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Development of Effective-Dose-Based Risk Profiles for Exhaustive Scenarios in Nuclear Power Plants

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Conventional probabilistic safety assessment (PSA) has primarily focused on frequency-based quantitative goals (e.g., CDF, LERF) and relied on a limited number of representative scenarios for consequence assessment. However, recent decision-making processes based on the Risk-Informed Performance-Based (RIPB) regulation, including design, licensing, and emergency planning zone (EPZ) determination, require comprehensive risk quantification that integrates both the frequency and consequences of all identifiable severe accident scenarios. Therefore, this study presents a methodology for developing a quantitative risk profile based on effective dose, comprehensively considering both frequency and consequences for exhaustive severe accident scenarios. In this study, risk profiles were constructed by combining the accident frequency and total effective dose equivalent (TEDE) of each scenario, thereby visualizing the risk distribution of the nuclear power plant. Furthermore, this study analyzed a risk contribution by source term categories (STCs) to quantitatively identify the dominant STCs driving the total risk. Moreover, the derived total risk was compared against quantitative health objectives (QHO). By comprehensively evaluating both accident frequency and consequences, the exhaustive scenario-based risk profiling framework proposed in this study can be utilized not only for the safety assessment of SMRs and advanced reactors but also for the implementation of RIPB regulations.

Keywords: Risk profile, exhaustive scenario, source term, total effective dose equivalent, frequency-consequence, OPR1000.

1. Introduction

Risk is a measure for evaluating the safety of nuclear facilities and is defined as the product of accident frequency and its consequence. In other words, ensuring safety from a risk perspective requires not only reducing the likelihood of accident occurrence but also minimizing the potential consequences. This probabilistic approach was formally introduced following the publication of the WASH-1400 report by the U.S. NRC in 1975 (U.S. NRC, 1975). As the first attempt to quantitatively analyze nuclear power plant accidents, this report laid the foundation for probabilistic safety assessment (PSA), which complements existing deterministic approaches in nuclear safety regulation.

Currently, the risk of nuclear power plants is evaluated through a three-level PSA framework. Level 1 evaluates the Core Damage Frequency (CDF); Level 2 assesses containment integrity and source term release, including Large Early Release Frequency (LERF); and Level 3 evaluates off-site radiological impacts on the public and environment. Despite this structure, the practical application of PSA has primarily focused on demonstrating compliance with frequency-based regulatory goals, such as maintaining the CDF and LERF below defined thresholds. Consequently, comprehensive assessment using Level 3 PSA has remained relatively limited.

Recently, the Risk-Informed Performance-Based (RIPB) regulatory framework has been introduced for the design and regulation of advanced reactors, such as Small Modular Reactors (SMRs) (Jang & Shim, 2020). This approach requires a comprehensive evaluation of system safety by considering both the frequency and consequences of all potential accident scenarios. In other words, it is essential to establish a comprehensive risk profile based on exhaustive scenarios. However, conventional risk assessment systems typically perform consequence analysis on only a small number of representative scenarios, failing to encompass the full accident spectrum.

To overcome these limitations of conventional PSA, this study proposes a methodology for developing a risk profile based on exhaustive scenarios. Specifically, this study aims to establish a comprehensive risk profile development framework by individually calculating the Total Effective Dose Equivalent (TEDE) for all identifiable accident scenarios in Level 2 PSA. This enables the development of a comprehensive risk profile that extends the scope of the RIPB approach beyond SMRs to include existing large-scale nuclear power plants.

The remainder of this paper is organized as follows: Chapter 2 provides an overview of source term analysis and offsite consequence analysis on the reference reactor. Chapter 3 describes the overall research methodology for the exhaustive effective-dose-based risk profile. Chapter 4 presents the results of the exhaustive Level 2 PSA accident scenario analysis, MAAP/MACCS-based TEDE calculations, and the derivation of risk profiles on the OPR1000 reactor as a case study. Finally, Chapter 5 provides the conclusions of this study.

2. Overview of Source Term Analysis and Off-site Consequence Assessment

As mentioned in the previous section, the RIPB approach requires comprehensive risk quantification that evaluates both the frequency and consequences of all potential accident scenarios. To apply this philosophy to large commercial reactors, this study aims to develop a risk profile based on exhaustive scenarios using the OPR1000 as a reference reactor. Therefore, this section describes the methodologies for source term analysis and off-site consequence assessment employed to construct such a risk profile.

2.1. *Analysis of Source Term Releases*

To evaluate the consequences of severe accidents, it is essential to analyze the released radiological source terms. Accordingly, this study used the Modular Accident Analysis Program (MAAP) 5.05 code for source term analysis. MAAP 5.05, developed by Fauske & Associates, LLC (FAI) and licensed by the Electric Power Research Institute (EPRI), is an integrated severe accident analysis tool widely employed to simulate severe accident progression and source term release (Fauske & Associates LLC (FAI), 2019).

The OPR1000 was selected as the reference nuclear power plant and was modeled using the MAAP 5.05 code. Fig. 1 presents the nodalization scheme applied in this study, showing the two-loop Reactor Coolant System (RCS) and Reactor Pressure Vessel (RPV) configuration (Ahn et al., 2020; Ahn et al., 2021). Through the source term analysis, key characteristics of radioactive release, such as release timing, release energy, and release fractions of radionuclide groups, were quantified. These data serve as essential input parameters for the subsequent consequence assessment using the MELCOR Accident Consequence Code System (MACCS) (Jow et al., 1990).

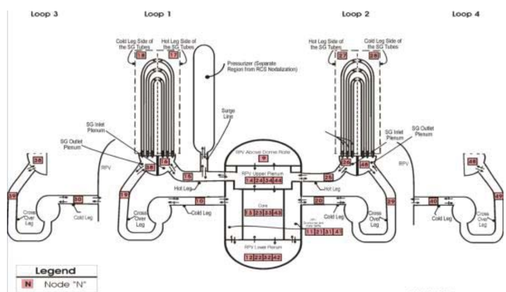


Fig. 1. Nodalization of two RCS loops and the RPV of the OPR1000.

2.2. *Off-site Consequence Assessment*

To quantitatively evaluate off-site radiological consequences, this study employed MACCS Version 2, developed by Sandia National Laboratories. As shown in the MACCS computational flow diagram in Fig. 2, the analysis involves atmospheric dispersion simulation, emergency response modeling, and dose calculations incorporating site-specific data. The key modeling inputs are summarized as follows: 1) the ATMOS module, which simulates the transport and deposition of radioactive materials using a Gaussian plume model based on a full year of hourly meteorological data (8,760 hours) and a standardized spatial grid; 2) the EARLY module, which quantifies the radiological impact during the emergency phase by calculating the Total Effective Dose Equivalent (TEDE) for two distinct groups, Cohort 1 (99.5% evacuating population) and Cohort 2 (0.5% non-evacuating population); and 3) the selection of consequence metrics, which focuses on the TEDE values of Cohort 2 to represent the unmitigated radiological consequences intrinsic to the source term as the basis for the final risk profile.

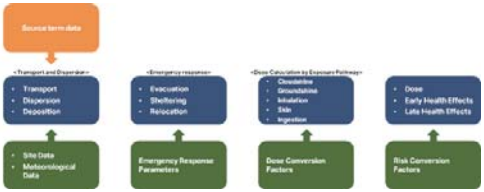


Fig. 2. MACCS computational flow diagram.

3. Research Methodology

This chapter describes the research methodology developed to establish exhaustive effective-dose-based risk profiles for a nuclear power plant.

3.1. *Framework for Developing Exhaustive Effective Dose-based Risk Profiles*

This study establishes a methodological framework to evaluate nuclear power plant risk by quantitatively integrating the frequency and consequences of all severe accident scenarios. As shown in Fig. 3, the framework follows a systematic procedure consisting of four major steps to construct a consolidated Frequency-Consequence (F-C) risk profile.

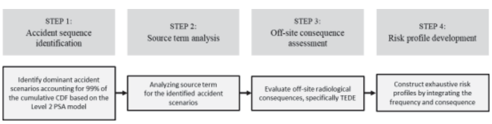


Fig. 3. Framework for the development of exhaustive effective dose-based risk profiles.

3.1.1. *Accident Scenario Identification*

The first step involves identifying accident scenarios based on the Level 2 PSA model of the reference plant. To ensure that the analysis captures the vast majority of potential risks as required by the RIPB approach, scenarios representing 99% of the cumulative CDF are selected. This approach follows the established methodology for comprehensive risk assessment previously developed (Cho et al., 2021). For the OPR1000, 690 scenarios were identified as accounting for 99% of the cumulative CDF. This means that by focusing on these 690 scenarios, nearly all of the risk can be effectively captured and evaluated, despite the existence of tens of thousands of potential sequences in a full Level 2 PSA. This selection process enables a comprehensive risk quantification that encompasses all relevant accident sequences, providing a robust dataset for subsequent analysis.

3.1.2. *Source Term Analysis*

For each identified scenario, detailed source term characteristics, including release timing, duration, and radionuclide release fractions, are evaluated using the MAAP code, and the resulting data are used as input for subsequent consequence assessment.

Since this study involves a large number of scenarios, automated analysis systems are essential to ensure efficiency and consistency. The system coordinates multiple code executions and automatically converts the resulting data into the source term input format required by off-site consequence analysis codes. In this study, automation tools developed by the Korea Atomic Energy Research Institute (KAERI), including the Multi-Unit Source Term Converter (MUST Converter) and Multi-run Manager, were employed to interface MAAP results with the MACCS code. The MUST Converter transforms MAAP output files into the MACCS source term input format (CombineSource.out), while the Multi-run Manager executes multiple MACCS calculations in parallel using the generated input files. Fig. 4 illustrates the MAAP–MACCS large-scale analysis framework adopted in this study. These automation tools significantly enhance the efficiency of large-scale PSA-based risk assessments.

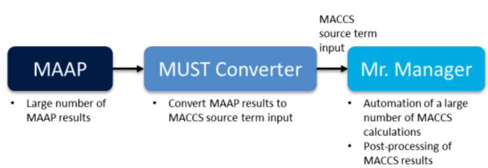


Fig. 4. MAAP–MACCS integration framework with automated tools.

3.1.3. *Accident Consequences Quantification*

Following the source term analysis, the radiological consequences for each individual scenario are quantified using the MACCS code. Unlike traditional PSA methods that rely on a small set of representative scenarios, this framework individually calculates the TEDE for all selected sequences. This exhaustive approach ensures that the specific release characteristics of each scenario are directly reflected in the consequence results. The calculated TEDE values serve as the primary consequence metric, which is then integrated with frequency data in the final risk quantification step.

3.1.4. *Development of the Risk Profiles*

The final step is to construct the risk profile by combining the accident frequencies (F_i) derived in the identification phase with the corresponding radiological consequences (C_i) calculated in the previous step. Specifically, the risk of an individual accident scenario ($Risk_i$) is defined as the product of its frequency and consequence, as expressed in Eq. (1):

$$Risk_i = C_i \times F_i \tag{1}$$

To evaluate the overall risk profile of the reference nuclear power plant, the total risk ($Risk_{Total}$) is determined by summing the individual risks of all scenarios (N=690), as shown in Eq. (2):

$$Risk_{Total} = \sum_{i=1}^{N} Risk_i \tag{2}$$

Consequently, risk is quantified and visualized via distance-dependent risk curves, thereby constructing the final risk profile for the nuclear power plant.

4. Development of Exhaustive Effective Dose-based Risk Profiles

This section presents the results of applying the quantitative risk assessment framework, described in Section 3, to the OPR1000 reactor as a case study. The assessment was performed using the OPR1000 Level 2 PSA model for full-power internal events, developed by KAERI (KAERI, 2015). The following subsections sequentially present the detailed analysis results corresponding to the four-step framework illustrated in Fig. 3.

4.1. *Identification of Accident Scenarios*

The target scenarios for analysis were selected based on the OPR1000 Level 2 PSA model. This model incorporates 39 Plant Damage States (PDSs), 100 Containment Event Trees (CETs), and 17 Source Term Categories (STCs). While a full risk profile ideally requires the analysis of every scenario identified at Level 2, the total number of accident sequences exceeds 9,900 when a cut-off frequency of $1.0 \times 10^{-14}/ry$ is applied. Analyzing this volume of scenarios is computationally impractical. Therefore, following the methodology proposed by Cho et al. (2022). this study selected the top 690 dominant scenarios. which cumulatively account for 99% of the total CDF.

These scenarios were grouped into 17 STCs based on similarities in containment failure modes and release characteristics. This categorization implies that each STC represents a unique set of radiological release parameters. By developing a risk profile based on these exhaustive scenarios, risk can be quantitatively evaluated by integrating both frequency and consequences. This approach facilitates the analysis of risk contributions from each STC to the overall plant risk, allowing regulatory resources to be efficiently concentrated on risk-dominant areas. Table 1 defines each STC and its respective frequency fraction.

Table 1. STCs in OPR1000 Level 2 PSA: Associated containment failure modes and frequency fractions.

STC #	Containment Failure Mode	Fraction
1	NOCF RV intact	11.1%
2	NOCF RV rupture	48.3%
3	ECF Leak CS-YES	0.4%
4	ECF Leak CS-NO	0.3%
5	ECF Rupture CS-YES	0.6%
6	ECF Rupture CS-NO	0.4%
7	LCF Leak CS-NO	4.1%
8	LCF Leak CS-NO	4.2%
9	LCF Rupture CS-NO	4.5%
10	LCF Rupture CS-NO	4.4%
11	BMT	0.6%
12	ECF In-vessel steam explosion	0.2%
13	CFBRB	12.5%
14	NOISO CS-YES	0.1%
15	NOISO CS-NO	0.0%
16	BYPASS ISLOCA	0.3%
17	BYPASS SGTR	8.1%

NOCF: No containment failure; ECF: early CF; LCF: late CF; BMT: basement melt-through; CFBRB: containment failure before reactor vessel breach; NOISO: no isolation; BYPASS: bypass failure

4.2. *Source Term Analysis using MAAP 5.05 Code*

Source term analysis was performed for each of the 690 scenarios using the MAAP 5.05 code. This analysis produced the necessary source term data for off-site consequence assessments, including release magnitudes and timings. To resolve compatibility issues between the MAAP output and the MACCS input formats, the integrated analysis framework described in section 3.1.2 was employed.

The simulations were conducted for up to 72 hours following the initial release. For scenarios with an intact containment, the design leakage rate of 0.1% vol/day was assumed. Cs-137 was used as the key indicator for radiological release. The results confirm that these characteristics vary significantly depending on the containment failure mode (e.g., NOCF, ECF, LCF, or BYPASS) and the operability of the containment spray system (CS–YES/NO). This variance highlights the importance of accounting for specific STC characteristics when evaluating the effectiveness of protective actions.

4.3. *Consequence Assessment using the MACCS Code*

Off-site consequences were assessed using the MACCS code, following the methodology described in Section 2.3. To evaluate the inherent consequences associated with source term characteristics, independent of emergency response effectiveness, the analysis focused on the non-evacuating cohort (Cohort 2).

As a representative example, Fig. 5 presents the TEDE as a function of distance for the TLOCCW4-CET14 scenario. The results show a typical decreasing dose trend as the distance from the release point increases. This calculation procedure was systematically applied to all 690 identified accident scenarios to quantify TEDE across all accidents.

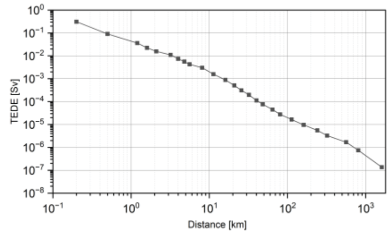


Fig. 5. TEDE for the TLOCCW4-CET14 across the distance

4.4. *Derivation and analysis of risk profiles*

In this section, a quantitative risk assessment was performed for the 690 severe accident scenarios of the OPR1000 by integrating the off-site consequence results with their respective accident frequencies. As defined in the methodology, the risk for each scenario was calculated by multiplying the TEDE [Sv] at each distance by its corresponding accident frequency [/ry]. The resulting risk unit, [Sv/ry], represents the expected effective dose per reactor-year attributable to potential severe accidents.

Fig. 6 presents the distance-dependent risk profile derived from the exhaustive set of scenarios. Unlike traditional approaches that rely on a limited number of representative cases, this exhaustive profile enables a comprehensive visualization of the total radiological risk across the entire accident spectrum. This distance-based risk information serves as a valuable technical basis for optimizing Emergency Planning Zones (EPZs), identifying risk-dominant sequences, and guiding risk-informed decision-making regarding design enhancements.

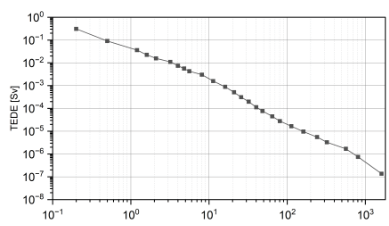


Fig. 6. Risk profile of all accident scenarios.

To further identify the primary drivers of the total plant risk, a contribution analysis was conducted for each STC. As shown in Fig. 7, which compares the individual risk profiles of the 17 STC groups against the total risk, the overall risk is primarily dominated by STC 17 (SGTR) and STC 13 (CFBRB). A more detailed evaluation using Pareto charts (Fig. 8) at representative distances reveals a significant shift in risk dominance as a function of distance. In the near-field region (to 4.8 km), STC 13 and STC 17 together account for over 90% of the total risk. However, as the distance increases to the mid- and far-field (16.1 km and 80.5 km), the contribution of STC 13 becomes increasingly dominant, exceeding 70%, while the influence of STC 17 declines significantly. This quantitative analysis confirms that the plant's total risk is governed by specific dominant STCs, providing a clear justification for prioritizing regulatory resources and enhancing emergency preparedness planning for these critical accident sequences.

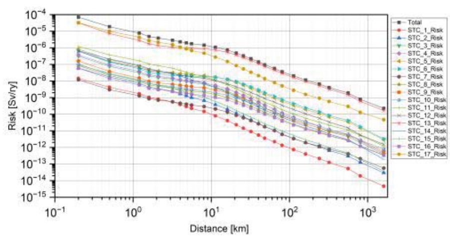
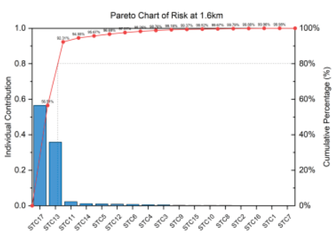
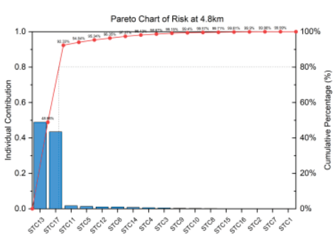


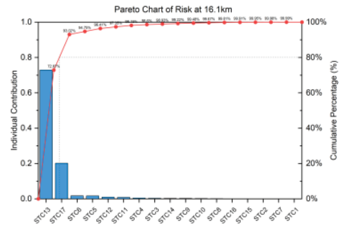
Fig. 7. Risk profiles of 17 STC groups and total radiation risk.



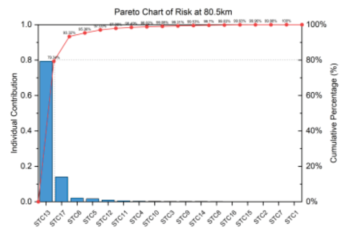
(a) 1.6 km



(b) 4.8 km



(c) 16.1 km



(d) 80.5 km

Fig. 8. Pareto charts of STC risk contribution at representative distances.

5. Conclusion

This study proposed a methodology for developing an exhaustive, effective-dose-based risk profile aligned with the RIPB regulatory requirements. By using the OPR1000 as a reference plant, a comprehensive risk profile was constructed from 690 scenarios derived from the Level 2 PSA, which together represent 99% of the cumulative CDF. The major findings are as follows: First, by integrating the MAAP 5.05 and MACCS codes, this study visualized a granular risk distribution that systematically combines frequency and radiological consequences. Second, the contribution analysis identified SGTR (STC 17) and CFBRB (STC 13) as the primary drivers of the total risk.

In conclusion, this study establishes a robust framework for generating comprehensive risk information essential for the implementation of RIPB regulations in both existing and advanced reactors, such as SMRs. Future research will focus on developing concrete strategies to integrate these exhaustive risk profiles into actual regulatory frameworks and extending the methodology to diverse reactor types.

Acknowledgement

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