Development of Risk Profile for Accident Sequences with Source Term Analysis

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EXTENDED ABSTRACT

As nuclear energy systems evolve toward advanced designs such as Small Modular Reactors (SMRs) and Generation IV reactors, the need for risk assessment methodologies that are both efficient and universally applicable becomes increasingly urgent. Traditional Probabilistic Safety Assessment (PSA) frameworks divided into Level 1, 2, and 3 remain fundamental for evaluating reactor safety, although they face certain challenges in broader applicability, efficiency and presence of uncertainty due to lack of data. These include high computational cost, inflexibility for early design-phase applications, and difficulty in encompassing diverse accident scenarios or new reactor configurations. Risk is generally quantified as a function of accident frequency and consequence. While frequency estimation is well-established within PSA, the quantification of accident consequences tends to receive comparatively less emphasis. The MELCOR Accident Consequence Code System (MACCS), developed by the U.S. NRC, is a widely used tool for Level 3 PSA consequence analysis. Level 3 PSA typically involves detailed modeling of site-specific parameters such as meteorology, emergency response, and population distribution, which can make its application resource-intensive in early-stage design assessments or generic reactor comparisons [1]. To address these challenges, this study proposes a novel methodology that develops risk profiles by correlating source term characteristics (e.g., fission product release amount and timing) directly with accident consequences. This correlation-based approach enables efficient risk quantification without requiring full Level 3 PSA. By bypassing detailed consequence modeling, the method enhances general applicability, reduces computational overhead, and offers intuitive insight into the relative risk of accident sequences. In this study, for simplification, only Cs-137, which is the most representative radionuclide among many fission products, was considered. A correlation-based equation model, as shown in Eq. (1), was developed, assuming that the accident consequence is proportional to the release amount and inversely proportional to the release start time.

$$C = k_{env} \cdot \left(k_q \cdot Q + \frac{k_t}{{t_{release}}^{a}} \right) \tag{1}$$

where C is the accident consequence, n is the number of analyzed fission products (FPs), Q is the Cs-137 release amount [TBq], $t_{release}$ is the start time of Cs-137 release [hr], and k_{env} is an environmental adjustment coefficient [hr/TBq]. As for the other variables, k_q is the relative influence of Q the accident consequence C, k_t is the relative influence of t on C, and a is an exponent adjusting the effect of time t.

Eq. (2) was derived through regression analysis using data from the State-of-the-Art Reactor Consequence Analyses (SOARCA) report published by U.S. Nuclear Regulatory Commission (NRC). In this equation, denotes the scenario-specific individual latent cancer fatality (LCF) probability at 10 miles reported in NUREG/CR-7110 [2,3]. Table 1 shows absolute and relative errors between simulation value of consequence in SOARCA and predicted C.

$$C = k_{env} \cdot \left(0.79 \times 10^{-8} \cdot Q + \frac{3.55 \times 10^{-4}}{{t_{release}}^{0.655}} \right) \tag{2}$$

Table I. Absolute and relative errors between simulation value and predicted value of the consequences

Nuclear power plant	Severe accident Scenario	Observed C	Predicted C	Absolute error	Relative error (%)
Surry	1	4.7e-05	5.307e-05	6.070e-06	12.91

	2	9.4e-05	8.222e-05	1.178e-05	12.53
	3	3.2e-04	3.003e-04	1.970e-05	6.16
	4	3.0e-04	3.016e-04	1.600e-06	0.53
	5	2.8e-04	3.003e-04	2.030e-05	7.25
Peach Bottom	6	8.9e-05	8.346e-05	5.540e-06	6.22
	7	7.1e-05	8.928e-05	1.828e-05	25.75
	8	2.1e-04	2.043e-04	5.700e-06	2.71

The study proposes a methodology for developing reactor risk profiles based on the above correlation model. As a case study, the framework was applied to the Korean OPR-1000 nuclear power plant, focusing on two initiating events: Loss of Feedwater (LOFW) and Small Break Loss of Coolant Accident (SLOCA). Each accident sequence was identified from the Level 1 PSA model, and corresponding source terms were generated using the MAAP 5.05 code [4]. The uncertainty in source term release was evaluated through Latin Hypercube Sampling of 46 parameters, generating 124 cases per sequence. To estimate accident consequences, a simplified correlation model derived from regression on SOARCA data was developed using the release amount and start time of Cs-137 as predictors of latent cancer fatality probability. Risk profiles were then visualized as Frequency-Consequence (F-C) curves, allowing for intuitive comparison of accident scenarios as shown in Fig 1. The results showed that SLOCA has a substantially higher risk approximately 50 times greater than LOFW at the 95th percentile.

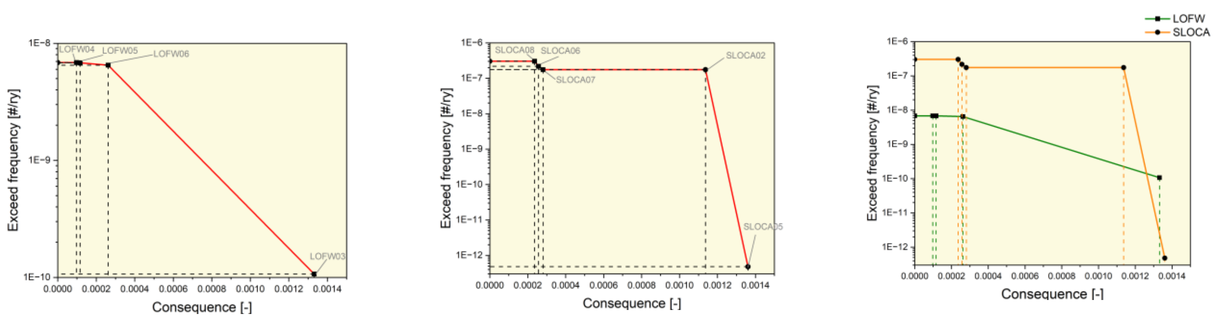

FIGURE 1. Comparison of the 95th percentile F-C curves for LOFW and SLOCA

This study highlights the potential of a source term-based consequence estimation model in enabling faster, more scalable, and broadly applicable risk assessments. The proposed methodology supports risk informed decision-making across design, licensing, and emergency planning stages, especially in the context of emerging reactor technologies [5].

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