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Development of Level 2 PSA Model and Uncertainty Analysis for WH600

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1. Introduction

The oldest type of nuclear power plant in Korea is the WH600, and there is increasing concern about its safety as the plants operating this type have recently applied for a life extension. Some environmental groups have claimed that there is a lack of proper assessment of the risk of a major accident related to the WH600 plant type. In this study, the safety of the reference unit, WH 600, was evaluated from a Level 2 PSA (Probabilistic Safety Assessment) perspective [1]. The accident progression that could occur within the containment building during core damage was investigated, and the timing and types of containment building damage were predicted. The performance of the containment building was analyzed by evaluating the probability of occurrence and radiological consequences of each accident scenario. Based on these results, the ability of the containment system, which is critical to reducing the risk of major accidents and improving the safety of the nuclear power plant, was evaluated, and the necessary information was provided to establish an accident management plan by reviewing the analysis results. Ultimately, it is expected that the safety of nuclear power plants can be improved [2].

2. PDS Analysis Process

The connection between Level 1 PSA and containment performance analysis lies in classifying the progression of a core damage accident into Plant damage states (PDS) that represent the plant's state. The variables used to define PDS include system operation variables such as reactor coolant system pressure, and important functions and early event types of critical systems that affect the analysis of failure modes and radiological consequences [3]. Based on these variables, PDS logic diagram (PDSLD) for each initiating event was created, resulting in 58 PDS for the reference plant

Each internal event PDS event tree (PDSET) is quantified in the following manner:

- The quantification of PDSET is performed in the same way as in Level 1 PSA.
- Low-pressure safety injection after core damaged and errors in operator actions required for containment cooling are not considered due to their very low probability of occurrence.
- AiMS-PSA code, which was used for internal event analysis, is used as the computational code for quantification.

In this study, we used COFUN code to quantify PDS using the results of PDSET quantification [4].

3. CET Analysis Process

In containment event tree (CET), events that can represent the differences in accident scenarios should be selected, and these events are called the CET top events. The CET top events include important events that can have a significant impact on the damage time, type, and location of the containment building, physical phenomena that can cause severe consequences, and events that can lead to different results in the accident progression or containment building behavior.

The order of the CET top events is arranged with dependent events coming after independent events. The events are arranged in the following order: events before core exposure, events after core exposure and before the rupture of the reactor pressure vessel, early events after the rupture, mid-term events after the rupture, and late-stage events after the accident. The CET for the shutdown operation of reference plant was based on the PDSLD for full-power operation events.

The branch probability of top events in CET are being calculated for each decomposition event tree (DET)s. Therefore, DET is composed of important conditions that greatly contribute to the branch probability of the top events.

4. STC Analysis Process

To define the source term categories (STC), cluster variables were first selected. These variables were defined based on their characteristics that affect the release of nuclear fission products and accident consequences, and the STC was defined by the classification criteria of each variable, including the size, composition, and timing of the radiation source term. Cluster variables are selected based on the unique characteristics of the power plant and containment building. The STC of reference plant was classified based on the following characteristics of selected accident scenarios

- Containment building bypass
- Containment building isolation status
- Core melting stop before vessel failure/In-vessel core cooling
- Containment building damage type and timing
- Containment building sump recirculation mode operation
- Nuclear fission product scrubbing (SCRUB)

The method of defining the STC is similar to that of defining the PDS. That is, the STC logic diagram for reference plant was prepared by taking the six STC variables as top events, and 13 STCs were defined for the prepared events for reference plant. The frequency of each STC is calculated by adding the frequency of the accident sequence of the containment building corresponding to each STC [5].

The frequency of incidents in the containment is calculated by multiplying the probability of incidents in the containment by the PDS frequency. This calculation is performed using the COFUN code [4].

Table 1. Containment Failure Mode Frequency

Containment Failure mode		Frequency (/RY)	Percentage (%)
NO CF		5.53E-06	38.3
CF	ECF	4.21E-07	2.9
	LCF	7.00E-06	48.6
	BMT	2.31E-07	1.6
	CFBRB	2.31E-09	<0.1
	NOT ISO.	5.93E-07	4.1
	BYPASS	6.48E-07	4.5
	Total of CF	8.90E-06	61.7
LERF		1.66E-06	11.5
Total		1.44E-05	100.0

5. Uncertainty Analysis

Uncertainties in the performance analysis of the containment building arise mainly from two sources. One is the uncertainty of the PDS frequency, and the other is the uncertainty of the branch probabilities used to quantify CET. To analyze PDS frequency uncertainty, the COFUN code generates a minimal cut set contributing to the PDS by quantifying PDSET (Level 1.5 PSA), and the minimal cut set can be directly assigned to the PDS [4]. When calculating PDS frequency, the uncertainty of the data (initial event frequency, component failure rate data, human error probability, etc.) is considered, and these data values are sampled using the Monte Carlo method [6]. The 5th percentile, median, 95th percentile, and mean values were calculated for the frequency of containment failure mode [7].

For the CET branch probability uncertainty analysis, we analyzed the culmination events of DET that are designated as branches with subjective probabilities, which are summarized in the Table 1. The uncertainty distribution of the selected branch probability must be set to perform the uncertainty analysis, and the branch probability distributions used in this analysis are summarized in the Table 2.

For the propagation of uncertainty, it starts from the minimal cut set generated during the PDS ET quantification and propagates through the PDSLD, CET, DET, and STC to the quantification results at each step. The Monte Carlo method is used to perform uncertainty propagation with 10,000 samplings. Table 3 shows the results of the containment failure mode uncertainty analysis.

Table 2. Variables and Distributions Used in CET Uncertainty Analysis

DET	Variables	Distribution
RCSFAIL	RCSFAIL	Lognormal
MELTSTOP	MELTSTOP	Lognormal
CR-EJECT	CR-EJECT	Lognormal
CF-EARLY	EVSE	Lognormal
	H2-MASS	Uniform
	H2-BURN	Uniform
CHR-LATE	CS-DEBRIS	Lognormal
CF-LATE	ERLY-BURN	Uniform
	HMS	Lognormal
	LATE-BURN	Lognormal
BMT-MELT	DB-DEPTH	Lognormal
	EXVCOOL	Lognormal/Uniform
	BMT-MELT	Lognormal

Table 2. Variables and Distributions Used in CET Uncertainty Analysis

Failure Mode	Point Estimate	Mean	5th Percentile	Median	95th Percentile
NO CF	5.526E-6	5.659E-06	2.043E-06	4.298E-06	1.169E-05
ECF	4.210E-7	4.214E-07	6.114E-08	2.577E-07	1.238E-06
LCF	7.004E-6	6.959E-06	1.554E-06	4.076E-06	1.983E-05
BMT	2.314E-7	2.183E-07	3.163E-08	1.227E-07	6.819E-07
CFBRB	2.309E-9	2.207E-09	1.929E-10	1.090E-09	7.058E-09
NOT ISO.	5.928E-7	5.878E-07	9.007E-08	3.325E-07	1.762E-06
BYPASS	6.483E-7	6.572E-07	1.654E-07	4.738E-07	1.716E-06
LERF	1.664E-6	1.666E-06	5.007E-07	1.228E-06	4.155E-06

The results of the uncertainty analysis of containment failure mode and STC are shown in the following Figures 1 and 2.

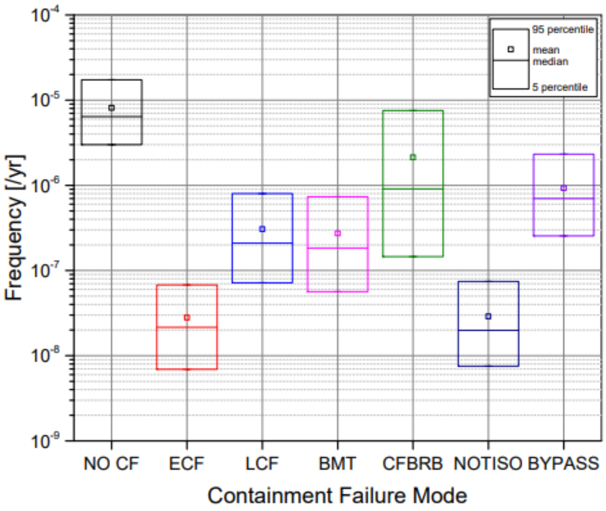

Fig. 1. Containment Failure Mode Uncertainty Analysis Results

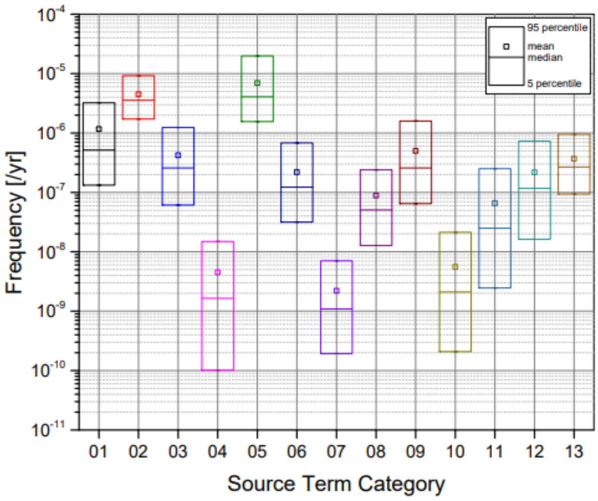

Fig. 2. STC Uncertainty Analysis Results

6. Results

According to the Level 2 PSA results for WH600, the containment building integrity failure probabilities due to core damage frequency (CDF) is 61.7%. The large early release frequency (LERF) is 6.650E-06/yr, which satisfies the quantitative target that the LERF for existing nuclear power plants should be less than 1.0E-05. The most frequent initiating event is PDS 28 with a frequency of 7.382E-05/yr. This PDS is a severe event that bypasses the containment isolation, and safety injection, emergency cooling, and containment venting systems all fail. The reactor cooling system pressure remains high until the reactor vessel is damaged, and the reactor cavity remains dry. The most frequent release pathway is STC 5 (late containment failure), with a frequency of 7.000E-06/yr. Most accidents in this release pathway are due to the failure of all engineered safety features, which results in the reactor vessel being damaged and the containment building being damaged by overpressure because of the absence of heat removal from the containment building.

7. Conclusion

This study conducted a Level 2 PSA analysis on the reference plant, WH600. Based on the results of the preceding Level 1 PSA analysis, the PDS, CET, and STC were quantified and uncertainty analysis was performed. Through the study, it was possible to identify which accident scenarios WH600 is particularly vulnerable to from the perspective of the containment building and to understand the potential consequences in the event of an accident. However, there are also limitations. In this study, the Multiple Barrier Accident Coping Strategy (MACST) strategy was not considered, so additional research reflecting the Accident Management Program (AMP) is necessary. In addition, in future research, it is necessary to select variables for uncertainty analysis based on appropriate evidence for their distribution.

Acknowledgments

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