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A Sensitivity Analysis for Containment Integrity of OPR1000

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1. Introduction

After Fukushima nuclear power plant accident in 2011, external injection of emergency cooling water and Filtered Containment Venting System (FCVS) have been considered as countermeasures for severe accident in Korea. Such severe accident management strategies essentially require risk assessment before its installation to nuclear power plant. However, the risk assessment of these strategies has not been carried out in detail. Therefore, we performed sensitivity analysis of containment integrity of OPR1000 in accordance with severe accident management strategies based on probabilistic risk assessment. Such sensitivity analysis is expected to be used as a reference for accurate risk assessment in the future. [1]

2. Methods and Results

Level 2 Probabilistic Risk Assessment (PRA) is performed to identify possible accident progression in the containment building during core damage, predict the timing and type of containment building damage and evaluate a possibility of each accident sequence and source terms. Based on the Level 2 PRA results, mitigation strategies for severe accident could be evaluated and vulnerable safety system for containment integrity could be detected. It is very important to improve the containment integrity ultimately. For these reason, we performed the sensitivity analysis of containment failure according to failure probabilities of severe accident strategies such as external injection of emergency cooling water and FCVS, after selection of reference power plant and reference accident scenario.

2.1 Severe accident management strategy

In this study, external injection of emergency cooling water into 1st and 2nd loop and FCVS were selected as severe accident management strategies.

FCVS could prevent a damage to containment building just by opening of manual valve in the situation of gradual overpressure in the containment building. [2]

External injection of emergency cooling water into 1st and 2nd loop is the countermeasure to maintain containment integrity and mitigate release of radioactive materials outside of containment. The installation of external injection could be considered as a severe accident management strategy with acquisition of the mobile diesel pump. [3]

2.2 Reference plant and scenario

In this study, Hanul unit 5 and 6, which reactor type is OPR 1000, were selected as reference plants. OPR 1000 was developed by receiving abroad design technology for independence of domestic nuclear power plant design technology. And, OPR 1000 was applied to Hanbit 3, 4, 5 and 6, Hanul 3, 4, 5 and 6, Kori 5 and 6 and Shinwolsung 1 and 2. Because it occupies the largest portion in Korea, OPR 1000 was selected as the reference plant.

As a reference accident scenario, Small Loss of Coolant Accident (SLOCA) was selected. When the accident occurs, there are core uncover and a rupture of reactor coolant system because of failure of high pressure injection and secondary heat removal. For this reason, reactor coolant system is not sufficiently decompressed before reactor vessel failure. In this accident, we selected Plant Damage State (PDS) 52 considering these accident characteristics.

PDS 52 is 8.8% of total PDS frequency of internal events. It includes the situations of operation safety injection system and failure of the recirculation mode operation. [1]

Quantified Containment Event Tree (CET) of PDS 52 without severe accident management strategies is shown in Fig. 1. Based on this CET, we constructed the modified CET as shown in Fig. 2. Also, a part of modified CET is shown in Fig. 3 for readability. And, the construction and quantification of the modified CET were performed by excel-based precision tree program created by PALISADE. The headings of the modified CET and its simple descriptions are shown in table 1.

Table I: Headings of the modified CET

Heading	Description
RCSFAIL	Mode of induced primary system failure at RV failure
MELTSTOP	Core melt progression stopped before RV failure
ALPHA	Alpha mode containment failure
EXINJECT	External injection of emergency cooling water
CR-EJECT	Amount of corium ejected out of cavity
CF-EARLY	Early containment failure
CS-LATE	No late recirculation spray failure
EXVCOOL	Debris cooled ex-vessel
FCVS	Filtered containment venting system
CF-LATE	Late containment failure

BMT-MELT	Containment basemat melt through

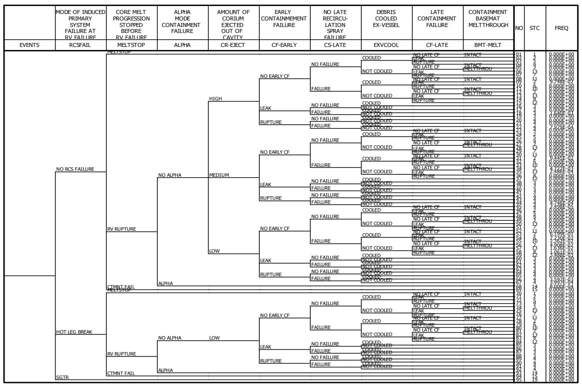

Fig. 1. Quantified containment event tree of PDS 52 without severe accident management strategies

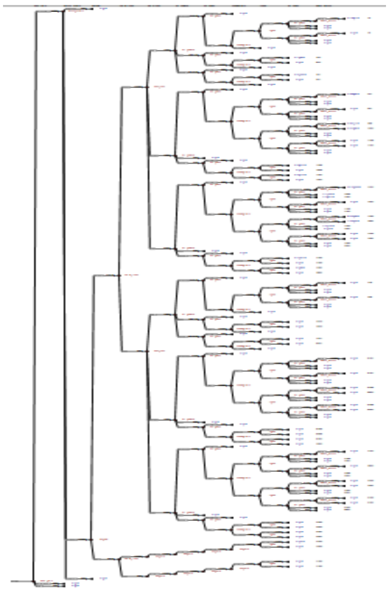

Fig. 2. The modified containment event tree by Precision Tree

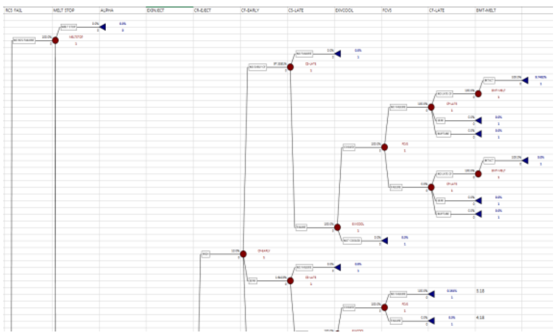

Fig. 3. A part of the modified containment event tree.

2.3 Result

We used the value of headings (RSCFAIL, MELTSTOP, ALPHA, CR-EJECT, CF-EARLY, CS-LATE, EXVCOOL and CF-LATE) from Level 2 PRA report of Hanul 5 and 6. Sensitivity analysis of 36 cases was respectively performed according to failure probabilities of EXINJECT and FCVS that are 0, 20, 40, 60, 80 and 100. In the case of PDS 52 by SLOCA, Early Containment Failure (ECF), Basemat Melt-Through (BMT) and other things are difficult to occur. Therefore, we performed the sensitivity analysis just for Late Containment Failure (LCF) and NO Containment Failure (NO CF). In this study, we supposed that all types of containment damage are LCF and NO CF.

Table II: The results of sensitivity analysis

FCVS Failure Probability	Minimum probability of LCF in PDS 52	%	Graph shape
1	0.12702609992	1	U
0.8	0.1270201893484	-0.004653037	W
0.6	0.1270142498	-0.009328886	V
0.4	0.1270142498	-0.009328886	V
0.2	0.1270201893484	-0.004653037	W
0	0.12702609992	1	U

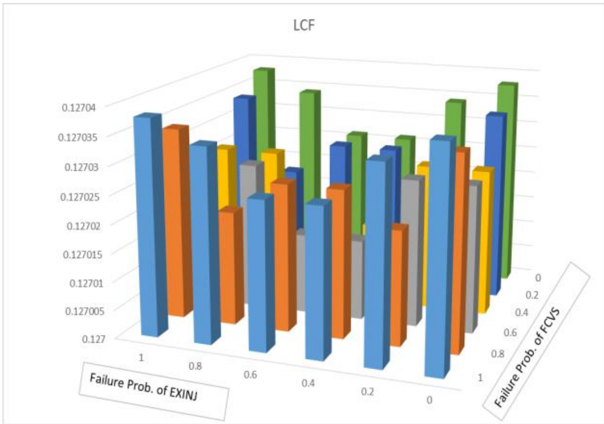

Fig. 4. Late Containment Failure

The above results in table 2 and Fig. 4 might reflect that the probability of LCF would be changed rely on the failure of not only the accident management strategies, but other strategies. Especially, we can see that the LCF probability has minimum value when the failure probability of both strategies is between 80 and 20.

3. Conclusions

Since Fukushima accident, various severe accident management strategies have been discussed in Korea. A risk assessment of the strategies must be performed before implementation of it. In this study, the sensitivity analysis of LCF was performed in accordance with failure probability of external injection and FCVS. We assumed that Hanul unit 5 and 6 were the reference nuclear power plant and SLOCA was the reference accident scenario. The methodology used in this study could be applied to other strategies based on a detailed thermo-hydraulic analysis. It might contribute to performing base studies for quantitative evaluation of severe accident management strategies. Afterward, a more detailed thermo-hydro analysis will be needed to account for changes in the accident process due to strategies. It would be helpful to construct more logical series of headings.

ACKNOWLEDGEMENT

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