Variable Importance Evaluation of Level 2 PSA based on Quantification of Containment Failure Probability

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1. Introduction

The prediction about a severe accident of nuclear power plants (NPPs) has inherently uncertainties due to complexity of phenomena and lack of experiences. To overcome these uncertainties, many experimental researches are needed as future work. A level 2 probabilistic safety assessment (PSA) quantifies the containment failure probability using logic diagram. Thus, based on level 2 PSA results, we can identify that which phenomena is important or not.

Recently, the level 2 PSA for OPR-1000 (Optimized Power Reactor) is being re-developed in KAERI as the one of the national project aiming at increasing and assuring the safety of the NPP.

This paper deals with the quantification results of the level 2 PSA model for OPR-1000. Based on the quantification results for containment failure probability, the important variables were derived.

2. Methods

Firstly, interface analysis between level 1 PSA and containment performance analysis is needed. It is grouping analysis to make the plant damage state (PDS) representing the plant state when core damage occurs. For the grouping analysis of PDS, the 10 variables were considered as shown in below. There are 54 PDS branches.

- Containment Bypass (No bypass, ISLOCA, SGTR)
- Containment Isolation (Isolated, Not Isolated, Rupture Before Core Melt)
- Accident Type (SBO, Transient, LLOCA, M/SLOCA)
- Power Recovery in SBO case (Recovery before RV failure, Recovery before containment failure, Not recovered)
- Safety Injection (On, Deadheaded, Failed)
- CSR (Yes, No)
- H2 Igniter (On, Fail)
- RCS Pressure (Low, Medium, High)
- SG Availability at initial stage (Yes, No)
- Cavity Condition (Flooded, Not flooded)

After that, a containment event tree (CET) is used to analysis the containment transients after the core damage. For the CET analysis, the 11 heading were used as shown in below. There are 100 CET branches.

- Containment Bypass (No bypass, ISLOCA, SGTR)
- Containment Isolation (Isolated, Not Isolated, Rupture Before Core Melt)
- RCS Failure (No failure, hot-leg break, TI or PI-SGTR)
- Core Melt Stop (Melt stop, RV lower plenum failure, Containment failure before RV failure)
- Alpha mode failure (Yes, No)
- Amount of Corium Ejected (High, Medium, Low)
- Early Containment Failure (No failure, leak, failure)
- Late Containment Spray Recirculation (Yes, No)
- Ex-Vessel Cooling (Yes, No)
- Late Containment Failure (No failure, leak, failure)
- Basement Melt-through (Yes, No)

To obtain the quantification results of containment failure mode, a source term category (STC) analysis is needed. 8 variables (CONBYPASS, CONISOLAT, MELTSTOP, NO-ALPHA, CF-TIME, CF-MODE, EXVCOOL, CSS) were considered and 21 ST categories were derived. With this STC results, we can identify that what failure mode are dominant.

In the quantification of CET many uncertain probabilities were used. When the probability of uncertain variables are changed, the containment failure modes distribution is also changed. By doing this sensitivity study, importance of variables could be identified.

3. Results and Discussion

Fig. 1 shows the containment failure mode distribution of OPR-1000 which is obtained by level 2 PSA. As an important measure in the level 2 PSA, large early release frequency (LERF) is defined to summation of early containment failure (ECF), isolation failure, and containment bypass. LERF is 9.4% and containment bypass (8.2%) is dominant. This is because the all core damage cases of SGTR directly contribute to the containment bypass.

In order to conduct uncertainty analysis, 37 case were developed by changing the probability of uncertain variable. Fig. 2 shows that the overall results of sensitivity analysis for uncertain variables in level 2 PSA of OPR-1000. In the results of case 2, 10, and 21, there were large discrepancies with base case results. The assumptions of the case-2, 10, and 21 are shown in the Table I, II, and III, respectively.

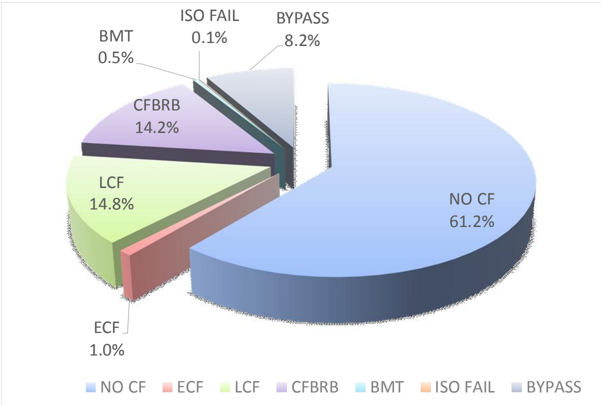

Fig. 1. Containment Failure Mode Distribution of OPR-1000 (CF: Containment Failure, ECF: Early CF, LCF: Late CF, CFBRB: Containment Failure Before Reactor Vessel Breach, BMT: Basement Melt-through)

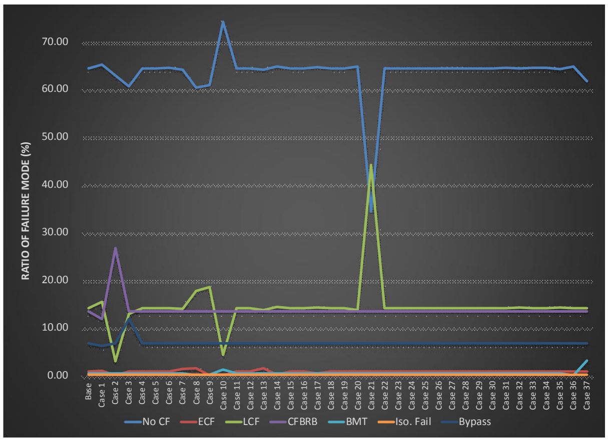

Fig. 2. Sensitivity analysis for uncertain variables in level 2 PSA of OPR-1000

CET Heading: RCSFAIL	Base case	Case-2
No RCS FAIL	0.89	0.0
	0.48	0.0
Hot Leg Break	0.1	0.99
	0.5	0.98
SGTR	0.01	0.01
	0.02	0.02

Table I. Probability of the case 2

In case-2, probabilities of hot leg break is intentionally increased. Then, RCS pressure is decreased before RV failure and it causes the positive effect to the severe accident transients. Because late containment failure (LCF) is sensitive to the RCS pressure, probability of LCF is largely decreased (-Δ11.0%). On the other hand, flooding condition probability is increased due to low RCS pressure and activating the LPSI pump. It makes containment failure before RV rupture increase (+Δ13.3%).

CET heading: CR-EJECT	Base case	Case-10
High	It depend on the situations	0.0
Medium		0.0
Low		1.0

Table II. Probability of the case-10

In case-10, amount of corium ejection is intentionally fixed to low condition. When the corium ejection mass is low, the early or late containment failure probabilities decreases due to absence of pressurizing (-Δ0.8%, -Δ9.7%). Thus, no containment probability is largely increased. On the other hand, because the all corium are accumulated in RV cavity, probability of basement melt-through is increased (+Δ0.8%).

DET heading: CS-DEBRIS	Base case	Case-21
No	0.99	0.0
Yes	0.01	1.0

Table III. Probability of the case-21

In case-21, probability of containment spray system failure by debris is intentionally 1.0. In this case, because the energy accumulated in containment cannot removed, then LCF probability is largely increased (+Δ30.2%).

4. Summary

In this paper, based on the PDS-CET-STC analysis of level 2 PSA results, containment failure mode distribution was obtained. By doing the sensitivity analysis for uncertain variables, probability of hot leg break, amount of corium ejection, and CSS failure by debris are most effectible to containment mode failure distribution.

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REFERENCES

[1] Ahn, K.I., Kim, D.H., Yang, J.E., 2002. Methodologies for Uncertainty Analysis in the Level 2 PSA and Their Implementation Procedures. KAERI/TR-2151/2002, KAERI, Daejeon, Korea.
[2] U.S. NRC, 1993, Evaluation of Severe Accident Risks: Methodology for the Containment, Source Term, Consequence, and Risk Integration Analyses, NUREG/CR-4551, Vol.1
[3] U.S. NRC, 1990, Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Platns, NUREG-1150