A Comparative Study of the Standardized MPAS Level 2 PSA Model and the Operator Model

Hyun-bin Chang [a*], Gunhyo Jung[a], Jung Hyun Ryu[a], Jaebeol Hong[a]
[a]*FNC technology Co., Ltd., 32fl, 3, Heungdeok 1-ro, Giheung-gu, Yongin-si, Gyeonggi-do*
[*]*Corresponding author: hbchang@fnctech.com*

1. Introduction

The Korea Institute of Nuclear Safety (KINS), a regulatory agency, has developed the Multi-purpose Probabilistic Analysis of Safety (MPAS) Level 1 Probabilistic Safety Assessment (PSA) model, but has yet to develop the Level 2 PSA model. In 2021, research was initiated to develop the MPAS Level 2 PSA model to future use of Risk-Informed Decision Making (RIDM). Last year, a standardized version of the MPAS Level 2 PSA model was presented after reviewing both domestic and foreign Level 2 PSA models [1].

The standardized MPAS Level 2 PSA model was developed by incorporating state-of-the-art research and considering the evaluation of portable equipment. The model was developed with reference to the APR1400 DC PSA model, which was developed by KHNP and certified by the U.S. NRC [2].

The basic concept of the standardized MPAS Level 2 PSA model is similar to the Level 2 PSA models used by domestic operators, such as the WH-type, Framatome-type, and APR1400-type models [3]. However, the Level 2 PSA model structures of the OPR1000-type are somewhat different from those of other types, particularly in terms of their Containment Event Tree (CET) and Decomposition Event Trees (DETs).

The objective of the study is to compare the operator model and the standardized and identify the major differences between the two models. The causes of the differences for each item will be explained, and sensitivity analysis will be conducted to demonstrate the validity and appropriateness of the standardized model. The comprehensive analysis of the differences between the two models aims to contribute to the development of a more accurate and reliable the standardized MPAS Level 2 PSA model.

2. Methods and Results

The Level 2 PSA process expands the core damage scenarios identified in the Event Tree (ET), which are defined as a result of the Level 1 PSA. This involves considering accident mitigation strategies and systems aimed at preventing containment failure as accident scenarios are expanded. The resulting expanded Event Tree is called the Plant Damage State Event Tree (PDSET). The PDSET's accident sequences are grouped into PDS with similar characteristics using the Plant Damage State Logic Diagram (PDSLD). The soundness of the containment building is evaluated by considering the characteristics of Plant Damage State (PDS) and the possibility of severe accident phenomena using the Containment Event Tree (CET) and Decomposition Event Trees (DETs).

The development of the CET and DETs may vary based on the developer's engineering judgment, such as reflecting the latest research results or simplifying/detailing uncertain severe accident models. This study aims to validate the standardized model by identifying major differences between the operator and standardized models, explaining the causes of these differences for each item, and conducting sensitivity analysis using the operator model's analysis method on each item.

To accomplish this, the study uses the same PDSET and conducts Level 2 PSA quantification using the standardized model's Plant Damage State Logic Diagram (PDSLD), CET, DETs, and Source Term Category Logic Diagram (STCLD).

2.1 Standardized Level 2 PSA model quantification

The PDSET determines the behavior of the mitigation system in the containment and the frequency of the accident scenario. The containment failure type and frequency are determined by evaluating the severe accident phenomenon and containment behavior using the PDSLD, CET, and DETs. To compare Level 2 PSA models, the same PDSET must be used, and the PDSLD, CET, and DETs can be modified and compared. Table I shows the results of the standardized MPAS model's quantification of OPR1000 Level 2 PSA.

Table I: Result of the MPAS Level 2 PSA Quantification

Failure Mode	Fraction (%)
NOCF	85.7
ECF	0.1
LCF	0.4
CFBRB	6.8
BMT	4.2
NOTISO	0.1
BYPASS	2.7

According to Table I, 85.7% of the cases analyzed maintained the containment building's integrity (NOCF). Among the cases analyzed, 6.8% resulted in damage to the containment building before the reactor vessel rupture (CFBRB). Containment base-mat melt through by core melt (BMT) was observed in 4.2% of the cases, and fission product release by bypassing containment (BYPASS) was observed in 2.7% of the cases. The percentage of cases with late containment building failure (LCF) was 0.4%, while early containment building failure (ECF) and containment isolation system failure (NOTISO) were both observed in 0.1% of the cases.

2.2 Compared to Level 2 PSA results of the operator model and the Standardized model

The table II shows comparison results obtained from the operator's Level 2 PSA model and the standardized MPAS Level 2 PSA model using the same PDSET. The change rate in percentage between the two models is also presented. It is appeared that the majority of the failure modes have a similar fraction, with the exception of ECF, LCF, and BMT.

Table II: Comparison of Level 2 PSA results

Failure Mode	Operator	Standardized	Change Rate
	Fraction (%)		(%)
NOCF	86.8	85.7	-1.3
ECF	0.6	0.1	-83.3
LCF	1.5	0.4	-73.3
CFBRB	6.8	6.8	0.0
BMT	2.0	4.2	110.0
NOTISO	0.1	0.1	0.0
BYPASS	2.3	2.7	18.4
CFF	13.2	14.3	8.6

The fraction of NOCF decreased slightly by 1.3% in the standardized model. The fractions of ECF and LCF decreased significantly by 83.3% and 73.3%, respectively, while the fraction of BMT increased by 110%. The fractions of CFBRB, NOTISO, and BYPASS showed relatively small differences, with the fraction of BYPASS increasing by 18.4%. The total fraction of cases with containment failure frequency (CFF) was 13.2% for the operator model and 14.3% for the standardized model, showing an 8.6% increase in the standardized model.

The significant differences in ECF, LCF, and BMT led to sensitivity analysis on the DETs related to these failure modes in the standardized model. The increased fraction of BYPASS in the standardized model was attributed to the latest study on thermal-induced steam generator tube rupture, which considered the effect of loop seal clearing for steam generator tubes in the BYPASS scenario [4].

2.3 Differences and Sensitivity analysis method

The first item is related to ECF, specifically the Alpha-mode Failure probability in the DET. The biggest difference between the model developed by the operator and the standardized model is that the operator model assumes a probability of 0.008 for Alpha-mode failure when the RCS pressure is low at the time of reactor vessel rupture, whereas the standardized model assumes a lower probability of 0.001 based on SERG-2. To perform sensitivity analysis, the Alpha-mode failure probability will be changed to 0.01 [5].

The first item is related to ECF, specifically the Alpha-mode Failure probability in the DET. The biggest difference between the operator model and the standardized model is that the operator model assumes a probability of 0.008 for Alpha-mode failure when the RCS pressure is low at the time of reactor vessel rupture, whereas the standardized model assumes a lower probability of 0.001 based on SERG-2. To perform sensitivity analysis, the Alpha-mode failure probability will be changed to 0.01 [5].

The second item pertains to LCF. The main difference between the operator model and the standardized model is that the operator model considers the possibility of containment failure during a reactor cavity dry and containment heat removal failure condition, while the standardized model assumes that LCF due to overpressure of the containment building will not occur even when containment heat removal fails with a dry Cavity. To perform the sensitivity analysis, a probability of 0.2 for containment failure due to overpressure when the Cavity dries and containment heat removal fails was assumed, as per the operator model.

The last item relates to BMT. The primary difference between the operator model and the standardized model in BMT DET is the level of phenomena considered when BMT occurs. The operator model comprehensively considers the pressure at the time of reactor vessel rupture, the amount of corium released outside cavity, the flooding condition of the cavity, and the shape of corium in the cavity as factors affecting the BMT probability. In contrast, the standardized model only considers the flooding condition of the cavity and the timing of the cavity flooding, as it is difficult to properly account for the uncertainty of each factor when considering too many factors. To perform the sensitivity analysis, the Reactor Vessel Rupture viewpoint was considered as an additional factor, with reference to the operator model. The detailed BMT occurrence probability for each case is presented in Table III.

Table III: BMT occurrence probability for Sensitivity III

RCS Press. At Vessel Rupture	Cavity Condition	BMT	Probability
High	Not Flooded	No	0.5
		Yes	0.5
	Flooded	No	1.0
		Yes	0.0
Medium	Not Flooded	No	0.1
		Yes	0.9
	Flooded	No	0.9
		Yes	0.1
Low	Not Flooded	No	0.0
		Yes	1.0
	Flooded	No	0.9
		Yes	0.1

2.4 Results of sensitivity analysis

Sensitivity analysis was conducted by modifying the corresponding DET in accordance with the sensitivity analysis method established for each individual sensitivity item. Table IV presents the results of the sensitivity analysis conducted for the ECF, LCF, and BMT sensitivity items.

Table IV: Sensitivity Analysis Results

Failure Mode	SEN1	SEN2	SEN3
	Fraction (%)		
NOCF	85.3	85.7	86.5
ECF	***0.5***	0.1	0.1
LCF	0.4	***1.2***	0.4
BMT	4.2	***3.5***	***3.4***
CFBRB	6.8	6.8	6.8
NOTISO	0.1	0.1	0.1
BYPASS	2.7	2.7	2.7

The analysis results for SEN1, which involved a sensitivity analysis reflecting the Alpha-mode failure probability of the Operator model for the ECF sensitivity item, showed that the fraction of ECF increased to 0.5%, which is similar to 0.6% of the Operator model.

The analysis results for SEN2, which conducted sensitivity analysis considering the LCF mechanism of the Operator model for the LCF sensitivity item, showed that the fraction of LCF increased to 1.2%, which is close to 1.5% of the Operator model. In addition, the analysis showed that the probability of occurrence of BMT was relatively reduced. This is due to the CET structure that considers the occurrence of BMT in a case where LCF is not occurred. Therefore, if the fraction of LCF increases, the fraction of BMT decreases relatively.

The analysis results for SEN3, which conducted sensitivity analysis considering the BMT occurrence conditions of the Operator model for the BMT sensitivity item, showed that the fraction of BMT decreased to 3.4%, which is somewhat different from the 2.0% of the Operator model. However, as described in the analysis results for SEN2, the frequency of occurrence of LCF is significantly lower than that of the Operator model, and therefore, much of it is classified as BMT. If sensitivity analysis is conducted by additionally reflecting the assumptions of SEN2, it is expected that the results will be closer to the results of the Operator model.

3. Conclusions

As the importance of safety regulation in nuclear power plants continues to grow, the regulatory agency is striving to establish a Risk-Informed Decision Making (RIDM) system through PSA. The standardized MPAS Level 2 PSA model has been developed to support RIDM. The development of this model was based on the APR1400 DC PSA model and considered the state-of-art studies and portable equipment application. However, the assumptions and configurations used in the model need to be reviewed to ensure their appropriateness.

This study aimed to validate the standardized MPAS Level 2 PSA model by comparing it to the Level 2 PSA model of the OPR1000 operator. To achieve this, major differences were identified, and sensitivity analysis was performed. The study derived three sensitivity analysis items related to ECF, LCF, and BMT, which were different between the two models.

After evaluation it was found that the significant differences were addressed by incorporating state-of-the-art studies or minimizing the analyst's judgment during uncertain conditions. Additionally, similar results were obtained when the assumptions of the Operator model were reflected in the analysis. These findings suggest that the standardized MPAS Level 2 PSA model, which was developed for regulatory testing, is reasonable and appropriate.

This work was supported by the Nuclear Safety Research Program through the Korea Foundation of Nuclear Safety (KoFONS) using the financial resource granted by the Nuclear Safety and Security Commission (NSSC) of the Republic of Korea. (No. 2101052)

REFERENCES

[1] H. Chang, G. Jung, J.H. Ryu, J. Hong, S. Lee, “Development of the Standardized Level 2 PSA Model Structure for the MPAS”, Transactions of the Korean Nuclear Society Autumn Meeting, Changwon, Korea, October 20-21, 2022.

[2] Korea Hydro & Nuclear Power Co., Ltd, Korea Electric Power, “Chapter 19 –Probabilistic Risk Assessment and Severe Accident Evaluation”, APR1400-K-X-FS-14002-NP, August 2018.

[3] G.Y Sung, J.S. Choi, D.S. Kim, D.H Kim, “Comparative study on the Level 2 PSA performed for the Surry plant, the ShinWolSong plant and the APR+ standard design”, Korea Institute of Nuclear Safety, KINS/RR-883, December 2011.

[4] U.S. Nuclear Regulatory Commission, Risk assessment of operational event. Handbook. Volume 5 Risk Analysis of Containment-Related Events (LERF), January 2018.

[5] Basu S and Ginsberg T, “Proceedings of the Second Steam Explosion Review Group (SERG-2) Workshop on A Reassessment of the Potential for an Alpha-Mode Containment Failure and a Review of the Current Understanding of Broader Fuel-Coolant Interaction Issues”, NUREG-1524, 1996.