC-17	**STC-20***	STC-21
Ctmt. Damage mode	NOCF	CFBRB	BYPASS	BYPASS
Representative scenario	SBOR38-CET4	SLOCA2-CET98	ISLOCA	SGTR
STC Frequency	**1.2E-06**	4.2E-07	1.0E-08	2.4E-07
First release time [sec]	108,351	222,673	**8,103**	70,225
Final release mass [TBq]	40	24,800	**220,000**	113,400

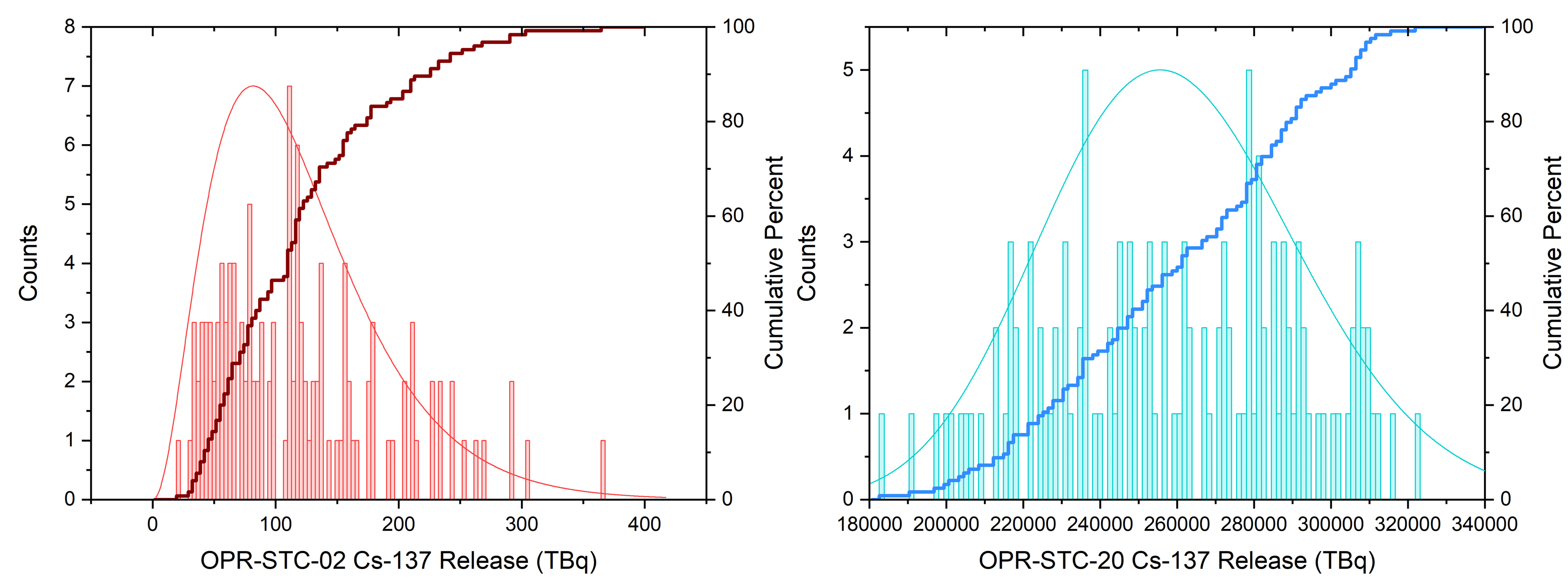
4) Application of Sampled Values and Run MAAP5

In this study, analysis was conducted on two representative scenarios, one from late release and one from early release. The scenario with the most rapid and significant release 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. And **STC-02 represents 'NOCF (No Containment Failure) with reactor vessel rupture'** was selected as the late release case, as it has the highest frequency among the late release scenarios.

3 Results



- In the case of STC-02, The maximum Cs-137 release amount was about 366 TBq, while the minimum was 20 TBq. In this analysis, Cs-137 release mass of 67 cases (over 124 cases, 54%) exceeded 100 TBq which is the threshold standard for regulatory adequacy assessment. Additionally, the third-largest release value was 293 TBq. It is noticeable that this scenario involves a relatively high frequency (1.2E-06) without containment failure.
- For STC-20, maximum final release amount was 322,575 TBq and minimum was 183,104 TBq. The third-largest release 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.



4 Conclusion & Further works

- ➢ The results suggest that the evaluation method should be improved to account for uncertainty through the analysis of various scenarios to ensure regulatory compliance heavily rely on the results presented in NUREG-4551.
- ➢ This analysis will help reduce the conservatism of Level 2 PSA and will serve as a critical foundation for developing a future framework that considers uncertainty not only in release amounts but also in release frequency. This methodology will become a key tool for safety assessment in the design of various reactors, including domestic advanced reactors and SMRs.

~ References ~

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