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STUDY ON EXTRACTION OF SAFETY IMPROVEMENT MEASURES UTILIZING SEISMIC PRA AND TSUNAMI PRA

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ABSTRACT

To enhance safety of nuclear power plants, it is effective to quantitatively understand the roles of equipment functions and operational management of those relevant to plant safety, through probabilistic risk assessment (PRA). In Japan, nuclear power plants that returned to operation after the Great East Japan Earthquake are required by regulation to conduct Safety Improvement Assessments within six months following the completion of periodic inspections and report the results to the Nuclear Regulation Authority.

In the Safety Improvement Assessment, a Probabilistic Risk Assessments (PRA) is required. At Sendai Nuclear Power Plant Unit 1, PRA has been conducted incorporating the latest plant design and operational information. Utilizing the risk insights obtained from the PRA, dominant accident scenarios and factors leading to core damage and loss of containment function are analysed, and additional safety enhancement measures have been extracted. By ensuring the implementation of these extracted measures, continuous safety enhancement at the nuclear power plant is achieved.

INTRODUCTION

At Sendai Nuclear Power Plant Unit 1, Safety Improvement Assessments have been continuously conducted since its restart in 2015. The first Safety Improvement Assessment (2017) was carried out in July 2017, where internal and external event PRA, reflecting the severe accident facilities established after the Fukushima accident, was conducted and the results were published. Subsequently, PRAs reflecting the added specific safety facilities was performed. In July 2022, during the fifth Safety Improvement Assessment, internal events Level 1 and Level 2 PRA was conducted, and in November 2023, during the sixth Safety Improvement Assessment (2023), Level 1 and Level 2 PRA for both seismic events and tsunami events were conducted, and safety improvement measures to be addressed in the future were extracted. This study presents the results of safety improvement measure extraction utilizing the risk insights obtained from the seismic and tsunami Level 1 and Level 2 PRA conducted in the first and sixth Safety Improvement Assessments.

PRA AND SAFETY IMPROVEMENT ASSESSMENT

In the Safety Improvement Assessment, after the completion of each periodic inspections that involve construction activities of safety measures, status of safety management activities is investigated along with the condition of the plant at the time of inspection completion. Based on the investigation results, PRA and safety margin evaluations are conducted to assess the effectiveness of the safety management activities and to extract additional safety improvement measures. With consideration of these assessment, a comprehensive evaluation is performed to establish a safety improvement plan. Figure 1 illustrates the implementation flow of the Safety Improvement Assessment. It should be noted that, in principle, the PRA is revised every five years, and revisions are also required in years where significant construction work, which is expected to change the results of the PRA, is undertaken.

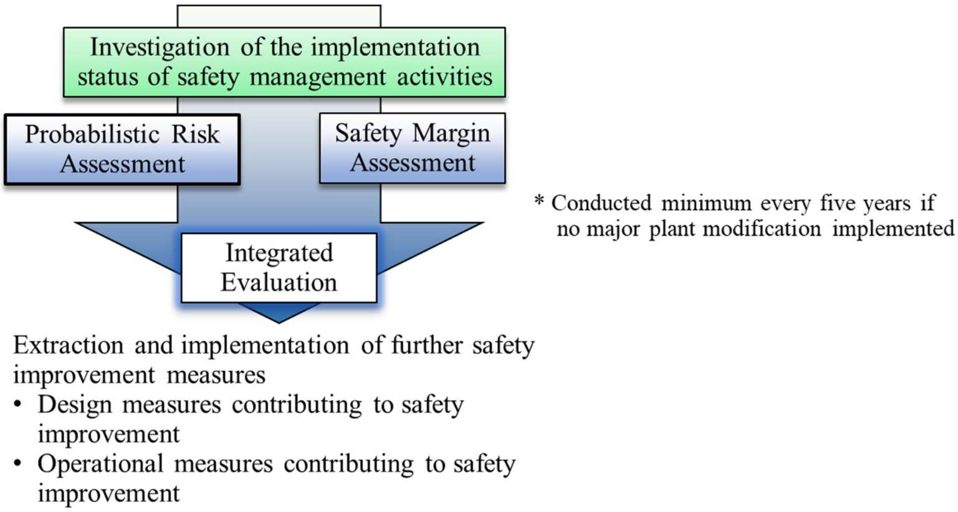

Figure 1. Implementation flow of Safety Improvement Assessment.

SEISMIC AND TSUNAMI PRA RESULTS OBTAINED DURING SAFETY IMPROVEMENT ASSESSMENT

Regarding the Level 1 and Level 2 PRAs for seismic events as well as those for tsunami events in the Safety Improvement Assessment of the Sendai Nuclear Power Plant Unit 1, PRAs reflecting severe accident managements were conducted in the first assessment (2017), and PRAs reflecting specific safety facilities were conducted in the sixth assessment (2023). The results of the seismic Level 1 and Level 2 PRA and tsunami Level 1 and Level 2 PRA are presented in Table 1.

Table 1: PRA results of seismic and tsunami events previously conducted.

Hazard group		**1st Assessment 2017 July 6**	**2nd Assessment 2019 Jan. 7**	**6th Assessment 2023 Nov.20**
Seismic events Level 1	CDF(/ry)	1.7E-06	8.5E-07	9.9E-07
Seismic events Level 2	CFF(/ry)	1.5E-06	6.3E-07	5.5E-07
Tsunami events Level 1	CDF(/ry)	1.0E-08	—	7.5E-09
Tsunami events Level 2	CFF(/ry)	9.2E-09	—	2.5E-09

CDF: Core Damage Frequency
CFF: Containment Failure Frequency

Additional Measures Extracted Through PRA In The First Safety Improvement Assessment

In the first Safety Improvement Assessment, seismic Level 1 and Level 2 PRA, as well as tsunami Level 1 and Level 2 PRA reflecting severe accident managements were conducted. Based on the evaluation results, dominant scenarios and their factors leading to core damage and containment failure were analysed, and additional measures for safety enhancement were evaluated. Specifically, from the perspective of contributions to core damage frequency (CDF) and containment failure frequency (CFF), Fussell-Vesely

(FV) importance was utilized to examine and extract additional measures for reducing significant contributors to risk.

Table 2 shows additional safety improvement measures extracted based on FV importances of seismic Level 1 and Level 2 PRA. Since FV importance results of seismic Level 1 and Level 2 PRA are nearly equivalent, consideration of additional measures was based on FV importances of the Level 1 PRA.

Regarding tsunami events, the CDF and CFF are approximately two orders of magnitude smaller than those of internal events or seismic events. In addition, improving capacity against tsunami of equipment with high FV importance would require intense construction activities, such as relocation of the equipment to higher ground levels. Therefore, no additional measures were extracted through tsunami PRA.

Table 2: Additional measures extracted through PRA in the first Safety Improvement Assessment.

Type of Enhancement	Additional Measure	Expected Effect	Hazards group
Design enhancement	Digitalization of metal-clad switchgear protection relay	Enhance seismic capacity and fragility	Seismic Level 1 and 2
Operational enhancement	Improve operations and procedures related to Component Cooling Water (CCW) inventory monitoring during seismic events	Early detection of CCW inventory reduction during seismic, thereby reducing the likelihood of loss of CCW events	Seismic Level 1 and 2
Education/Training	Strengthen training for monitoring CCW inventory to prevent loss of the CCW events during seismic	Early detection of CCW inventory reduction during seismic, thereby reducing the likelihood of loss of CCW events	Seismic Level 1 and 2

Seismic PRA Reflecting Additional Measures (Second Safety Improvement Assessment)

In the second Safety Improvement Assessment (2019), seismic Level 1 PRA and Level 2 PRA was updated reflecting the digitalization of the safety related metal-clad switchgear protection relay that has been extracted as addition measures in the first Safety Improvement Assessment and later incorporated into the plant. Figure 2 shows the protection relay of the metal-clad switchgear before digitalization. Table 3 presents a comparison of seismic fragility before and after the digitalization of the metal-clad switchgear. As a result of the improved fragility of the protection relay, the vulnerable point of the metal-clad switchgear shifted from the protection relay to the circuit breaker. Due to the improved fragility, CDF decreased from 1.7E-06 to 8.5E-07, and the CFF decreased from 1.5E-06 to 6.3E-07.

Figure 2. Metal-clad switchgear protection relay before digitalization.

Table 3: Seismic fragility of the metal clad switchgear before and after digitalization.

	Failure Mode	Subcomponent	Seismic Fragility				Vulnerable Point
			Median capacity (G)	β_R	β_U	HCLPF (G)	
Before	Structural failure	Base weld	9.56	0.19	0.21	5.01	
	Functional failure	Protection relay (Analog)	1.46	0.13	0.23	0.82	○
		Circuit breaker	1.84	0.08	0.15	1.26	
After	Structural failure	Base weld	9.56	0.19	0.21	5.01	
	Functional failure	Protection relay (Digital)	3.41	0.13	0.23	1.90	
		Circuit breaker	1.84	0.08	0.15	1.26	○

G: Dimensionless acceleration obtained by dividing acceleration by gravitational acceleration (9.8 m/s^2).
β_R: Logarithmic standard deviation of the capacity due to aleatory uncertainty.
β_U: Logarithmic standard deviation of the capacity due to epistemic uncertainty.
HCLPF: High Confidence of Low Probability of Failure. The seismic acceleration level corresponding to a 5% damage probability at a 95% confidence level, serving as a lower bound reference value for equipment damage.

ADDITIONAL MEASURES EXTRACTED FROM PRA IN THE SIXTH SAFETY IMPROVEMENT ASSESSMENT

In the sixth Safety Improvement Assessment (2023), Level 1 and Level 2 PRA of seismic events, as well as those of tsunami events, reflecting specific safety facilities were conducted. To further enhance safety of the current plant, dominant accident scenarios leading to core damage and containment failure were analysed along with their contributing factors, and additional measures for safety improvement. The specific process for examining these additional measures is outlined below.

Analysis of Accident Scenarios Leading to Core Damage and Containment Failure, and Consideration of Additional Measures

CDF for each accident sequence groups for the hazard groups (at-power operation seismic events and tsunami events) were analysed, along with the contribution of hazard group CDF to the total CDF. Additionally, the CFF for each containment failure mode of the hazard groups (at-power operation seismic event and tsunami event) and the contribution to the total CFF were analysed. Using the flow illustrated in Figure 3, accident sequence groups and containment failure modes subjected to consideration of additional measures were selected.

Using the severe accident management standard (2019) published by the Atomic Energy Society of Japan (hereinafter referred to as the "SAM Standard") as a reference, CDF for each accident sequence group and CFF for each containment failure mode were classified either "High", "Medium" or "Low" in terms of their importance. Additional safety improvement measures deemed effective were evaluated according to their classified importance. The classification results are presented in Tables 4 and 5. A list of extracted additional measures is shown in Table 7. Furthermore, since there were no items classified as "High," "Medium," or "Low" for the accident sequence group and containment failure modes of tsunami events, no additional measures were considered for tsunami events.

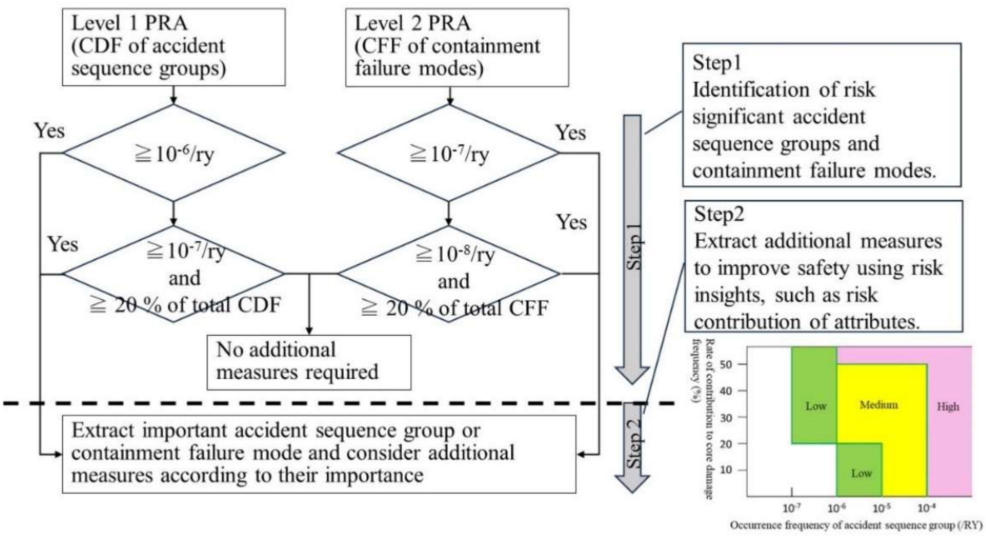

Figure 3. Flow of Extracting Safety Improvement Measures.

Table 4: Accident sequence group classification results of seismic and tsunami events.

Accident Sequence Group	Seismic Events		Tsunami Events	
	CDF (/ry)		CDF (/ry)	
Loss of secondary side heat removal	1.1E-07	11.0%	8.7E-11	1.2%
Total loss of AC power	8.9E-08	9.0%	2.5E-09	33.4%
Loss of component cooling water	**6.2E-07**	**62.6%**	4.8E-09	64.0%
Loss of containment heat removal	6.2E-11	< 0.1%	negligible	< 0.1%
Loss of reactor shutdown capability	3.7E-08	3.8%		
Loss of ECCS during injection mode	8.1E-08	8.2%	negligible	< 0.1%
Loss of ECCS during recirculation mode	1.5E-08	1.5%	negligible	< 0.1%
Direct damage to reactor building	negligible	< 0.1%		
Direct damage to containment vessel	3.3E-08	3.3%		
Multiple steam generator tube rupture	5.6E-09	0.6%		
Direct damage to multiple signals			1.1E-10	1.4%
Total	9.9E-07	100%	7.5E-09	100%

Red shaded: High importance, Yellow shaded: Medium importance, Green shaded: Low importance

Table 5: Containment failure mode classification results of seismic and tsunami events.

Containment Failure Mode	Seismic Events		Tsunami Events	
	CFF (/ry)		CFF (/ry)	
Steam explosion in reactor vessel (α)	negligible	< 0.1%	negligible	< 0.1%
Containment isolation failure (β)	**2.1E-07**	**38.3%**	3.3E-10	13.3%
Hydrogen combustion before reactor vessel breach (γ)	negligible	< 0.1%	negligible	< 0.1%
Hydrogen combustion after reactor vessel breach (γ’)	negligible	< 0.1%	negligible	< 0.1%
Hydrogen combustion late after reactor vessel breach (γ’’)	negligible	< 0.1%	negligible	< 0.1%
Over pressure due to steam and non-condensable gas accumulation (δ)	**3.0E-07**	**54.3%**	2.1E-09	86.3%
Base mat melt-through (ε)	1.1E-09	0.2%	5.8E-12	0.2%
Steam explosion outside reactor vessel (θ)	9.6E-10	0.2%	4.3E-13	< 0.1%
Steam explosion inside reactor vessel (η)	1.2E-10	< 0.1%	2.7E-12	0.10%
Direct heating od containment atmosphere (σ)	negligible	< 0.1%	negligible	< 0.1%
Steam generator tube rupture (g)	6.0E-09	1.1%	negligible	< 0.1%
Containment failure due to over-temperature (τ)	2.1E-10	< 0.1%	negligible	< 0.1%
Direct contact of debris with containment (μ)	negligible	< 0.1%	negligible	< 0.1%
Direct failure of containment due to seismic (χ)	3.3E-08	6.0%		
Total	5.5E-07	100%	2.5E-09	100%

Red shaded: High importance, Yellow shaded: Medium importance, Green shaded: Low importance

Level 1 Seismic Events PRA At-power

In the Level 1 seismic events at-power PRA, the core damage accident sequence groups were classified by their importance based on CDF contribution. As a result, "loss of component cooling water", which had the most highest importance among other accident sequences, was classified as "low" importance.

In the event of a seismic, damage to the seawater intake structure can lead to a total loss of seawater system, which, when combined with loss of offsite power, results in complete loss of component cooling water and total loss of alternating current (AC) power. When this sequence of events is followed by failure to control the flow rate by the turbine-driven auxiliary feedwater valve, which leads to the loss of the auxiliary feedwater system, eventually the core will be damage. Additionally, if the seismicity cause damage to the non-seismically qualified sections of the component cooling water system, which are relatively vulnerable against seismicity compared to other system structures and components (SSCs), and the damaged section cannot be isolated from the component cooling water system, the component cooling water system becomes inoperable.

Components and structures of the component cooling water system have a relatively higher probability of damage during seismic events compared to other SSCs, and may lead to loss of the component cooling inventory in the event of seismic induced failures. Therefore, it is considered effective in terms of risk reduction to introduce enhanced monitoring strategies of the component cooling water system to facilitate early detection of leaks during seismic events and to isolate the leakage. Continuous education and training, as well as incorporation of the content into procedures will be undertaken to prevent loss of component cooling water by enhancing reliability to isolate leaks through early leak detection.

Level 2 Seismic Events PRA At-power

In the Level 2 seismic events at-power PRA, the containment failure mode of "Over-pressure due to accumulation of steam/non-condensable gas (δ mode)" was classified as "High" importance, while "Containment isolation failure (β mode)" was classified as "Medium" importance. Regarding the δ mode, a typical scenario is which the seismic event causes loss of offsite power and also loss of component cooling seawater system due to seismic damage to the seawater intake structure. Loss of component cooling water seawater system results in the failure of emergency diesel generators, which is cooled by seawater system, and eventually result in loss of all AC power if offsite power in not available. If this sequence is followed by failures to start the large-capacity air-cooled power generator, alternative containment spray will be inoperable, and all containment heat removal functions, including containment heat removal by natural convection, will be lost. The SSCs of the offsite power system and component cooling water system are relatively vulnerable against seismicity when compared to other SSCs, and damage to these components may result in total loss of both AC power and component cooling water. In this accident sequence, the large-capacity air-cooled generator and the alternate power source of the specific safety facilities are relied upon to mitigate the accident. However, component failure rates of these types of power generators are not available so substitute reliability parameters taken from similar components are applied in the PRA. By collecting and reflecting operational performance of these components to develop reliable and realistic component reliability parameters, a more realistic risk analysis can be achieved, and robust risk insights can be gained from the PRA. Therefore refinement of the PRA has be chosen as safety improvement measure.

Representative scenario of containment isolation failure (β mode) is that initiated by seismic damage to the auxiliary reactor building, leading to damage to various equipment within the building failing plant monitoring instruments. Degradation in the plant monitoring function is assumed to result in failure of containment isolation function upon core damage. Although this scenario becomes a dominant contributor to the CFF in the high seismic acceleration categories, the PRA assumes complete failure of the building making it difficult to refine the model to evaluate a realistic plant response.

As shown in Table 6, an analysis was conducted for each seismic acceleration category using the conditional containment failure probability (CCFP) given seismic event. In acceleration category 1, under conditions where the offsite power supply is intact and AC power available, the pressure within the containment at core damage is still low since the reactor coolant boundary is likely to be intact. This leads to a situation where the containment spray signal will not activate at core damage, and prevent the isolation valves that actuate only upon containment spray signal to close in a timely manner, resulting in the likelihood of containment isolation failure. Therefore, it was decided to add procedures to manually close specific containment isolation valves even when the containment spray signal is not activated. This change is expected to reduce the risk from containment isolation failure (β mode). A list of the extracted additional safety measures is presented in Table 7.

Table 6: CDF, CFF and CCFP of each seismic acceleration category

Seismic Acceleration Category	Mean Frequency of Seismicity (/year)	CDF (/ry)	CFF (/ry)	Conditional Containment Failure Probability CCFP
Category 1 (0.2 G to 0.4 G)	8.9E-04	2.7E-07	7.7E-08	2.9E-01
Category 2 (0.4 G to 0.6 G)	1.0E-04	7.8E-08	1.8E-08	2.3E-01
Category 3 (0.6 G to 0.8 G)	2.3E-05	2.0E-08	4.0E-09	2.0E-01
Category 4 (0.8 G to 1.0 G)	6.5E-06	2.1E-08	3.6E-09	1.7E-01
Category 5 (1.0 G to 1.2 G)	2.2E-06	1.2E-07	4.1E-08	3.4E-01
Category 6 (1.2 G to 1.4 G)	8.1E-07	4.8E-07	3.8E-07	7.9E-01

Table 7: Additional measures extracted through PRA in the sixth Safety Improvement Assessment.

Type of Enhancement	Additional Measure	Expected Effect	Hazards group
Operational enhancement	Addition procedures to close specific containment isolation valves when containment spray signal not activated	Reduction of risk from containment isolation failure (β mode) in situations when containment spray signal is not activated at core damage	Seismic Level 2
Education/Training	Strengthen training for monitoring CCW inventory to prevent loss of the CCW events during seismic	Early detection of CCW inventory reduction during seismic, thereby reducing the likelihood of loss of CCW events	Seismic Level 1
PRA Model Refinement	Refine component reliability parameters (Continuous collection and reflection of operational data, particularly for component types not covered by Japanese generic component reliability parameter database, such specific safety facility power generators)	Reduction of uncertainties associated with the PRA to achieve more realistic risk analysis	Seismic Level 2

CONCLUSION

By utilizing the risk insights obtained from the PRA of the Safety Improvement Assessment, effective safety enhancement measures have be extracted. The purpose of the PRA in the Safety Improvement Assessment is to understand the vulnerabilities of nuclear power plants based on the risk analysis and to extract effective additional measures contributing to safety enhancement. Therefore, there is a need to refine the PRA by increasing realism and eliminating conservatisms embedded in the model, which will involve incorporation of domestic and international operating experiences and state of the art PRA methodologies. In risk informed decision-making (RIDM) in nuclear power plant operation, it is necessary to make decisions not only from the PRA but also utilising various risk information. By promoting and establishing the process of RIDM, Kyushu Electric Power will continue to strive for safety enhancement of the nuclear power plants.

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