Risk Profile Development for Accident Seqeunces with Source Term Analysis

Yunho Kim [a], Jaehyun Cho [a*]
[a]*Energy Systems Engineering, Chung-Ang University, 84 Heukseok-ro, Dongjak-gu, Seoul, Republic of Korea*
[*]*Corresponding author: jcho@cau.ac.kr*

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1. Introduction

Evaluating the risks associated with Nuclear Power Plant (NPP) accidents is important for the safe design and operation of NPP. A risk assessment of NPP includes a series of Probabilistic safety assessment (PSA) at three levels: level 1 PSA, level 2 PSA and level 3 PSA. Through this series of three-level PSA, the risk assessment of radioactive materials released into the environment is finally conducted.

This process has become a systematic methodology for evaluating the risks associated with NPP. However, recent research has been actively focused on developing reactors that differ from conventional large NPP, such as small modular reactors (SMRs) and Generation IV reactors [1]. Therefore, concerns about the effectiveness of applying conventional risk assessment methodology to these types of reactors is emerging. Moreover, conventional methodology group numerous accident scenarios into a few source term categories (STC). The grouping method makes it difficult to quantify the risk of each core damage scenario in the level 1 PSA event trees (ET).

In this study, we provide a new methodology to quantify risk of NPP accidents without performing the series of PSA at three levels. Therefore, this study aims to quantify the risk of core damage scenarios in the level 1 PSA ET by uncertainty analysis using severe accident analysis code and to develop risk profiles for each initiating event.

The contents of this paper are organized as follows. Section 2 describes the methodology and result. Section 3 presents the conclusion and discusses future work.

2. Method and Result

Fig. 1 illustrate the overall structure of new methodology to quantify risk of NPP accidents. There are four main steps in this overall structure as follows: 1) accident scenarios identification, 2) uncertainty analysis using severe accident analysis code, 3) consequence quantification, 4) risk quantification and developing risk profile.

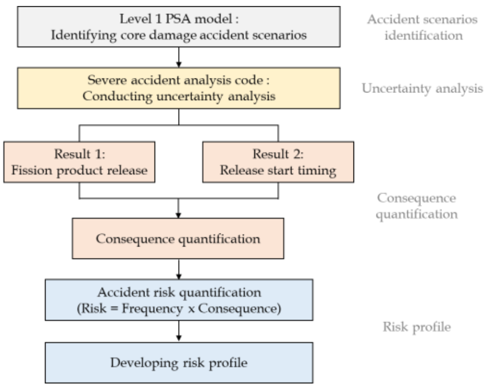
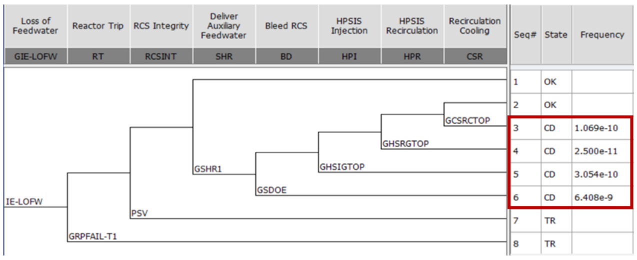


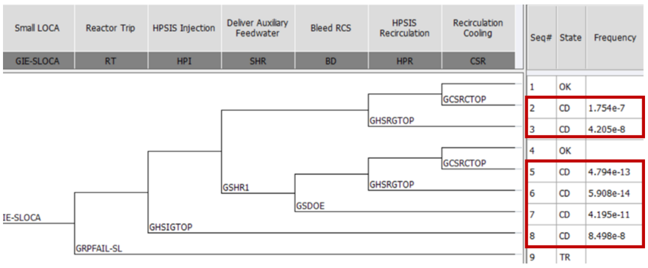
Fig. 1. Overall structure for accident scenarios risk quantification and development of risk profile

2.1 Accident scenarios identification

Based on the level 1 PSA model of OPR-1000, we analyzed loss of feedwater (LOFW) and small break loss of coolant accident (SBLOCA) as initiating events. Fig. 2 shows the LOFW and SBLOCA ET and core damage scenarios. There are four core damage scenarios in LOFW and six core damage scenarios in SBLOCA.



(a)LOFW

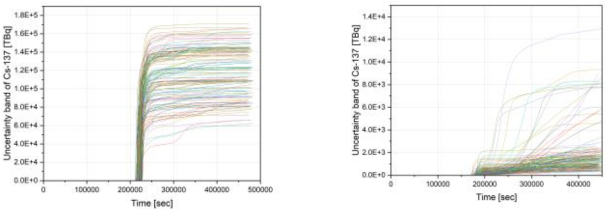


(b) SBLOCA

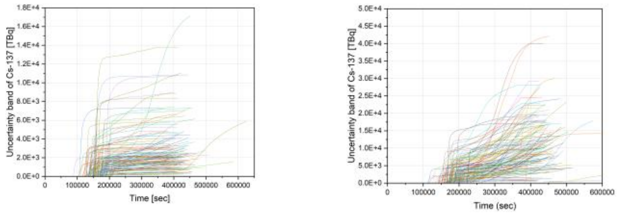
Fig. 2. OPR-1000 level 1 PSA ET of LOFW and SBLOCA

2.2 Uncertainty analysis

An uncertainty analysis of LOFW and SBLOCA accident scenarios was conducted. The reference code was MAAP5.05, developed by Fauske & Associates, LLC (FAI) and licensed by Electric Power Research Institute (EPRI) [3]. A total of 124 simulations were conducted for each 10 core damage scenarios. Fig. 2 and 3 show the uncertainty band of Cs-137 release for each scenario

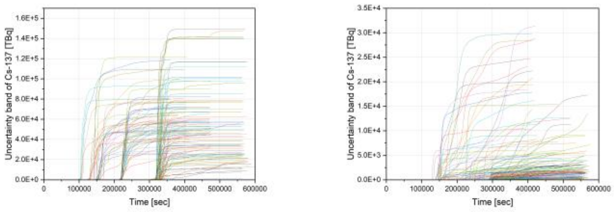


a) LOFW03 b) LOFW04

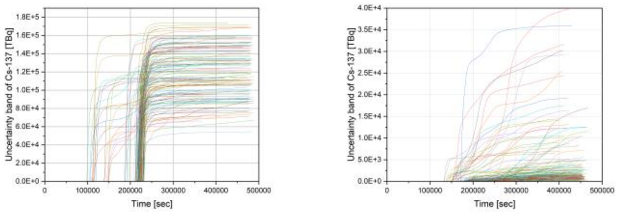


c) LOFW05 d) LOFW06

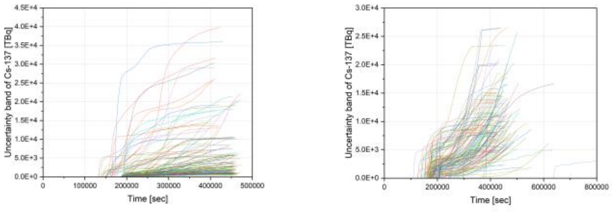
Fig. 2. Uncertainty band of Cs-137 release of LOFW core damage scenarios



a) SBLOCA02 b) SBLOCA03



c) SBLOCA05 d) SBLOCA06

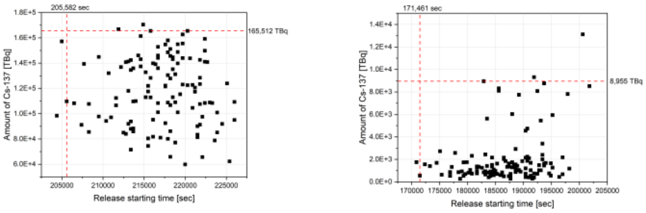


e) SBLOCA07 f) SBLOCA08

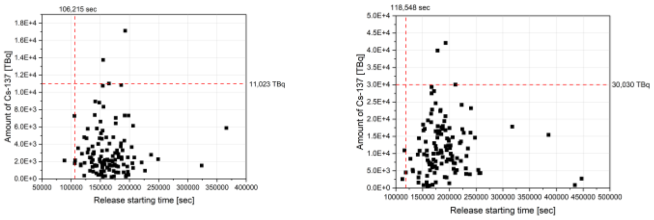
Fig. 3. Uncertainty band of Cs-137 release of SBLOCA core damage scenarios

2.3 Consequence quantification

In this study, to quantify the consequences of LOFW and SLOCA, two output values were used: 1) the amount of released Cs-137 (as representative fission product), 2) Cs-137 release timing [4]. As shown in Fig. 5 and 6, the 95% confidence points for the Cs-137 release and release starting time of LOFW and SBLOCA core damage scenarios are as follows: LOFW03 is 166,512 TBq and 205,582 s, LOFW04 is 8,955 TBq and 171,461 s, LOFW05 is 11,023 TBq and 106,215 s, LOFW06 is 30,060 TBq and 118,548 s, SBLOCA02 is 141,448 TBq and 100,849 s, SBLOCA03 is 26,787 TBq and 138,968 s, SBLOCA05 is 169,165 TBq and 103,852 s, SBLOCA06 is 31,651 TBq and 138,421 s, SBLOCA07 is 31,651 TBq and 138,334 s, SBLOCA08 is 35,534 TBq and 126,872 s.

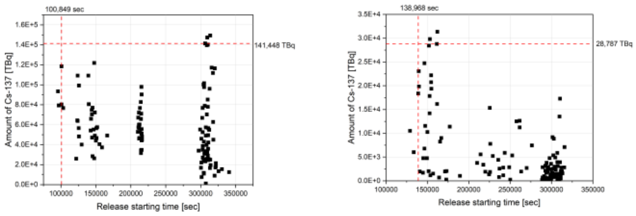


a) LOFW03 b) LOFW04

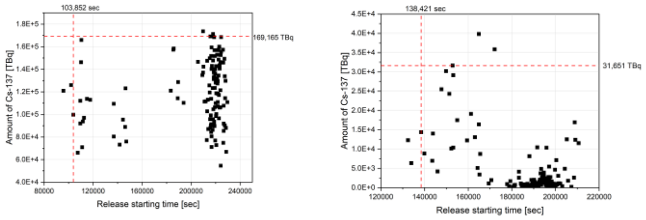


c) LOFW05 d) LOFW06

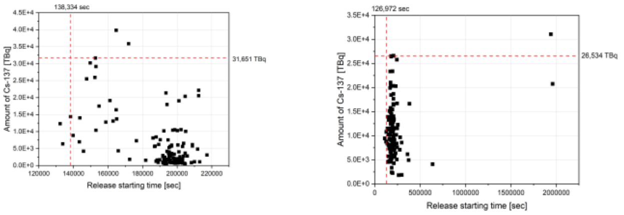
Fig. 4. Cs-137 release and release timing 95% confidential estimation of LOFW



a) SBLOCA02 b) SBLOCA03



c) SBLOCA05 d) SBLOCA06



e) SBLOCA07 f) SBLOCA08

Fig. 5. Cs-137 release and release timing 95% confidential estimation of SLOCA

The consequence of each scenario will be proportional to the amount of the fission product release and inversely proportional to release starting time. Therefore, we developed an equation to quantify the consequences of each simulation, which is expressed as

$$C_k = \frac{{m_k}^2}{t_k} \tag{1}$$

Where C_k is a consequence of k-th iteration of simulations, m_k is the amount of Cs-137 release to the environment of k-th iteration, t_k is the release starting time of k-th iteration. Fig. 5 shows the range of the calculated consequence values for each scenario.

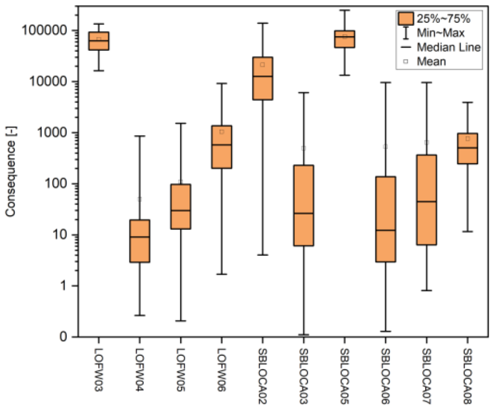


Fig. 5. Consequence of each scenario

2.3 Risk quantification

Risk is expressed by multiplication of frequency and consequences [5] as the following equation.

$$Risk = frequency \times consequence \tag{2}$$

Fig. 6 shows the risk obtained by multiplication of the normalized frequency of each scenario and consequences of each simulation. Fig 7 shows the nomalized frequency of each scenario. Fig. 8 shows the 95%/95% estimation of the confidential risk by 3rd order Wilks' method.

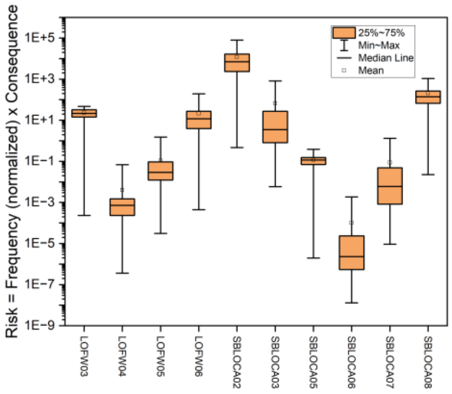


Fig. 6. Risk of each scenario

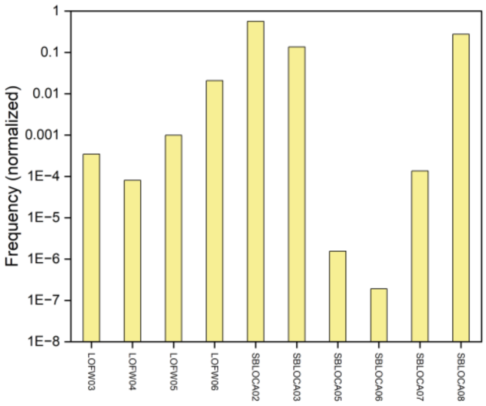


Fig. 7. Normalized frequency of each scenario

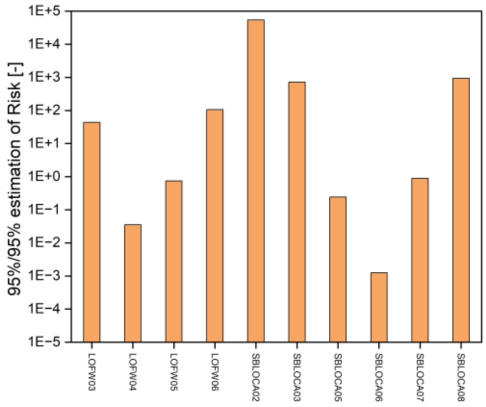


Fig. 8. 95%/95% estimation of the confidential risk of each scenario

3. Conclusions

This study quantified the risk of core damage scenarios based on a level 1 PSA model through uncertainty analysis using MAAP 5.05. We proposed a new methodology that can identify the risk of internal events without conducting level 2 PSA and level 3 PSA.

However, the method for calculating the consequence is not precise, and level 1 PSA ET only includes the branches related to the core damage accident sequence. Therefore, future work will include level 3 PSA results and additional branches at ET to calculate the consequences more accurately.

4. Acknowledgment

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REFERENCES

[1] Subki, Hadid. "Advances in small modular reactor technology developments." (2020).
[2] Cho, Jaehyun, et al. "Exhaustive simulation approach for severe accident risk in nuclear power plants: OPR-1000 full-power internal events." *Reliability Engineering & System Safety* 225 (2022): 108580.
[3] Kindred, T. "Modular Accident Analysis Program 5 (MAAP5) Applications Guidance: Desktop Reference for Using MAAP5 Software–Phase 3 Report." Electric Power Research Institute (2017).
[4] Cho, Jaehyun, and Sang Hoon Han. "Identification of risk-significant components in nuclear power plants to reduce Cs-137 radioactive risk." *Reliability Engineering & System Safety* 211 (2021): 107613.
[5] Oh, Kyemin, et al. "Study on multi-unit level 3 PSA to understand a characteristics of risk in a multi-unit context." *Nuclear Engineering and Technology* 52.5 (2020): 975-983.