Application of Dynamic PRA to Nuclear Power Plant Operation Support
- Evaluation of Plant Operation Support Using a Simple Plant Model -

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Abstract: The need to enhance safety after the Great East Japan Earthquake in 2011 highlighted the importance of risk assessment and the use of information from that assessment. In particular, dynamic probabilistic risk assessment (PRA) coupled with plant dynamics analysis is increasingly expected to be applied to accident management (AM) and operation support. In the development of AM, it is difficult to determine the priority of countermeasures due to the diversity of accident scenarios. Furthermore, in multi-unit operations, it is expected that scenarios will become more complex in the event of simultaneous disasters, making it increasingly difficult to determine priorities of response operations.

It is possible to generate and evaluate complex scenarios efficiently and exhaustively with a dynamic PRA method, where multiple scenarios are generated in accordance with changes in plant state. This paper proposes using the Continuous Markov chain Monte Carlo (CMMC) method, a dynamic PRA approach, for determining countermeasure priorities to support nuclear power plant operations. The proposed method consists of the following three steps: 1) various scenarios consisting of events, operator countermeasures, and system actions are exhaustively generated, 2) the scenarios are classified based on the countermeasure pattern, and 3) priority is determined based on risk information for each pattern.

We carried out a trial evaluation of the event countermeasure patterns using a simple plant model. As a countermeasure to the occurrence of steam generator tube rupture in a single-unit operation, the following patterns of scenarios were created: depressurization by opening a pressurizer relief valve (DP), depressurization by heat removal using a steam generator (DSG), and both (DP+DSG). For each pattern, the timing of implementing the response operations was randomly changed and multiple scenarios were generated. As a result of judging whether each scenario leads to core damage based on the water level of the reactor pressure vessel, we found that the time margin to the core damage differed depending on the countermeasure pattern. This result suggests that it is possible to evaluate the effectiveness of each countermeasure and to judge the countermeasure to be carried out preferentially.

Keywords: Dynamic PRA, Plant Operation Support, Use of Risk Information

1. INTRODUCTION

The need to enhance safety at nuclear power plants after the Great East Japan Earthquake in 2011 highlighted the importance of risk assessment and the use of information from that assessment. In particular, dynamic probabilistic risk assessment (PRA) coupled with a plant dynamic analysis is increasingly expected to be applied to the development of accident management (AM) and operation support in addition to conventional safety assessments. In the development of AM, it is difficult to determine the priority of countermeasures due to the diversity of accident scenarios, the indeterminacy of the order in which events occur, and the time dependence of the branching probability. In plant operation, the range of automation for the purpose of efficiency improvement and precision in plant operation has been expanding in recent years in line with technological advances. As a result, it is considered possible to carry out multi-unit operation in which the scope of responsibility for monitoring and operation per operator is increased. In response to an event, it is difficult to rely entirely on automatic operation, so intervention, including manual operation in accordance with the situation, using risk information is essential. However, in the case of a simultaneous multi-unit disaster, the scenario is expected to become more complex and the burden of determining the response priority to be higher than in single-unit operations.

This paper proposes a countermeasure priority decision function using dynamic PRA as a support function for safer operation of nuclear power plants. First, a simple plant dynamic characteristic analysis model of a single unit is constructed and the results of a trial evaluation of the priority of human operation by the operator at the time of the event are reported.

2. COUNTERMEASURE PRIORITY DETERMINATION FUNCTION USING DYNAMIC PRA

2.1. Continuous Markov Chain Monte Carlo Method

The continuous Markov chain Monte Carlo (CMMC) method, which is a dynamic PRA approach, has been studied [1,2]. In the CMMC method, the transition of the scenario is considered to be a Markov process. That is, the plant state at the current time (a combination of the states of each component) depends only on the plant state at the previous time and not on the plant state at a time earlier than that. Physical processes in plants are generally modeled by Markov processes and plant dynamics analysis codes are used for their evaluation. Events such as damage occurring at a plant depend on the temperature, pressure, and other factors at that location. While it is difficult to deal directly with changes in plant conditions with static PRA, the CMMC method can be evaluated by plant simulation. Figure 1 shows the concept of the CMMC method. This procedure is repeated until an arbitrary analysis end time to create a unique scenario.

The flow of scenario generation in the CMMC method is shown in Figure 2. After the analysis of one scenario is completed, the same process is repeated to create a number of accident scenarios until all of the predetermined number of scenarios scheduled to be created have been created. In each scenario, the Monte Carlo method is used to determine the state transitions, so they are unique to each other. Therefore, it is possible to quantify an exhaustive set of accident scenarios by analyzing a large number of scenarios and to obtain statistical probability information with uncertainties such as the proportion of scenarios resulting in core damage among all scenarios.

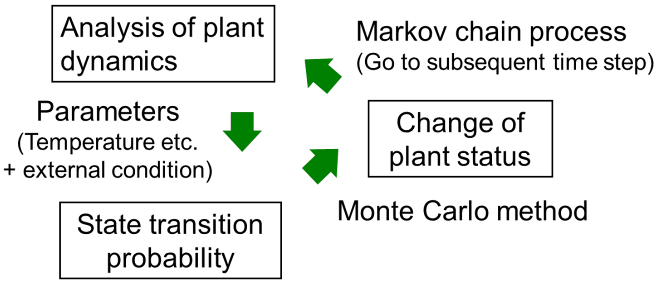


Figure 1. Concept of CMMC Method [2]

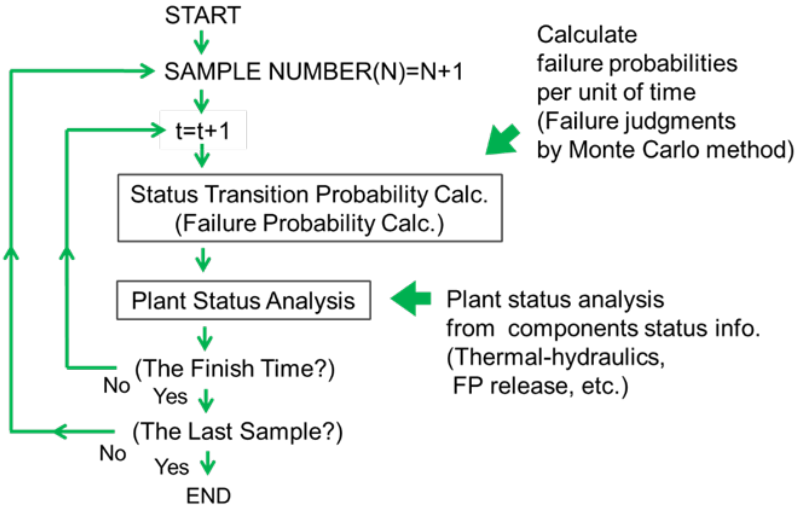


Figure 2. Scenario Generation Flow using CMMC Method [3]

2.2. Countermeasure Priority Determination

The priority of countermeasures in the event of a nuclear power plant event is expected to vary depending on the status of the plant at the time of the event and the progress of the event. Changes in plant conditions include uncertainties and are not uniquely determined. Plant conditions are also affected by the countermeasures to be carried out and the timing of their implementation; therefore, evaluation in a scenario considering various conditions is required. In response to the above problems, this paper proposes a countermeasure priority decision function utilizing the CMMC method, which can efficiently and exhaustively generate and evaluate scenarios in response to ever-changing situations. The CMMC method is used to generate a number of scenarios including events occurring at the plant and event countermeasures, and to prioritize countermeasures in order of decreasing risk based on the risk information of each scenario. This will support the evaluation of the effectiveness of countermeasures and decision making considering various conditions when developing AM. In addition, when an event occurs during plant operation, the countermeasures priority is presented to the operator, thereby reducing the load on determining the countermeasure in complex scenarios.

Figure 3 shows the flow of determining the countermeasures priority using the CMMC method. A number of scenarios are generated in which events, including an event occurring in a plant and response operations, is generated using a state transition probability corresponding to a plant situation. For each generated scenario, risk information such as the presence of core damage and the reactor pressure vessel (RPV) level in the final state is calculated. Next, scenarios are classified into patterns according to whether or not event countermeasures are performed and the timing of implementation, and risk information is aggregated for each pattern. Finally, based on the risk information for each pattern, which pattern is effective as a countermeasure is evaluated as a priority.

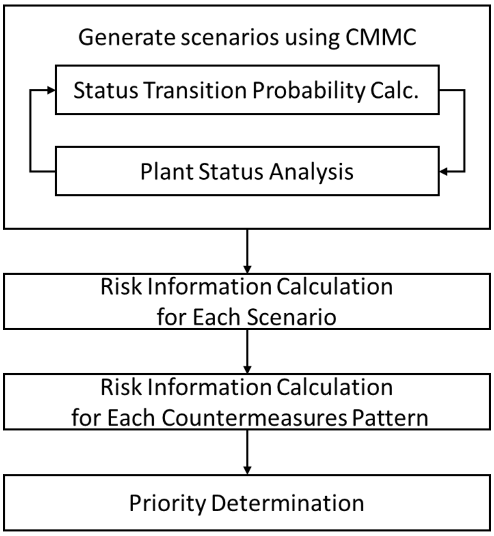


Figure 3. Determination Flow of Countermeasures Priority

3. EVALUATION of COUNTERMEASURE PRIORITIZATION for SGTR SCENARIOS

Prototyping and evaluating the effectiveness of the proposed method will be carried out in steps as follows:

Step 1. Scenario generation and evaluation in a single unit using dynamic PRA (uniform distribution)
Step 2. Scenario generation and evaluation in a multi-unit using CMMC method

In this paper, the prototype and evaluation of Step 1 were carried out.

3.1. Evaluation Scenario

To evaluate the countermeasure priority determination function, the event and the countermeasures to be evaluated were selected. One scenario in which the priority of the operator's countermeasures may change according to the progress of the event is the failed steam generator isolation scenario in the event of a steam generator tube rupture (SGTR). This scenario is one in which the primary coolant continues to leak out of the containment vessel due to an isolation failure of the steam generator on the failure side during the SGTR, resulting in core damage.

When this scenario occurs, the operator performs core cooling by cooldown and recirculation. There are two main response operations for cooldown and recirculation. One is heat removal by a steam generator using a main steam relief valve and an auxiliary water supply system (cooling from the secondary system). The other is the decompression of the primary system by a pressure relief valve. These response operations control leakage by maintaining pressure equilibrium between the primary cooling system and the steam generator on the failure side, and then the core is cooled in the long term by the residual heat removal system. To equalize the primary and secondary systems in cooldown and recirculation, the decision whether to perform depressurization by heat removal using the steam generator (DSG) first or depressurization by opening the pressure relief valve (DP) first depends on the reactor’s cooling condition. Therefore, for example, the following two scenarios can be considered as scenarios in which the priority of the operator's countermeasure changes.

Scenario 1: Event occurrence → High pressure water injection → Depressurization by heat removal using the steam generator (DSG) → Depressurization by opening the pressure relief valve (DP)

Scenario 2: Event occurrence → High pressure water injection → Depressurization by opening the pressure relief valve (DP) → Depressurization by heat removal using the steam generator (DSG)

In this way, the priority of the operator's countermeasures changes according to the reactor’s cooling condition. Therefore, the steam generator isolation failure scenario on the damaged side during an SGTR was selected as the scenario to be evaluated in this study.

3.2. Analytical System

To simulate plant behavior in the selected scenario, we constructed a simplified plant dynamic analysis model of the primary system of a pressurized water reactor (PWR) assuming a single phase. The plant model overview is shown in Fig. 4. As shown in the figure, the primary system has four nodes (RPV, Hot leg (including Pressurizer line), Steam Generator, and Cold leg (including pump)). Although a four-loop configuration is assumed, three loops of the normal system are simulated with one loop, resulting in a two-loop configuration: the normal loop and the abnormal loop. For each node, the governing equations for conservation of mass, conservation of energy, and conservation of momentum are solved.

The level of the RPV is used as the criterion for core damage. The coolant volume is calculated based on the inflow and outflow to the RPV and the RPV level is simply calculated by dividing by the representative area. Core damage is identified if the liquid level falls below a certain value within the analysis time.

When an SGTR occurs, the emergency core cooling system (ECCS) is activated in the initial stage. In this analysis, the ECCS was simulated simply by injecting a certain amount of water into the RPV.

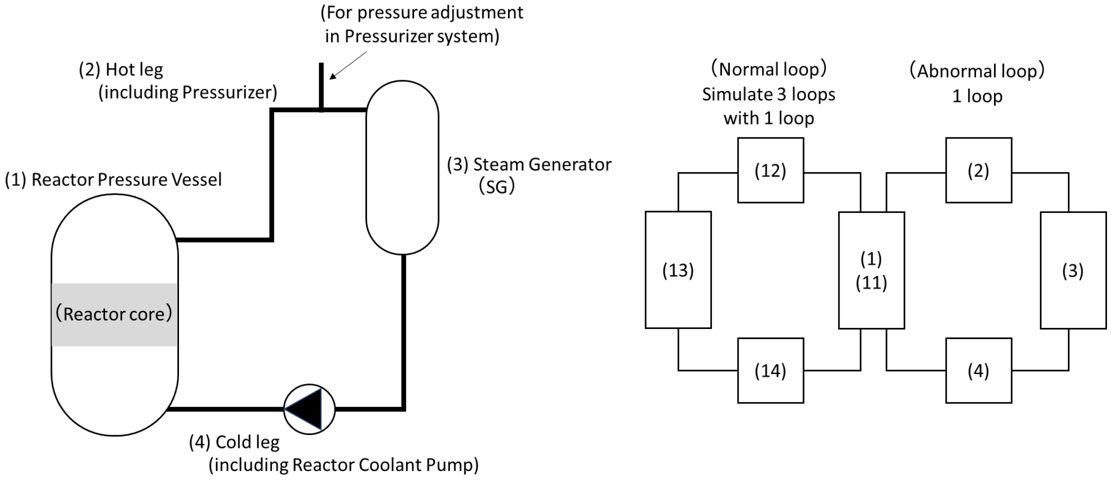


Figure 4. Plant Model Overview

3.3. Analysis Conditions

The analysis conditions are listed in Table 1. The SGTR event occurred after the plant state was brought to a steady state. To confirm the difference between the with and without uncertainty, four cases were evaluated: one case with and three cases without uncertainty in the initial leakage rate of the SGTR. The initial leakage rate of the former case was determined by a uniform distribution between 40 to 60 [kg/s] with uncertainty. For the latter cases, fixed values of 40, 50, and 60 [kg/s] were set, respectively. The leakage rate after the second step of the analysis was determined by the pressure difference. The ECCS was started in all scenarios, and its start and duration times were held constant. To evaluate the differences between the two response operations, DP and DSG, depending on whether and when they were performed, the success probability of DP and DSG activation was set to 0.5 (only one decision per sample) and the timing of their initiation was determined based on a uniform distribution. The analysis period was set to 3600[s]. For each case, 10,000 sample scenarios were generated.

Table 1. Analysis Conditions

		CASE 1	CASE 2	CASE 3	CASE 4
SGTR	Initial leakage rate [kg/s]	40 to 60	40	50	60
	Start Time [s]	5			
ECCS	Start Time [s]	150			
	Duration [s]	1350			
DP	Start Time [s]	500 to 2000			
	Duration [s]	500			
DSG	Start Time [s]	500 to 2000			
	Duration [s]	500			

3.4. Priority Determining Method

Each scenario generated based on the analysis conditions shown in 3.3 is classified into 7 patterns shown in Table 2 according to the end state of the analysis.

Table 2. Pattern Classification by Final State

PATTERN	DESCRIPTION
0	Neither DP nor DSG is activated
1	Only DP is activated
2	Only DSG is activated
3	DP, DSG are both activated in that order (with overlapping periods)
4	DP, DSG are both activated in that order (no overlapping periods)
5	DSG, DP are both activated in that order (with overlapping periods)
6	DSG, DP are both activated in that order (no overlapping periods)

For each classified pattern, the conditional core damage probability and the average level of the RPV coolant at the end of the analysis are calculated. Based on these values, each pattern is prioritized and evaluated, which would be the effective response operations. The prioritization rules are as follows:

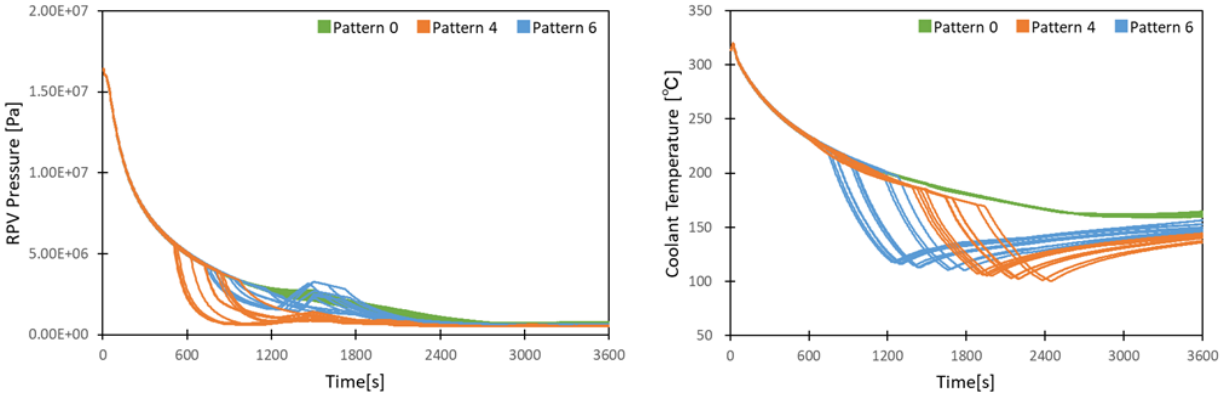
1. Patterns with 0 analysis results (samples) are not prioritized.
2. If samples exist, priority is given to the pattern with a lower conditional core damage probability.
3. If the conditional core damage probabilities are the same, priority is given to the pattern with the higher average level of the coolant in the RPV.

4. RESULTS and DISCUSSION

Table 3 shows the number of scenarios generated in each pattern, the number of samples resulting in core damage, the core damage probability, the RPV average level at the end state, and the priority order results for CASE 1 with an uncertainty of 40 to 60 [kg/s] for the leakage rate. The high priority of Patterns 3 through 6 indicates that it is more effective to perform both response operations than to perform the response operations alone. In particular, Patterns 4 and 3, which were activated from the DP, have a high ranking regardless of whether they were duplicates or not. Figure 5 shows the trend graphs of RPV pressure, coolant temperature, leak rate, and RPV level in Patterns 0, 4, and 6. The dashed line in Figure 5(d) represents the threshold of the RPV level for core damage, which is the criterion for success or failure of the event response. Patterns 4 and 6 are both patterns in which DP and DSG are executed without overlapping time, but the execution order is different. In Pattern 4, in which DP starts first, it appears that the amount of leakage decreased by lowering the RPV pressure and the time to reach the RPV level, which is considered to be core damage, increased. On the other hand, Pattern 6, in which the DSG starts first, has a slight pressure drop due to a sudden drop in the coolant temperature but is not so effective in lowering the leakage amount and the RPV level. After the operation of the DSG, the coolant temperature gradually rises and the pressure slightly increases again. As a result, the leakage rate increases and the time to reach core damage is shorter than that in Pattern 4.

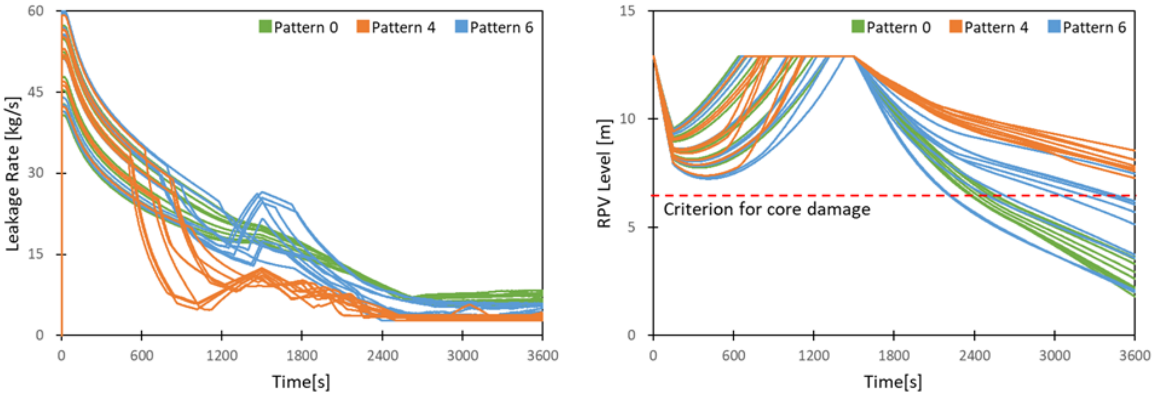
Table 3. Analysis Result (CASE 1)

PATTERN	SAMPLE	CD-SAMPLE	CD-PROBABILITY	AVG-WATER LEVEL	PRIORITY
0	2488	2488	1.00E+00	2.49E+00	7
1	2464	2258	9.16E-01	5.75E+00	5
2	2572	2572	1.00E+00	2.91E+00	6
3	666	124	1.86E-01	8.05E+00	2
4	529	0	0.00E+00	8.05E+00	1
5	690	222	3.22E-01	7.15E+00	3
6	591	514	8.70E-01	4.56E+00	4
ALL	10000	8178	8.18E-01	4.51E+00	-



(a) RPV Pressure

(b) Coolant Temperature



(c) Leakage Rate

(d) RPV Level

Figure 5. Analysis Results (Patterns 0, 4, 6)

Table 4 shows the priority order where scenarios are generated with uncertainties in the SGTR initial leakage rate from 40 to 60 [kg/s] and the case where the initial leakage rate is fixed at 40, 50, and 60 [kg/s], respectively. For all leakages, the top three priority patterns were 4, 3, and 5. In addition, the priority order when uncertainty is included is the same as when the leakage is fixed at 50 [kg/s], which is the middle value.

Thus, it was found that it was possible to classify scenarios into patterns according to whether or not and when response operations were performed, and to judge priorities of response operations based on risk information such as core damage probability and RPV water level. We also found that, by generating scenarios with various uncertainties, such as the scale of events occurring in plants and the timing of response operations, it was possible to analyze the effectiveness and characteristics of response operations considering the uncertainties. In summary, the results show that CMMC is an effective method to determine the priority of response operation in complex scenarios.

Table 4. Priority of Countermeasures (All Cases)

PATTERN	**CASE 1** (40 to 60[kg/s])	**CASE 2** (40[kg/s])	**CASE 3** (50[kg/s])	**CASE 4** (60[kg/s])
0	7	6	7	7
1	5	4	5	4
2	6	7	6	6
3	2	2	2	2
4	1	1	1	1
5	3	3	3	3
6	4	5	4	5

4. CONCLUSION

This paper proposed a countermeasure priority decision function using the CMMC method, a dynamic PRA method, to address the issue of the high burden of determining the priority of countermeasures to be taken when an event occurs due to the diversity and complexity of accident scenarios in the development of AM countermeasures and multi-unit operation.

We developed a simple single-unit plant dynamic characteristic analysis model for a trial evaluation of the countermeasure priority decision function. Then, we generated a number of scenarios with various uncertainties using a uniform distribution for the failure scenario of the SGTR breakage-side steam generator isolation and evaluated the response operations patterns. The results show that CMMC is an effective method to determine countermeasure priority in complex scenarios because it can evaluate the effectiveness of the response operations based on risk information considering uncertainty.

In the future, we plan to use the CMMC method to evaluate scenarios in which plant events and response operations are generated based on state transition probabilities depending on conditions that change over time, as well as scenarios in multi-units.

References

[1] S. Shinzaki, et al. Quantification of severe accident scenarios in level 2 PSA of nuclear power plant with continuous Markov Chain Model and Monte Carlo Method. 10th International Probabilistic Safety Assessment & Management Conference, PSAM10-132, 2010.

[2] T. Takata and E. Azuma. Event sequence assessment of deep snow in sodium-cooled fast reactor based on continuous Markov chain Monte Carlo method with plant dynamics analysis. Journal of Nuclear Science and Technology, 53, 11, 1749-1757, 2016.

[3] S. Shinzaki. Scenario quantification of dynamic level 2 PSA based on continuous Markov Monte Carlo method and plant system analysis code. Master thesis, Osaka University, 2012.