Comparison and Validation of MPAS Level 2 PSA model with Operator's Level 2 PSA model: A Regulatory Perspective

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1. Introduction

Domestic nuclear power plant operators must develop and update their Probabilistic Safety Assessment (PSA) models and submit the results in fulfillment of the Nuclear Safety and Security Commission's (NSSC) regulatory requirements.

The Korea Institute of Nuclear Safety (KINS), Korea's regulatory body for nuclear power plants, is working to implement a Risk-Informed Regulation (RIR) system. KINS continuously develops and updates the Multi-purpose Probabilistic Analysis of Safety (MPAS) Level 1 PSA model to validate the risk assessment results submitted by operators for each nuclear power plant. However, to verify the Large Early Release Frequency (LERF) and the frequency of exceeding 100 TBq of Cs-137 release, which are regulatory risk requirements, Level 2 PSA models are also required.

As the need for developing a Level 2 PSA model for regulatory purposes, research began in 2021 to develop Level 2 PSA models for domestic nuclear power plants. By 2024, pilot MPAS Level 2 PSA models were completed for each reactor type operating in Korea.

This study was conducted to compare the quantification results of the MPAS Level 2 PSA model developed for the APR1400 with those of the operator's APR1400 Level 2 PSA model, and to assess the MPAS model's validity [1].

2. Methods and Results

The Level 2 PSA model for regulatory purpose, which began development in 2021, considered the following three key elements:

- Application of the Multi-barrier Accident Coping Strategy (MACST) included in the Accident Management Plan (AMP) submitted by domestic operators
- Application of severe accident progression analysis results from the SOARCA project conducted in the United States (e.g., NUREG-1935) [2].
- Review of domestic and international latest Level 2 PSA practices and modeling methods and integrate into MPAS model.

Based on these three elements, a standardized Level 2 PSA model was developed that can be commonly applied to pressurized water reactors [3]. The developed Standardized Level 2 PSA Model consists of the Plant Damage State Logic Diagram (PDSLD), Containment Event Tree (CET), Decomposition Event Tree (DET), and Source-Term Category Logic Diagram (STCLD).

To verify the suitability and validity of the Standardized Level 2 PSA Model, a pilot quantification was performed using the Standardized Level 2 PSA Model with the PDSET of the operator's OPR1000 model. This was followed by a comparison analysis with the operator's OPR1000 results [4].

The model underwent several updates following reviews of the NRC's Level 3 PRA Project and the UK-EPR PSA Report [5,6].

2.1 Development of APR1400 MPAS Level 2 PSA Model

Based on the Standardized Level 2 PSA Model and the MPAS Level 1 PSA model, MPAS Level 2 PSA models were developed to account for the design characteristics of each reactor type. For example, the APR1400 MPAS Level 2 PSA model includes features such as the Emergency Containment Spray Back-up System (ECSBS) for coping with the loss of containment heat removal, the Containment Flooding System (CFS) for reactor cavity flooding, and the 3-way valve for preventing hydrogen accumulation in the In-containment Refueling Water Storage Tank (IRWST). These features were incorporated into the standardized Level 2 PSA model by modifying its structure. Fig. 1 presents the revised APR1400 PDSLD accounting for ECSBS and CFS in the standardized model. Fig. 2 presents the revised APR1400 ECF DET considering the characteristics of the 3-way valve in the standardized model.

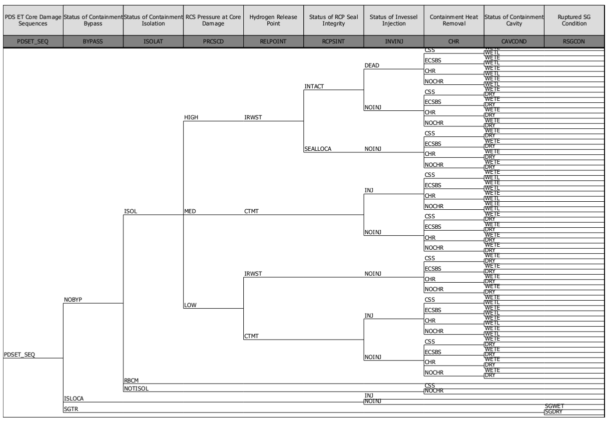

Fig. 1. MPAS APR1400 PDSLD

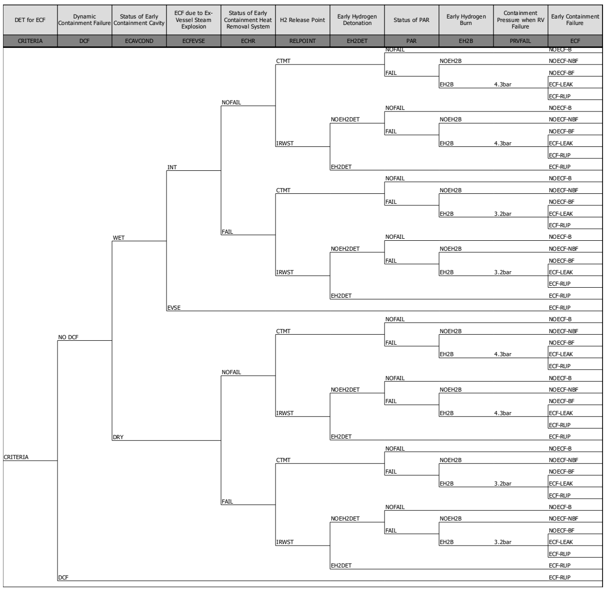

Fig. 2. MPAS APR1400 ECF DET

2.2 Quantification Results of APR1400 MPAS Level 2 PSA

The quantification results of the APR1400 MPAS Level 2 PSA are presented in Table I.

Table I: Quantification Result of APR1400 Level 2 PSA

Containment Failure Mode	Frequency (/ryr)	Contribution (%)
NOCF	~1.5E-06	73.2
ECF	~1.0E-08	0.5
LCF	~3.0E-07	16.2
BMT	~2.5E-10	0.0
CFBRB	~1.5E-07	7.6
NOTISO	~1.0E-08	0.4
BYPASS	~5.0E-08	2.0

According to Table I, 73.2% of the analyzed cases maintained the integrity of the containment building (NOCF). Among the containment failure modes, the contribution of cases with late containment building failure (LCF) was 16.2%, the containment building failure before the reactor vessel breach (CFBRB) was 0.6%, and fission product release bypassing containment (BYPASS) was 2.0%. Containment base-mat melt through (BMT), early containment building failure (ECF), and containment isolation system failure (NOTISO) were observed in less than 1%.

2.3 Comparison with the Operator's Model

To verify the suitability and validity of the APR1400 MPAS Level 2 PSA quantification results, a comparison was made with the quantification results of the APR1400 model submitted by the operator for U.S Nuclear Regulatory Commission (NRC)'s design certification (DC). The comparison results are presented in Table II.

Table II: Comparison of Quantification Result of APR1400 Level 2 PSA

Containment Failure Mode	MPAS (%)	NRC DC (%)
NOCF	73.2	83.0
ECF	0.5	0.1
LCF	16.2	2.4
BMT	0.02	1.9
CFBRB	7.6	1.5
NOTISO	0.4	0.3
BYPASS	2.0	10.9

The comparison of the two models' quantification results showed significant differences in three containment failure modes. The MPAS model showed contributions of approximately 16.2% for LCF and 7.6% for CFBRB, while the NRC DC model showed contributions of 2.4% and 1.5%, respectively. For BYPASS, the MPAS model's quantification result was approximately 2.0%, while the APR1400 NRC DC model was approximately 10.9%. The main reasons for these differences were:

- Differences in the modeling approach and reliability data for mobile equipment.
- Differences in considering additional mitigation measures when Induced Steam Generator Tube Rupture (SGTR) occurs.

LCF and CFBRB occur due to the failure of long-term containment heat removal as accidents progress over time. The MPAS model incorporating a long-term containment heat removal strategy in the PDSET by referencing the MACST. This strategy was incorporated into the PDSET because it allows for proper consideration of system dependencies using ET/FT modeling.

The reliability data for equipment used in the PDSET modeling were based on PWROG-18042-NP, which analyzes the operational experience of the Flexible Mitigation Equipment Strategy (FLEX) in U.S. nuclear power plants [7]. The key data from PWROG-18042-

NP accounting for in the MPAS Level 2 PSA model are presented in Table III.

Table III: Portable Equipment Reliability Data from the PWROG-18042-NP

Component	Failure Mode	Mean
Combustion Turbine Generator	Fail to Run	1.03E-02/d
	Fail to Start	4.35E-02/h
Diesel-Driven Pump	Fail to Run	1.55E-02/d
	Fail to Start	3.38E-02/h

The unavailability of the long-term containment heat removal using portable equipment, as modeled with reference to Table III and MACST, was found to be very high, exceeding 0.5. In contrast, the APR1400 NRC DC model was developed before MACST was established. Therefore, instead of incorporating a specific accident mitigation strategy with portable equipment, it considered an alternative containment heat removal function with a failure probability of 0.1 when installed equipment failed in the DET.

BYPASS, a containment damage mode, is primarily contributed by SGTR. SGTR can be categorized into SGTR initiating event and induced SGTR. Induced SGTR includes Pressure-Induced SGTR (PI-SGTR) due to pressure differences between the primary and secondary sides after an accident and Thermal-Induced SGTR (TI-SGTR) due to high pressure and high-temperature steam in the reactor cooling system after core damage.

The APR1400 NRC DC model classified as BYPASS when PI-SGTR occurs accompanied with failures of isolation an affected steam generator without considering additional mitigation measures. In contrast, the APR1400 MPAS Level 2 PSA model developed a PDSET that considers additional accident mitigation measures, such as depressurization of reactor cooling system, even if steam generator isolation fails during PI-SGTR

2.4 Sensitivity Analysis

To validate the main differences found in the comparison between the two models, sensitivity analyses were performed on the following two items

- Change in the unavailability of mobile equipment for long-term containment cooling strategies.
- Exception of accident mitigation measures when PI-SGTR with steam generator isolation fails.

When the unavailability of portable equipment was set to 0.1, like the APR1400 NRC DC model, and sensitivity analysis was performed, the contribution of LCF and CFBRB decreased to 3.5% and 0.6%, respectively, becoming similar to the results of the APR1400 NRC DC model.

When accident response strategies were not considered for PI-SGTR and steam generator isolation failure, leading to immediate classification as BYPASS accidents, the contribution of BYPASS increased to approximately 7.7%, similar to the APR1400 NRC DC model. However, a difference of about 3.2% still existed, this gap is due to differences in references for TI-SGTR. The NRC DC model mainly referred to NUREG-1570 to derive TI-SGTR probabilities in high-pressure scenarios, while the MPAS model referred to NUREG-2195 [8,9].

3. Conclusions

This study was conducted to compare the quantification results of the APR1400 MPAS Level 2 PSA model with the quantification results of the APR1400 Level 2 PSA model submitted by the operator and to verify the validity of the model.

The comparison analysis revealed notable differences between the APR1400 NRC DC Level 2 PSA model and the APR1400 MPAS Level 2 PSA model, primarily due to variations in accident mitigation strategies and reliability data. These differences are typically addressed during PDSET development and quantification, and may vary based on the analyst's judgment. The PDSLD, CET, DET, and STCLD used for evaluating containment failure modes in the MPAS and NRC DC models showed some differences but did not significantly affect the overall risk insights.

The most significant discrepancies between the operator's model and the MPAS model were observed in the containment failure modes LCF and CFBRB, primarily due to differences in portable equipment strategies and reliability data. The reliability data for portable equipment referenced in the MPAS model, based on PWROG-18042-NP, showed very low reliability. The document also noted that failure to run rates for FLEX equipment were calculated to be higher than expected, driven by a small number of failures with very low run hours (flex equipment is rarely run for more than an hour).

In conclusion, when the results of the PSA models used for regulatory verification purposes differ from those submitted by operators due to the use of uncertain reliability data, it is necessary to reassess with the alternative reliability data sources. On the other hand, if the detailed modeling results can provide better conclusions by reducing excessive conservatism in the analysis results, it may be possible to discuss the application of such detailed modeling method with the operators.

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REFERENCES

[1] 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..

[2] Richard Chang, Jason Schaperow, Tina Ghosh, "State-of-the-Art Reactor Consequence Analyses (SOARCA) Report", U.S. Nuclear Regulatory Commission, NUREG-1935, November 2012.

[3] 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.

[4] H. Chang, G. Jung, J.H. Ryu, J. Hong, "A Comparative Study of the Standardized MPAS Level 2 PSA Model and the Operator Model", Transactions of the Korean Nuclear Society Autumn Meeting, Jeju, Korea, May 18-19, 2023.

[5] U.S.NRC, "U.S. NRC Level 3 Probabilistic Risk Assessment (PRA) Project", U.S. Nuclear Regulatory Commission, Volume 3c: Reactor, At-Power, Level 2 PRA for Internal Events and Floods, April 2022.

[6] F. Godefroy, A. Marechal, G. Craig, "PCSR – Sub-chapter 15.4 – Level 2 PSA", AREVA NP & EDF, UKEPR-0002-154 Issue 06, November 2012.

[7] Matthew M. Degonish, "FLEX Equipment Data Collection and Analysis", Westinghouse Electric Company LLC, PA-RMSC-1651 R1, February 2022.

[8] SGTR Severe Accident Working Group, "Risk Assessment of Severe Accident-Induced Steam Generator Tube Rupture", U.S. Nuclear Regulatory Commission, NUREG-1570, March 1998.

[9] S. Sancaktar, M. Salay, R. Iyengar, "Consequential SGTR Analysis for Westinghouse and Combustion Engineering Plants with Thermally Treated Alloy 600 and 690 Steam Generator Tubes", U.S. Nuclear Regulatory Commission, NUREG-2195, May 2018.