Development of Risk Importance Measures for Dynamic PRA Based on Risk Triplet (2) Trial Measurement of Risk Importance Through Dynamic Level 2 PRA With RAPID

Xiaoyu Zheng[a]*, Hitoshi Tamaki[a], Yasuteru Sibamoto[a], Yu Maruyama[a], Tsuyoshi Takada[a], Takafumi Narukawa[b], Takashi Takata[b]
[a]Japan Aotmic Energy Agency (JAEA), Tokai-mura, Ibaraki, Japan
[b]The University of Tokyo, Tokyo, Japan
*Contact Author. E-mail address: zheng.xiaoyu@jaea.go.jp

Abstract: Traditional frequency/probability-based risk importance measures (RIMs) have demonstrated its practicability in the nuclear regulation. In the paper, the authors investigate the definitions of existing RIMs and associated applications in risk-informed nuclear regulations, for instance, the risk-informed categorization of structures, systems, and components (SSCs), risk-informed changes to technical specifications, application of maintenance rule, the reactor oversight process, and licensing application activities of small modular reactors. However, when evaluating mitigation effects of accident countermeasures, importance assessments involving consequence and timing has the potential of providing valuable information for decision making. Advanced probabilistic risk assessment (PRA) is required to obtain such detailed information. By widely using numerical simulations of possible accident progressions, dynamic PRA enables a straightforward assessment of risk triplets including scenario, frequency, and consequence, $R = \langle S_i, P_i, C_i \rangle, i = 1,2,\cdots,N$. Recent advancements in the development of dynamic PRA tend to explicitly incorporate the dynamics of accident progression and failure events into risk assessment, and it allows a provision of more detailed risk information. On the other hand, the approach to appropriate estimation of risk importance within this framework has not been established, exposing a significant research challenge in the use of risk information for decision making in the nuclear industry. In the first of the two consecutive presentations, the authors proposed a new RIMs based on the risk-triplet concept, written as RIM= ⟨TBW, FBW, CBW⟩, consisting of timing-based worth (TBW), frequency-based worth (FBW), and consequence-based worth (CBW). To demonstrate the calculability and variety of risk information that the new RIM can provide, the severe accident code of MELCOR and the JAEA's dynamic PRA tool of RAPID are coupled to quantify the risk triplets for a dynamic Level 2 PRA. Possible accident sequences are sampled using RAPID by randomly branching, and risk triplets are quantified, including key quantities such as source term release amount and release timing to the environment, and the associated frequencies. Risk triplets are used to calculate the new RIMs to rank the importance of pivotal headings in the event tree model. As the exemplary results of the analysis, source term release amount and timing are largely influenced by the mode of containment failure and the termination timing of reactor coolant injection. As the conclusion, when issues such as timing or seriousness of consequence are important for judgement, dynamic PRA and the new RIMs is capable of supporting decision making by providing more detailed risk information.

Keywords: Dynamic PRA, Risk importance measure, Risk triplet, Risk-informed decision making, Level 2 PRA, MELCOR/RAPID

1. INTRODUCTION

Risk-informed decision making (RIDM) considers relative risk in conjunction with engineering analyses and operating experience to ensure that rational decisions can be made for the safe operation of nuclear power plants. Nuclear regulatory bodies, including U.S. Nuclear Regulatory Commission (USNRC) and Nuclear Regulation Authority of Japan (JNRA), believe that a risk-informed approach would optimize resource apportionment and reduce unnecessary regulatory burden on utilities without reducing safety [1][2]. Regulatory bodies use the probabilistic risk assessment (PRA) techniques to examine potential risk of nuclear facilities, and identify what problems could have the most impact on public health and safety and the environment. PRA enhances and extends traditional, deterministic approach by (1) allowing consideration of broader set of potential challenges to safety, (2) providing a logical means for prioritizing these challenges based on risk significance, (3) allowing consideration of a broader set of resources to defend against these challenges [3].

Particularly, under the risk-informed performance-based regulatory framework, the USNRC is extensively using PRA and risk information to complement the deterministic approaches [4]. The USNRC has successfully practiced risk-informed approaches to regulations, for example, the amendment of maintenance rule (10 CFR 50.65) [5], reactor oversight process [6], risk-informed decisions on plant-specific changes to the licensing basis [7], risk-informed in-service inspection (ISI) [8], risk-informed categorization and treatment of structures, systems, and components (SSCs) in 10 CFR 50.69 [9], and risk-informed performance-based fire protection programs in 10 CFR 50.48 [10], to name a few.

For example, in 10 CFR 50.59, risk-informed insights are used to for categorizing SSCs based on the principles of NRC Regulatory Guide (RG) 1.174. To facilitate an overall assessment of the risk significance of SSCs, an integrated computation is performed using importance measures. Four categories are divided for SSCs according to their risk significances. The risk significance assessment process uses two standard PRA importance measures, risk achievement worth (RAW) and Fussell-Vesely (FV), as screening tools to identify candidate safety-significant SSCs [11]. In general safety assessment of complex engineering systems, some components and their arrangement may be more critical than others in terms of system reliability. There are traditional indices for measuring the importance of component: Birnbaum, criticality, FV, risk reduction worth (RRW), RAW [12].

Traditional risk importance measures (RIMs) in PRA generally use frequencies to estimate the influence of a component on the entire system. This process may provide biased estimation of importance when other risk factors have greater influence. Dynamic PRA is one of advanced PRA approaches and it can provide a better consideration of time-dependent issues and stochastic behaviors of systems and components and their influences on the overall accidental progression. Dynamic PRA uses a time-dependent phenomenological model of plant evolution along with its stochastic behavior [13]. Stochastic and deterministic behaviors of plant elements are modeled as building blocks of the risk model [14]. Dynamic PRA is capable of providing extensive estimation of time-related risk information, which is generally described by risk triplets [15]. Importance measures that can reflect timing thereby need to be developed for identifying the safety significance of pivotal events such as the actuation of accident mitigation and the loss of containment function, in which the actuation and dysfunction timing are crucial for system reliability.

The paper is the second of two consecutive presentations at the PSAM17 & ASRAM2024 conference. The objectives of the two papers are to propose additional RIMs that can provide more information for risk-informed decision making, from the perspectives such as source term release time and amount to the environment during severe accidents. The authors try to demonstrate the applicability of the new RIMs by using the dynamic PRA approach, which are proposed in the first of the two consecutive presentations [16]. The new RIMs are proposed based on the definition of risk triplet, and it is written as RIM= ⟨TBW, FBW, CBW⟩. It includes timing-based worth (TBW), frequency-based worth (FBW), and consequence-based worth (CBW). While the FBW is in accordance with the traditional PRA importance measures such as the RAW, the TBW and CBW indicate the adequacy of time-margin increasement and consequence mitigation for assessing the effects of accident countermeasures.

This paper is organized as follows. Section 2 investigates the traditional importance measures and their regulatory applications. Section 3 introduces the dynamic PRA approach and computational tool of RAPID (risk assessment with plant interactive dynamics), both of which are under development at Japan Atomic Energy Agency (JAEA) [17]. Section 4 provides a trial measurement of the proposed risk metrics through a level 2 dynamic PRA by coupling RAPID and the severe accident code MELCOR [18]. Section 5 provides concluding remarks and further development plans.

2. INVESTIGATION OF IMPORTANCE MEAURES AND THEIR REGULATORY APPLICATIONS

2.1. Review of Traditional Risk Importance Measures

Table 1 summarizes the RIMs frequently used in nuclear PRA [19]. Two principal factors determine the importance of a component in a system: the reliability/unreliability of the component and the structure of the system [12]. For instance, the Birnbaum measure completely depends on the structure of the system, e.g. whether the system is dominated by a parallel or series configuration; the FV measure is widely used in practice, both reliability/unreliability of components and system structure can be taken into account; The criticality importance is related to Birnbaum's measure, but it is also affected by the reliability/unreliability of the components and the system. The relative measures (FV, RRW, RAW) have the advantage of being more robust than the absolute measure (Birnbaum, criticality). The RIMs share characteristics of identifying safety-significant components but differs in some details for practical use. The FV measure identifies the component failures that are most likely leading to a systematic failure, and it is often used for the selection of candidates for improvement. The RAW measure is useful for identifying elements that should be prevented from failing using prognostics, planned maintenance or other failure avoidance method. The RRW measure is useful for identifying elements, if they were improved or maintained to a large extent, would result in a great reduction of risk. Components that have the greatest worths in reducing risk are not necessarily the same as those that have the highest worths in assuring that risk is maintained at low levels. RRW in general are small indicating that little risk reduction could be obtained by simply increasing the component reliability. On the other hand, RAW might be large for certain key systems, indicating system risk could dramatically increases if the components are not maintained [20].

Table 1. Traditional RIMs and definitions [21]

RIM	Definition	Interpretation and Comments
Birnbaum	$Bi_i = R_i^+ - R_i^-$	Absolute measure; Shows how often component i is needed to prevent system failure.
Criticality	$Cr_i = (R_i^+ - R_i^-)\frac{p_i}{R_0}$	Absolute measure; Shows the sensitivity of system failure probability with respect of failure probability of component i.
FV	$FV_i = \frac{R_0 - R_i^-}{R_0}$	Dimensionless and relative measure; Shows the fraction of system risk involving failure of component i.
RRW	$RRW_i = \frac{R_0}{R_i^-}$	Dimensionless and relative measure; Shows relative overall system improvement by improving component i.
RAW	$RAW_i = \frac{R_i^+}{R_0}$	Dimensionless and relative measure; Shows relative overall deterioration by the failure of component i.
Here, R_0: the present "nominal" risk level; R_i^+: the increased risk level with component "i" assumed failed; R_i^-: the decreased risk level with component "i" assumed to be perfectly reliable; p_i: failure probability of the component i.		

2.2. Regulatory Applications

Other than the traditional RIMs, incremental risk (e.g. ΔCDF and ΔLERF) in terms of core damage frequency (CDF) or large early release frequency (LERF) can be found in the regulatory activities for importance judgment. The risk increment caused by the failure of component i can also be written in a unified form the same as the RIMs in Table 1.

$$\Delta R = R_i^+ - R_0 \tag{1}$$

Table 2 lists the application of the aforementioned RIMs in risk-informed regulations, mainly taking the regulatory activities of USNRC as an example. The USNRC actively uses importance measures for risk-informed decision making, such as the risk-informed categorization of SSCs, risk-informed changes to technical specifications, application of maintenance rule, the reactor oversight process, which has also been introduced to the regulatory system by the JNRA, and licensing application activities of small modular reactors.

The literature survey on the applications of importance measures also provides successful risk-informed regulatory examples to the Japanese nuclear society, and may inspire the further development of PRA and risk-informed practices in Japan.

Table 2. The use of RIMs in regulatory activities [19]

	Risk-informed regulatory activities	Representative RIMs	References
1	Risk-informed SSC categorization and treatment of SSCs for nuclear power reactors	FV, RAW	10 CFR 50.69 [9] NEI 00-04 [11]
2	Risk-informed changes to technical specifications, evaluation of completion time (CT) and surveillance frequency (SF)	Incremental CDF (ICDF), incremental core damage probability (ICDP), incremental LERF (ILERF), incremental large early release probability (ILERP)	USNRC RG 1.177 [22] NEI 06-09 [23] NEI 04-10 [24]
3	Maintenance rule	RRW, RAW, CDF contribution (ranking of cut sets inclusion)	10 CFR 50.65 [25] USNRC RG 1.160 [26] NUMARC 93-01 [27]
4	Reactor oversight process (ROP): significance determination process (SDP), inspection planning	ΔCDF, ΔLERF, Birnbaum, FV, RRW, RAW	NUREG-1649 [28] USNRC Inspection Manual • Manual chapter 0609 [29] • Inspection procedure 71111 [30]
5	Licensing application activities of small modular reactors (SMRs)	FV, RAW, Conditional CDF, Conditional LRF	Design Certification Applications [31]: NuScale VOYGR, Westinghouse AP300, HOLTEC SMR-160

2.3 The Proposed New RIMs Based on Risk Triplets for Dynamic PRA

The use of traditional RIMs for dynamic PRA has been presented previously, and it is confirmed that traditional RIMs (e.g., FV and RAW) can be determined from simulation-based data instead of minimal cut sets (MCSs) [32]. To exclusively consider factors such as time and consequences, this study applies the newly proposed RIMs to a Level 2 dynamic PRA. The RIMs are proposed by Narukawa et al., from the perspectives of the risk triplet, and a complete definition of the RIMs can be found in the reference [16]. They are capable of assessing time dependency, evaluating resilience including accident management (AM), and aligning with existing risk importance measures. The RIMs include three elements of ⟨TBW, FBW, CBW⟩. Here, the authors describe the RIMs in a simpler manner, without the logarithmic function, as follows.

$$TBW = \frac{E[T(end\ state)]}{E_i[T(end\ state)]} \tag{2}$$

$$FBW = \frac{E_i[F(end\ state)]}{E[F(end\ state)]} \tag{3}$$

$$CBW = \frac{E_i[C(end\ state)]}{E[C(end\ state)]} \tag{4}$$

Here, in Equation (2), to ensure a lager metric represents more importance, the TBW uses an inverse value of the ratio of source term release timing, before and after the assuming failures, and i is a basic event such as the failure of component i or a heading event, $E[T(end\ state)]$ is the expected occurrence time of the end state over the entire analysis, $E_i[T(end\ state)]$ is the expected occurrence time of the end state, conditioning on the occurrence of the basic event i; in Equation (3), $E[F(end\ state)]$ is the expected frequency of the end state over the entire analysis, and $E_i[F(end\ state)]$ is the expected frequency or probability of R, conditioning on the occurrence of the basic event i ; in Equation (4), $E[C(end\ state)]$ is the expected consequence of the end state over the entire analysis, and $E_i[C(end\ state)]$ is the expected consequence involving the end state, conditioning on the occurrence of the basic event i.

All the measures indicate an improvement in safety with positive values. The key differences from existing risk importance measures lie in TBW and CBW. TBW indicates the time margin until the occurrence of an evaluation event, making it an important measure for assessing the feasibility and success probability of AM, thus contributing to resilience evaluation. CBW represents the consequence, such as the release of fission products, serving as a measure to evaluate the consequence mitigation effect of AM, etc.

3. JAEA's DYNAMIC PRA APPROACH AND COMPUTATIONAL TOOL

3.1. Dynamic PRA Approach

By explicitly considering time-dependent issues and stochastic behaviors of systems and components, dynamic PRA is one approach to advancing traditional PRA, and it is potentially capable of alleviating some epistemic uncertainties of PRA. Dynamic PRA can provide a time-dependent risk assessment which tracks changes, interdependences and interactions among plant elements, so the overall methodology is more dynamic than logic-based PRA. Over the past decades, dynamic PRA approaches have been developed and advanced, and numerous computational tools have emerged worldwide, with various applications to nuclear reactor risk assessment. Known as integrated deterministic and probabilistic safety assessment (IDPSA) [33] as well, dynamic PRA tightly couples probabilistic and deterministic simulations to address aleatory (stochastic aspects of accident scenarios) and epistemic (model and parameters) uncertainties in a consistent manner [34]. To instantiate the concept of IDPSA and referring to previous risk assessment approaches such as the RISMC approach [35], JAEA is developing the dynamic PRA approach as shown in Figure 1. It consists of two interactive parts of probabilistic and deterministic analyses, for scenario development and consequence simulation, respectively. After a number of Monte Carlo simulation, the brute-force approach splits the plant state space into a finite number of accident sequences, and checks all sequences using deterministic simulations codes to see if undesired results occur, such as core damage or fission product release. When the sampling number is large enough, the approach is capable of providing risk information in the form of risk triplet and temporal information, all of which are beneficial to rational decision makings.

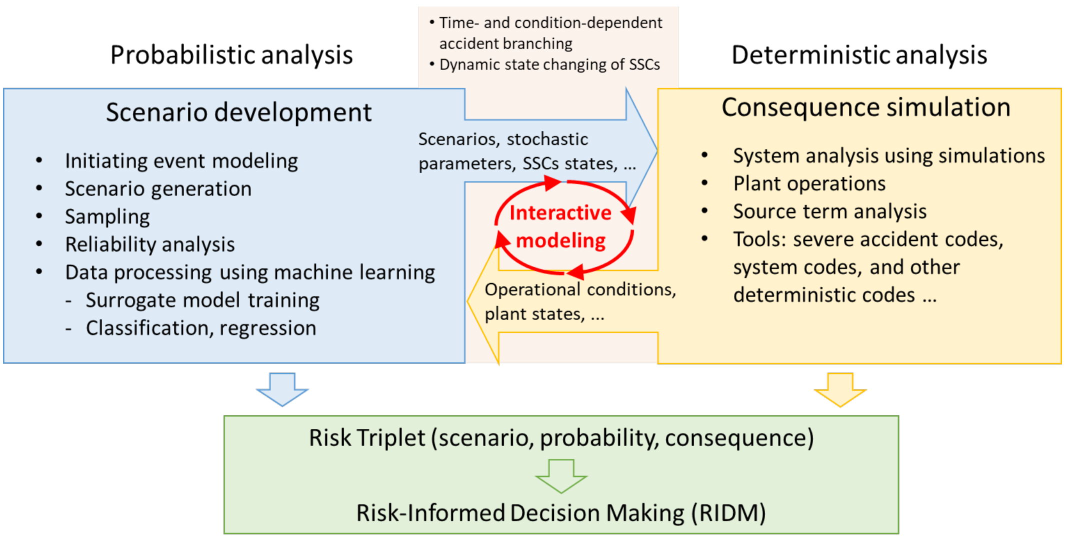


Figure 1. The integrated probabilistic and deterministic approach for dynamic PRA

3.2. RAPID for Dynamic PRA

To implement the dynamic PRA approach, JAEA is also developing the computational tool of RAPID. Figure 2 illustrates the consisting components and how it controls and interacts with deterministic codes, such as system codes, severe accident codes, surrogate models and PRA models, for providing risk information. The tool consists of three parts: scenario generator, simulation controller and postprocessor.

4. TRIAL MEASUREMENT OF THE PROPOSED RISK METRICS USING LEVEL 2 DYNAMIC PRA

A simplified event tree (ET) of a boiling water reactor (BWR) station blackout scenario [36] has been built for the demonstration, and to simply the calculation, the containment event tree (CET) model is reduced to two failure modes of the Mark I containment, including bypass and overpressure failure [37]. Figure 3 illustrates the computational process of the dynamic PRA. Starting from the random sampling of eleven stochastic parameters from the ET and CET models, accident progressions are simulated using the severe accident code of MELCOR 2.2. The input deck has been built based on BWR test case input of Sandia National Laboratories (SNL). As shown in the diagram, the BWR plant model includes two main parts of hydrodynamics and core. Core channel has been divided in two control volumes of core and bypass. The reactor coolant system (RCS) is modeled as a lower plenum, downcomer, upper plenum with reactor pressurized dome (RPV). Control volumes are connected with flow paths, which allow mass and energy exchange. Containment system consists of wetwell and drywell. Drywell is equipped with a filtered vent to the environment. Drywell is accepting mass from lower plenum leak and releasing mass to the environment after when the containment fails.

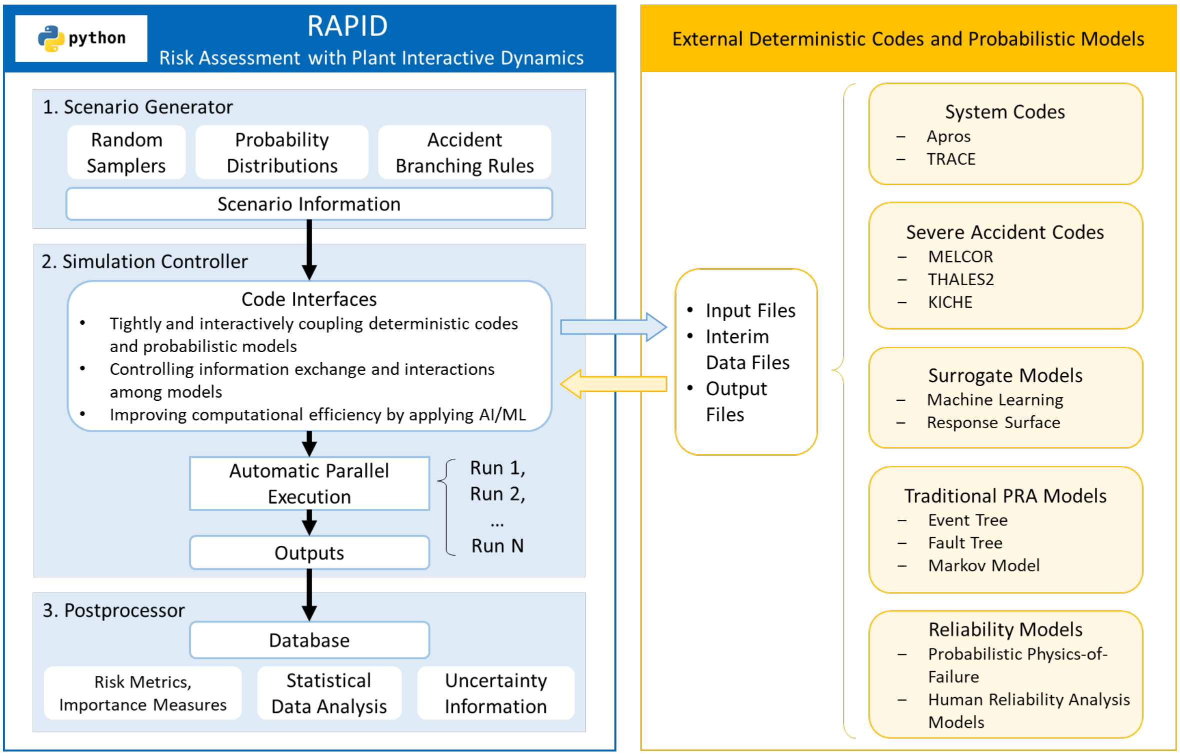
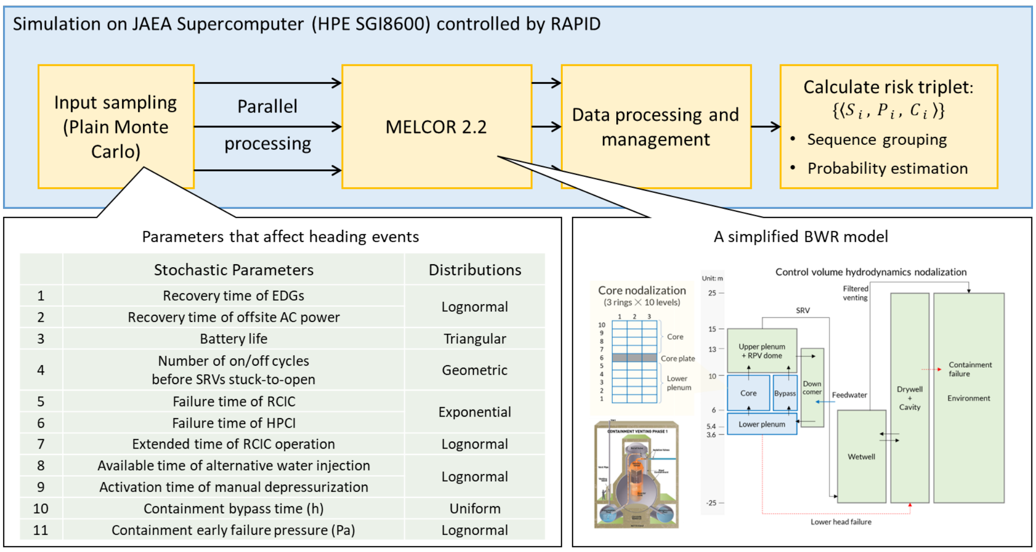
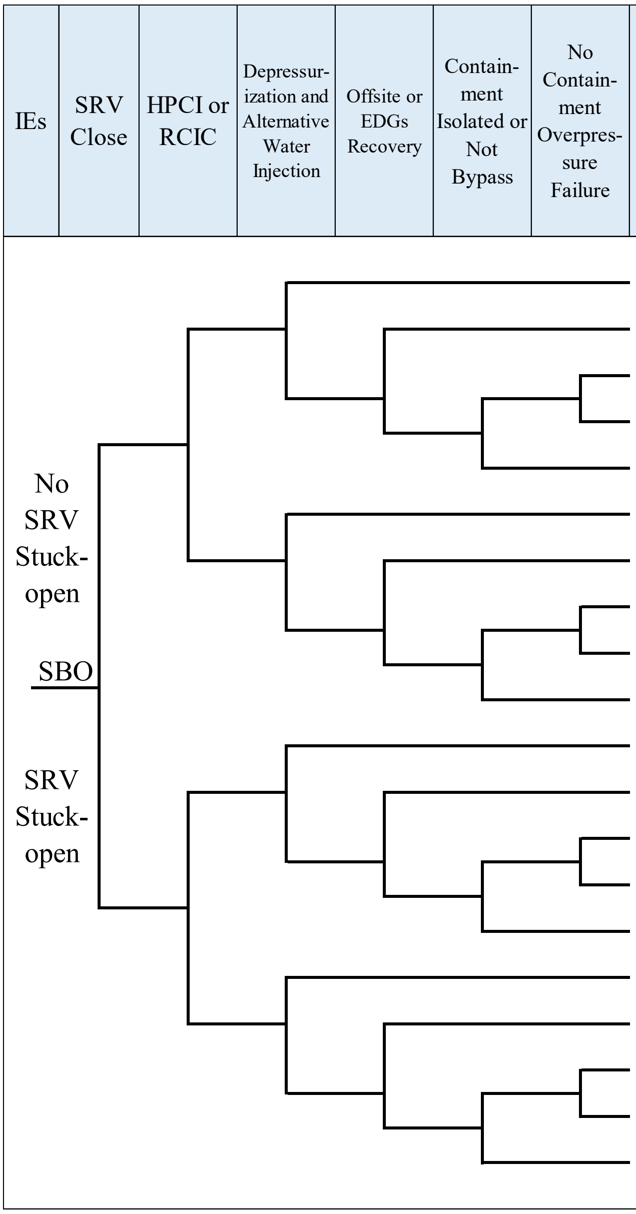


Figure 2. The RAPID tool coupling probabilistic models and deterministic codes for dynamic PRA



	Stochastic Parameters	Distributions
1	Recovery time of EDGs	Lognormal
2	Recovery time of offsite AC power	
3	Battery life	Triangular
4	Number of on/off cycles before SRVs stuck-to-open	Geometric
5	Failure time of RCIC	Exponential
6	Failure time of HPCI	
7	Extended time of RCIC operation	Lognormal
8	Available time of alternative water injection	Lognormal
9	Activation time of manual depressurization	
10	Containment bypass time (h)	Uniform
11	Containment early failure pressure (Pa)	Lognormal

Figure 3. Calculation process, MELCOR modelling and stochastic parameters of SBO ET and CET



#	Source Term Release	Probability	Averaged Release Start Time (hour)	Averaged Release Fraction (-)
1	No	2.27E-01	INF	0
2	No	7.54E-01	INF	0
3	No	9.50E-03	INF	0
4	Yes	2.38E-03	2.24E+01	1.42E-02
5	Yes	6.25E-04	1.95E+01	1.86E-01
6	No	8.99E-04	INF	0
7	No	2.90E-03	INF	0
8	No	1.06E-03	INF	0
9	Yes	2.66E-04	1.33E+01	1.55E-02
10	Yes	7.00E-05	1.01E+01	1.78E-01
11	No	1.90E-04	INF	0
12	No	6.04E-04	INF	0
13	No	4.42E-05	INF	0
14	Yes	1.10E-05	2.11E+01	2.00E-02
15	Yes	2.91E-06	1.64E+01	2.00E-01
16	No	1.37E-06	INF	0
17	No	2.31E-06	INF	0
18	No	4.55E-07	INF	0
19	Yes	1.14E-07	1.42E+01	2.50E-02
20	Yes	3.00E-08	9.65E+00	2.16E-01

Figure 4. Results of Level 2 PRA including probabilities, source term release timing and fractions of sequences

To alleviate the computational cost for simulating low-frequency sequences, a machine-learning-based surrogate model has been trained to predict input sets with similar features. Combining the surrogate and MELCOR code, totally 1.34×10^6 accident sequences has been sampled.

With the safety relief valve (SRV) stuck-open probability setting as 8.56×10^{-4}, Figure 4 summarizes the probabilities of accident sequences, the associated averaged source term release timing and release fractions to the environment (the severity of consequences). Sequences 4, 5, 9, 10, 14, 15, 19 and 20 are severe accidents with core damage and source term released to the environment. Reflecting the estimated values (probabilities, the averaged values of release timing and release fractions) to Equations (2)-(4) , the results of new RIMs=⟨TBW, FBW, CBW⟩ are obtained in Table 3. The FBW represents the influence of the status of a heading event on eliminating the occurrence of a severe accident. As the final barrier of source term release, containment integrity dominates importance based on the FBW measure (similar to FV and RAW), so the headings of "Containment Isolated or Not Bypass" and "No Containment Overpressure Failure" rank the highest, in which the metric reveals that without a functional containment, the source term release frequency would be 298 times higher. On the other hand, the heading of "high-pressure coolant injection (HPCI) or reactor core isolation cooling (RCIC)" shows lager influences on the source term release timing by delaying core damage, which results in a higher ranking based the measures of TBW. The results show that without a functional HPIC or RCIC, the averaged release timing of source term would be (1-1/1.34) = 25.4% earlier. On the other hand, the heading of "Containment Isolated or Not Bypass" shows greater influences on release fractions since containment bypass will give rise to the unscrubbed source term releases, causing a more serious radiological consequence, so the corresponding CBW is higher. The results show that if the "Containment Isolated or Not Bypass" fails, the averaged estimate of source term would be 1.82 times higher. However, if assuming that the containment never fail during the early phase of an accident and the "late and overpressure failure" is the only failure mode, the release time would be postponed and release amount would be mitigated. The complete success or failure states of headings of "Depressurization and Alternative Water Injection" and "Offsite or EDGs Recovery" do not change the distributions of release timing and fractions among sequences, but they moderately affect the occurrence frequencies of sequences, with the values of RIMs ($\langle 1,\ 71.4, 1\rangle$) showing that both two heading events would provide long-time water cooling to the reactor. Such numeric results can provide risk insights from the perspectives of occurrence timing and severity of consequences.

Table 3 Preliminary results of importance analysis using the new RIMs

Event Tree Headings	TBW	FBW	CBW
SRV Close	1.03	4.88	1.07
HPCI or RCIC	1.34	19.2	1.02
Depressurization and Alternative Water Injection	1	71.4	1
Offsite or EDGs Recovery	1	71.4	1
Containment Isolated or Not Bypass	1.14	298	1.82
No Containment Overpressure Failure	0.885	298	0.175

5. CONCLUSIONS

The paper investigated use cases of risk importance measures in nuclear regulation, and performed a dynamic PRA to estimate a new RIM that is based on the concept of risk triplet. The following conclusions are obtained.

- RIMs have shown applicability in nuclear regulation, for instance, the risk-informed categorization of SSCs, risk-informed changes to technical specifications, application of maintenance rule, the reactor oversight process, which has also been introduced to the regulatory system by the JNRA, and licensing application activities of small modular reactors, to name a few.
- Frequency-based traditional RIMs show disadvantages in the proper treatment of timing issues in PRA. The authors proposed a new RIM based on risk triplet, and simulation-based dynamic PRA is applied to assess the occurrence timing of pivotal events.
- JAEA is developing the approach and computational tool for dynamic PRA. Trial assessment of the new RIMs has been performed to demonstrate the computational practicability. With coupling severe accident code of MELCOR and dynamic tool of RAPID, it is capable of generating possible accident sequences and calculating risk triplets as well as release timing of source term. Pivotal events are ranked from the perspective of their combined influences on the accident frequencies, source term release timing and fractions to the environment.
- The study provides an alternative measure to the traditional importance measures. When issues such as timing or seriousness of consequence are important for the regulatory judgement, dynamic PRA and the new RIMs are capable of supporting decision making by providing more detailed risk information. However, it seems more reasonable when only comparing RIMs between heading events of the similar functions.

Acknowledgements

The authors are grateful for valuable discussions with Mr. Yoshikane Hamaguchi of the JNRA on the use of risk information in the reactor oversight process at the JNRA. The Dynamic PRA simulations have been conducted using the supercomputer system of HPE SGI8600 at Japan Atomic Energy Agency (JAEA).

References

[1] United States General Accounting Office. Strategy needed to regulate safety using information on risk. GAO/RCED-99-95, United States General Accounting Office, 1999
[2] Fuketa T. Use of risk information in regulation. In: Proceedings of Atomic Energy Society of Japan 2015 Annual Meeting, Ibaraki, Japan, March 20-22, 2015 (in Japanese)
[3] U.S. Nuclear Regulatory Commission. Use of probabilistic risk assessment methods in nuclear regulatory activities, final policy statement. Federal Register 60(158): 42622-42629, 1995
[4] Nourbakhsh HP, Apostolakis G and Powers DA. The evolution of the U.S. nuclear regulatory process. Progress in Nuclear Energy 102:79-89, 2018
[5] U.S. Code of Federal Regulations, Title 10, Part 50, Section 50.65. Requirements for monitoring the effectiveness of maintenance at nuclear power plants, 1999
[6] U.S. Nuclear Regulatory Commission. Reactor oversight process. NUREG-1649, Revision 6. U.S. Nuclear Regulatory Commission Washington, DC, 2016
[7] U.S. Nuclear Regulatory Commission. An approach for using probabilistic risk assessment in risk-informed decisions on plant-specific changes to the licensing basis. Regulatory Guide 1.174, Revision 3. U.S. Nuclear Regulatory Commission Washington, DC, 2018
[8] U.S. Nuclear Regulatory Commission. Issues and recommendations for advancement of PRA technology in risk-informed decision making. NUREG/CR-6813, U.S. Nuclear Regulatory Commission Washington, DC, 2003
[9] U.S. Code of Federal Regulations, Title 10, Part 50, Section 50.69. Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors, 2023
[10] U.S. Code of Federal Regulations, Title 10, Part 50, Section 50.48. Fire protection, 2023
[11] Nuclear Energy Institute. 10 CFR 50.69 SSC categorization guideline. Nuclear Energy Institute, Washington, DC, 2005
[12] Modarres M, Kaminskiy MP and Krivtsov V. Reliability engineering and risk analysis: a practical guide, Third edition. CRC Press, Taylor & Francis Group. Boca Raton, FL, 2017
[13] Aldemir T. A survey of dynamic methodologies for probabilistic safety assessment of nuclear power plants. Annals of Nuclear Energy 52:113–24, 2013
[14] Mosleh A. PRA: a perspective on strengths, current limitations, and possible improvements. Nuclear Engineering and Technology 46(1):1–10, 2014
[15] Kaplan S, Garrick BJ. On the quantitative definition of risk. Risk Analysis 1(1):11–27, 1981
[16] Narukawa T, Takata T, Zheng X, Tamaki H, Sibamoto Y, Maruyama Y and Takada T. Development of risk importance measures for dynamic PRA based on risk triplet (1) the concept and measures of risk importance. In: Proceedings of the international conference on probabilistic safety assessment and management & Asian symposium on risk assessment and management (PSAM 17& ASRAM2024). Miyagi, Japan. October 7-11, 2024
[17] Zheng X, Tamaki H, Sugiyama T and Maruyama Y. Dynamic probabilistic risk assessment of nuclear power plants using multi-fidelity simulations. Reliability Engineering and System Safety 223:108503, 2022
[18] Humphries LL, Beeny BA, Gelbard F, Louie DL and Phillips J. MELCOR computer code manuals vol. 1: primer and users' guide version 2.2.11932. SAND2018-13559 O, Sandia National Laboratories. Albuquerque, NM, 2018
[19] Vrbanic I, Samanta P and Basic I. Risk importance measures in the design and operation of nuclear power plants. BNL-114389-2017-BC, Brookhaven National Laboratory. The American Society of Mechanical Engineers (ASME), New York, 2017] [Vesely WE, Davis TC, Denning RS and Saltos N. Measures of risk importance and their applications. NUREG/CR-3385, U.S. Nuclear Regulatory Commission, Washington, DC, 1983
[20] Vesely WE and Davis TC. Two measures of risk importance and their application. Nuclear Technology 68(2):226-234, 1985

[21] Cheok MC, Parry GW and Sherry RR. Use of importance measures in risk-informed regulatory applications. Reliability Engineering and System Safety 60(3): 213-226, 1998
[22] U.S. Nuclear Regulatory Commission. Plant-specific, risk-informed decisionmaking: technical specifications. Regulatory Guide 1.177, Revision 2, U.S. Nuclear Regulatory Commission Washington, DC, 2021
[23] Nuclear Energy Institute. Risk-informed technical specifications initiative 4b risk-managed technical specifications (RMTS) guidelines, NEI 06-09, Revision 0. Nuclear Energy Institute, Washington, DC, 2006
[24] Nuclear Energy Institute. Risk-informed technical specifications initiative 5b: risk-informed method for control of surveillance frequencies, NEI 04-10, Revision 0. Nuclear Energy Institute, Washington, DC, 2006
[25] U.S. Code of Federal Regulations, Title 10, Part 50, Section 50.65. 50.65 Requirements for monitoring the effectiveness of maintenance at nuclear power plants, 2023
[26] U.S. Nuclear Regulatory Commission. Monitoring the effectiveness of maintenance at nuclear power plants. Regulatory Guide 1.160, Revision 4, U.S. Nuclear Regulatory Commission Washington, DC, 2018
[27] Nuclear Energy Institute. Industry guideline for monitoring the effectiveness of maintenance at nuclear power plants, NUMARC 93-01, Revision 4. Nuclear Energy Institute, Washington, DC, 2010
[28] U.S. Nuclear Regulatory Commission. Reactor oversight process. NUREG-1649, Revision 6, U.S. Nuclear Regulatory Commission Washington, DC, 2016
[29] U.S. Nuclear Regulatory Commission. Inspection manual, manual chapter 0609, https://www.nrc.gov/reading-rm/doc-collections/insp-manual/manual-chapter/index.html, 2020
[30] U.S. Nuclear Regulatory Commission. Inspection manual, inspection procedure 7111 https://www.nrc.gov/reading-rm/doc-collections/insp-manual/inspection-procedure/index.html, 2022
[31] U.S. Nuclear Regulatory Commission. Licensing Activities for Small Modular Reactors, https://www.nrc.gov/reactors/new-reactors/smr/licensing-activities.html, accessed in 2024
[32] Mandelli D, Ma Z, Parisi C, Alfonsi A and Smith C. Measuring risk-importance in a dynamic PRA framework. Annals of Nuclear Energy 128: 160–170, 2019
[33] Zio E. Integrated deterministic and probabilistic safety assessment: concepts, challenges, research directions. Nuclear Engineering and Design 280:413-419, 2014
[34] Aldemir T. A survey of dynamic methodologies for probabilistic safety assessment of nuclear power plants. Annals of Nuclear Energy 52:113-124, 2013
[35] Smith C, Rabiti C and Szilard R. Risk-informed safety margins characterization (RISMC) pathway technical program plan. INL/EXT-17-43243 Revision 0, Idaho National Laboratory, Idaho Falls, Idaho, 2017
[36] Mandelli D, Smith C, Riley T, Nielsen J, Alfonsi A, Cogliati J, Rabiti C and Schroeder J. BWR station blackout: a RISMC analysis using RAVEN and RELAP5-3D, Nuclear Technology, 193(1):161-174, 2016
[37] Pratt WT, Mubayi V, Chu TL, Martinez-Guridi G and Lehner J. An approach for estimating the frequencies of various containment failure modes and bypass events, NUREG/CR-6595, Rev. 1. U.S. Nuclear Regulatory Commission, Washington, DC, 2004