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OVERVIEW OF SEISMIC PRA AND TSUNAMI PRA FOR SAFETY IMPROVEMENT ASSESSMENT ON SENDAI NUCLEAR POWER STATION

Hirotaka Yano[1], Kentaro Kawazu[2], Daigo Hiratsuka[3], and Mitsuharu Sakai[4]

[1]Staff, Kyushu Electric Power Co., Inc., Fukuoka, Japan (hirotaka_yano@kyuden.co.jp)
[2]Staff, Kyushu Electric Power Co., Inc., Fukuoka, Japan (kentaro_kawazu@kyuden.co.jp)
[3]Deputy Manager, Kyushu Electric Power Co., Inc., Fukuoka, Japan(daigo_hiratsuka@kyuden.co.jp)
[4]Chief Manager, Kyushu Electric Power Co., Inc., Fukuoka, Japan(mitsuharu_sakai@kyuden.co.jp)

ABSTRACT

In response to the lessons learned from the Fukushima Daiichi Nuclear Power Plant accident, a safety improvement assessment notification system has been introduced with the aim of continuously enhancing the safety of nuclear power plants. This safety improvement assessment notification system mandates that, within six months following the completion of the periodic operator inspection, electric utilities are required to evaluate the implementation status and effectiveness of their safety enhancement initiatives of their nuclear power plants. Electric utilities must summarize the results of the safety improvement assessment and report to the Nuclear Regulation Authority.

In the safety improvement assessment of the Sendai Nuclear Power Plant Unit 1, Probabilistic Risk Assessment (PRA) was utilized to quantify and analyse the risks associated with the plant, and identify measures that contribute to safety enhancement. To prioritize the additional measures that can be taken to effectively mitigate accident scenarios contributing to risk, the PRA results were investigated. This activity included investigation of the core damage frequencies (CDF) of the accident sequence groups of various hazards and operation modes (internal events during at-power and shutdown operation, seismic events during at-power operation, and tsunami events during at-power operation), as well as the contribution of each hazard group to the total CDF. In addition, Containment Failure Frequency (CFF) of each containment failure mode for the various hazards and operation modes was assessed, along with the relative contribution of the CFFs to the total CFF. Based on this analysis, accident sequence groups and containment failure modes subjected to consideration for additional safety measures were selected. In making this selection, reference was made to the "Implementation Standard Concerning Preparation, Maintenance and Improvement of Severe Accident Management in Nuclear Power Plants: 2019" published by the Atomic Energy Society of Japan (AESJ).

Level 1 and Level 2 at-power PRA for seismic events and tsunami events have been utilized in the first, fourth (Level 2 only), and sixth safety improvement assessment of Sendai Nuclear Power Plant Unit 1. In the sixth safety improvement assessment, the updated PRA model reflecting the design and operational changes of the plant from the first safety improvement assessment, including newly installed specific safety facilities was used. A refined human reliability analysis methods was also incorporated in the updated PRA used.

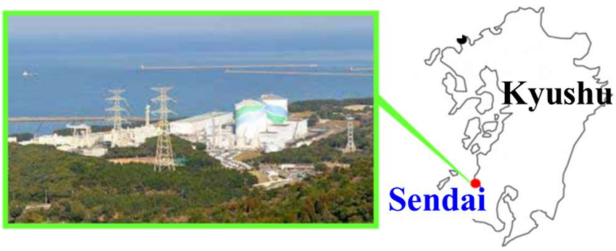
This paper provides a comparison of the PRA results of the first and sixth safety improvement assessment of the Sendai Nuclear Power Plant Unit 1, focusing on the Level 1 and Level 2 at-power PRA for seismic events and tsunami events. Discussions regarding the impacts of the design, operation and model updates on the PRA results are also provided.

SENDAI NUCLEAR POWER PLANT UNIT 1

The Sendai Nuclear Power Plant Unit 1 is a three-loop pressurized water reactor (PWR) located in Kagoshima Prefecture, south of Kyushu, Japan. It commenced commercial operations in 1984 as the third nuclear power plant in Kyushu. The plant was temporally shut down in 2011 following the Fukushima Daiichi Nuclear Power Plant accident; however, it successfully passed the review and pre-operational inspections under the post-Fukushima nuclear regulation in 2015, and become the first Japanese nuclear power plant to restart after the Fukushima Daiichi Nuclear Power Plant accident. Since then, the plant has continued to operate safely and stably. An overview of the Sendai Nuclear Power Plant and its main topics are presented in Figure 1.

Plant Specification

Electrical output	890 MW
Reactor type	Pressurized Water Reactor (PWR)
Containment vessel	Large dry containment



Major activities

July , 1984	Operation start
May , 2015	License approval under post-Fukushima regulatory standards
September , 2015	Passed Pre-use inspection (restart after Fukushima accident)
July , 2017	1st Safety Improvement Assessment submittal (Internal events at-power and shutdown, seismic, and tsunami PRA)
March , 2020	Licensing of Specialized Safety Facility completed.
November , 2020	Passed pre-use inspections of Specialized Safety Facility (re-operation)
July , 2022	5th Safety Improvement Assessment (Internal events at-power PRA)
November , 2023	6th Safety Improvement Assessment (Shutdown, seismic and tsunami PRA)

Figure 1. Overview of the Sendai Nuclear Power Plant unit 1.

EVOLUTION OF THE AT-POWER SEISMIC EVENTS AND TSUNAMI EVENTS PRA

The PRA developed for the first safety improvement assessment for Sendai Nuclear Power Plant Unit 1was based on the plant design and operation that complies with the new regulatory standards, in which the post-Fukushima severe accident management measures have been required. The PRA model for the sixth safety improvement assessment included updated design and operation, operational experience information, and latest knowledge and PRA assessment methods. PRA update enabled the safety improvement assessment to accurately reflects the current state of the plant. Main changes to the PRA incorporated after the first safety improvement assessment is summarized in Table 1.

The specific safety facility is a facility designed to prevent the containment failure in the event of a substantial core damage accident scenario, such as intentional large aircraft crash or other sabotage, that may disable the core cooling function. Schematic of the specific safety facility is presented in Figure 2.

Table 1: PRA model changes between the first and sixth safety improvement assessment.

Item	First Assessment	6th Assessment	PRA model changes
Initiating Events	—	Expanded set of initiating event	Added and refined initiating event grouping based on FMEA and additional assessment.
Component Reliability	Generic component reliability (JANSI (2009) data)	Plant specific component reliability	Use plant specific reliability data estimated by Bayesian updating the latest Japanese generic data (Yoshida (2021)) with operational experience of Sendai Unit 1 and 2.
Human Reliability Analysis (HRA)	THERP method (NUREG/CR-1278)	EPRI HRA Calculator	Adopt standard HRA method used in US. Evaluate human error probability with consideration of time windows base on operator interviews.
Specific safety facility	Not modelled (not implemented at plant)	modelled	Modelled and credited specific safety facility.
Seismic acceleration range	Up to 1.2G	Up to 1.4G	Increase the upper range of seismic acceleration to 1.4G.
Seismic hazard	—	Updated	Utilize results of refined seismic probabilistic hazard analysis.

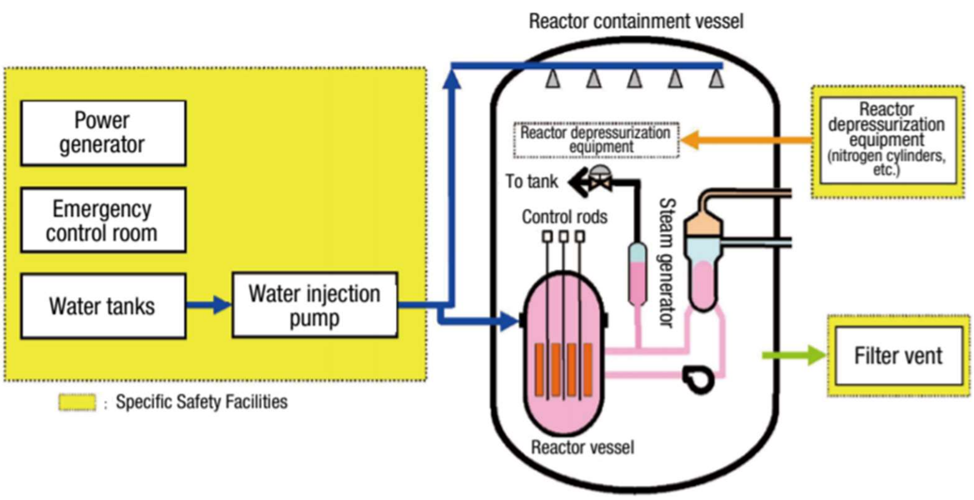


Figure 2. Schematic of the specific safety facility.

SEISMIC PRA FOR AT-POWER OPERATION

The seismic PRA for at-power operation was conducted with reference to the "A Standard for Procedure of Seismic Probabilistic Risk Assessment for Nuclear Power Plants: 2015" published by the AESJ, as well as the Level 2 PRA also published by AESJ.

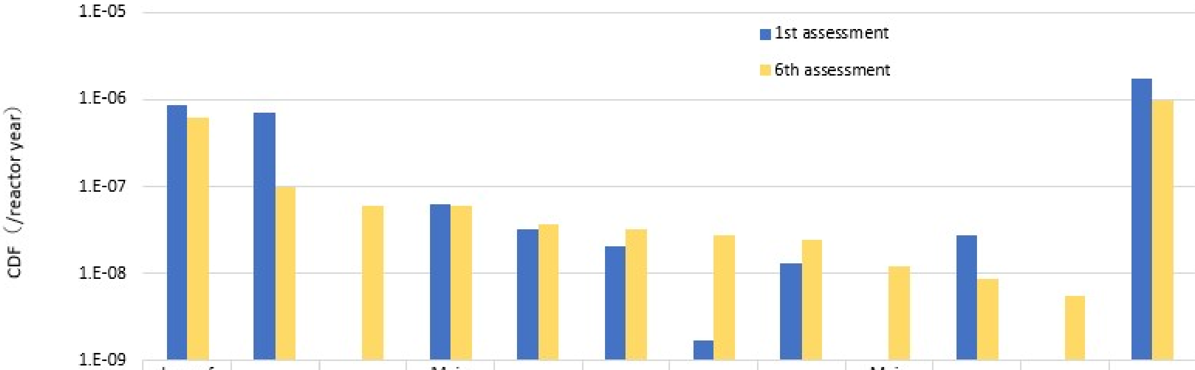
The seismic PRA study for the first safety improvement assessment indicated a CDF of 1.7×10^{-6} (/reactor-year) and a CFF of 1.5×10^{-6} (/reactor-year). In the sixth safety improvement assessment, the installation of the specific safety facility power generator acted as an efficient measure to mitigates accident scenarios involving loss of power supply and led reductions in both CDF and CFF, resulting in a CDF of 9.9×10^{-7} (/reactor-year) and a CFF of 5.5×10^{-7} (/reactor-year).
The impact of the PRA model updates from the first to the sixth assessment on the seismic PRA results is summarized in Table 2.

Table 2: Impact of the PRA model updates to the level 1 and 2 seismic PRA results.

PRA Model Change	Impact on PRA results
Initiating Events	Refined set of initiating events enabled detailed modelling of seismic induced Partial Loss of Component Cooling Water (loss of C header) accident scenario. However, impact on results was minor.
Component Reliability	Use of a different data source for the generic component reliability data resulted in increase to the failure probabilities of components relied upon during loss of offsite power events, and led to increases in CDF and CFF.
Human Reliability Analysis	The HRA calculator, when compared with THERP method, estimates higher human error probabilities for actions with short available time for operators. Accident scenarios at high seismic acceleration bins have short available time for actions due to the degraded accessibility caused by seismic, and therefore, the use of HRA calculator tends to increase CDF and CFF.
Specific safety facility	The power generator in the specific safety facility provides an effective accident mitigation function in the containment failure scenario where the air cooled power supply, reducing the CDF and CFF.
Seismic Acceleration Range	The upper range of the seismic acceleration subjected to risk quantification was elevated from 1.2G to 1.4G, resulting an increase to the quantified CDF and CFF.
Seismic hazard	The refined seismic hazard resulted in a lower seismic exceedance frequency within the seismic acceleration range of the PRA, resulting in increases to the CDF and CFF.

To prioritize additional measures effective against accident scenarios with high risk contributions, the CDF results were to organized according to the seismic initiating events that led to core damage. The results are shown in Figure 3.

In both the first and sixth safety improvement assessments, it was found that the CDF associated with the total loss of component cooling water system had the largest contribution to seismic CDF. The PRA for the sixth safety improvement assessment showed a reduction in the total CDF compared to the first assessment, due to the installation of the specific safety facility power supply that is effective to mitigate core damage accident scenario that involve seismic failure of the large-capacity air-cooled power generator.

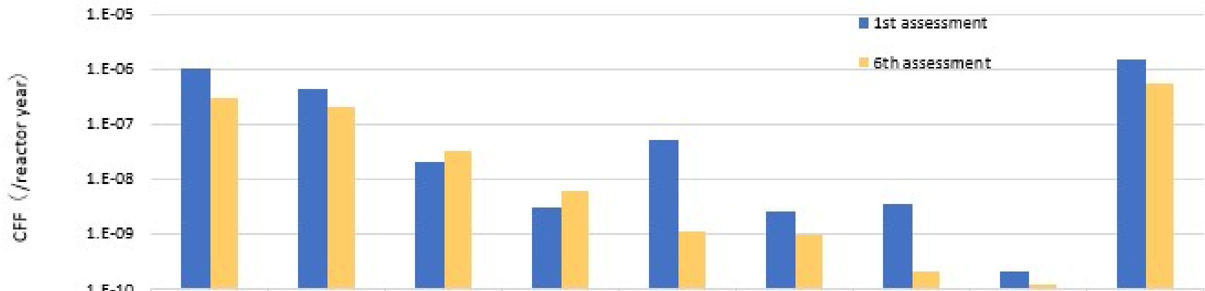


	Loss of component cooling water (CCW)	Loss of offsite power	Partial loss of CCW	Main steamline break (upstream MSIV)	ATWS	Containment failure	Large break LOCA	Loss of main feedwater flow	Main steamline break (downstream MSIV)	Excessive LOCA	Steam Generator Rupture	Total
1st assessment	8.7E-07	7.0E-07	<1E-10	6.3E-08	3.2E-08	2.1E-08	1.7E-09	1.3E-08	<1E-10	2.8E-08	<1E-10	1.7E-06
6th assessment	6.2E-07	1.0E-07	6.0E-08	5.9E-08	3.7E-08	3.3E-08	2.7E-08	2.4E-08	1.2E-08	8.9E-09	5.6E-09	9.9E-07

Figure 3. At power CDF contributions per seismic induced initiating events.

Next, seismic CFF was quantified for each containment failure modes. CFF contributions of seismic events based on containment failure modes are presented in Figure 4.

In both the PRA study of the first and sixth safety improvement assessments, the dominant containment failure modes were found to be containment overpressure due to the accumulation of steam and non-condensable gas and containment isolation failure. In the sixth safety improvement assessment, a reduction in the seismic CFF from the first safety improvement assessment was observed, as a result of the installation of the specific safety facility power supply that is effective to mitigate containment failure accident scenarios involving seismic failure of the large-capacity air-cooled power generator.



	Containment overpressure due to gas accumulation	Containment isolation failure	Seismic containment failure before core damage	Steam generator rupture	Basemat melt-through	Containment failure before core damage due to overpressure	Containment failure due to over-temperature	Steam explosion	Total
1st assessment	9.9E-07	4.5E-07	2.1E-08	2.9E-09	5.1E-08	2.6E-09	3.6E-09	2.1E-10	1.5E-06
6th assessment	3.0E-07	2.1E-07	3.3E-08	6.0E-09	1.1E-09	9.6E-10	2.1E-10	1.2E-10	5.5E-07

Figure 4. At-power seismic CFF contributions per containment failure modes.

TSUNAMI PRA FOR AT-POWER OPERATION

The Tsunami PRA for at-power operation was conducted with reference to the "Implementation Standard Concerning the Tsunami Probabilistic Risk Assessment of Nuclear Power Plants: 2016" published by AESJ, as well as the Level 2 PRA standards also published by AESJ.

The Tsunami PRA study for the first safety improvement assessment indicated a CDF of 1.0×10^{-8} (/reactor-year) and a CFF of 9.2×10^{-9} (/reactor-year). In the sixth safety improvement assessment, the installation of the power generator of the specific safety facility acted as an efficient measure to mitigates accident scenarios involving loss of power supply and led reductions in both CDF and CFF, resulting in a CDF of 7.5×10^{-9} (/reactor-year) and a CFF of 2.5×10^{-9} (/reactor-year).

The impact of the Tsunami PRA model updates from the first to the sixth assessment on the tsunami PRA results is summarized in Table 3.

Table 3: Impact of the PRA model updates to the level 1 and 2 tsunami PRA results.

PRA Model Change	Impact on PRA results
Component Reliability	Use of a different data source for the generic component reliability data resulted in an increase to the failure probability of turbine driven auxiliary feedwater pump, and led to increase in CDF.
Human Reliability Analysis	The HRA calculator, when compared with THERP method, estimates higher human error probabilities for actions with short available time for operators. Since the available time for operator actions to cope with loss of component cooling water sea water system, which is an risk significant accident for tsunami event, is relatively short, the use of HRA calculator results in high human error probabilities in these accident scenarios and tends to increase the CDF.
Specific safety facility	The power generator in the specific safety facility provides an effective accident mitigation function in the containment failure scenario where the air cooled power supply, reducing the CDF and CFF.

To prioritize additional measures effective for accident scenarios with significant risk contributions, the CDF results were to organized according to the tsunami induced initiating events that led to core damage. The results are shown in Figure 5.

In both the first and sixth safety improvement assessments, it was found that the CDF associated with the tsunami induced initiating events of loss of component cooling water sea water system and loss of offsite power have large contribution to tsunami CDF. The PRA for the sixth safety improvement assessment showed a reduction in the total CDF compared to the first assessment, due to the addition of the specific safety facility power supply effective to mitigate the core damage accident scenario involving tsunami induced failure of the large-capacity air-cooled power generator.

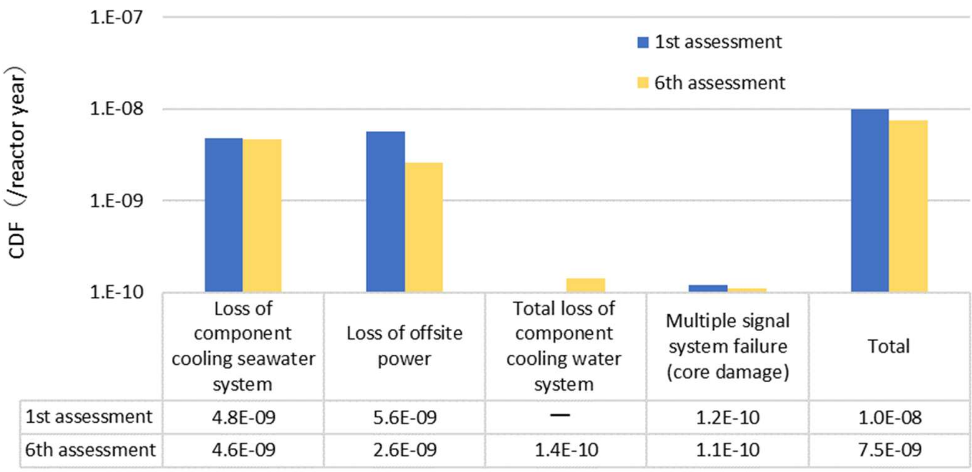


Figure 5. At power CDF contributions per tsunami induced initiating events.

Next, tsunami CFF was quantified for each containment failure modes. CFF contributions of tsunami events based on containment failure modes are presented in Figure 6.

In both PRA studies of the first and sixth safety improvement assessments, dominant containment failure modes were found to be containment overpressure due to the accumulation of steam and non-condensable gas. As a result of the addition of the specific safety facility power supply, which is effective to mitigate the containment failure accident scenario involving tsunami induced failure of the large-capacity air-cooled power generator, a reduction in the tsunami CFF was observed in the sixth safety improvement assessment.

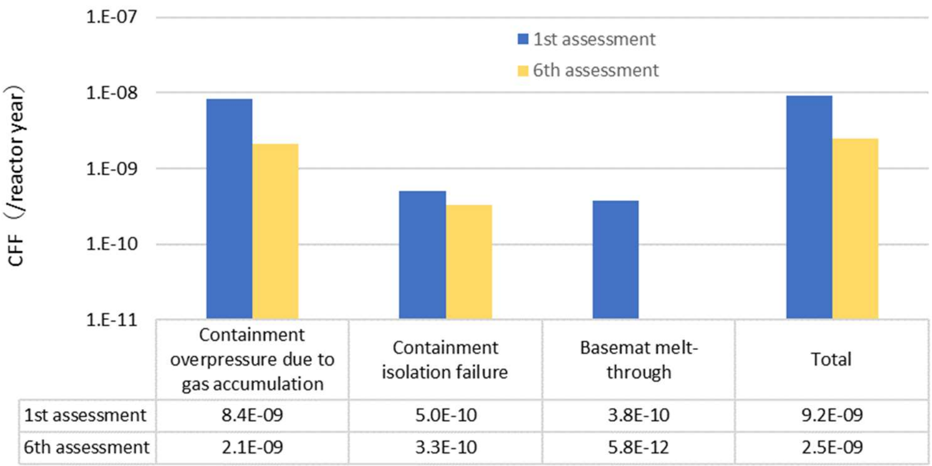


Figure 6. At-power tsunami CFF contributions per containment failure modes.

CONCLUSION

Comparison of PRA results of seismic event at-power (Level 1 and Level 2) and tsunami events at-power (Level 1 and Level 2) from the first and sixth safety improvement assessment for Sendai Nuclear Power

Plant Unit 1 was performed. It has been confirmed that the installation of the specific safety facility power supply has significantly contributed to the reduction of both the CDF and the CFF.

In order to utilize the risk insights derived from the PRA to improve design and operation of nuclear power plants, it is essential to engage in development and researches to achieve more realistic risk evaluations. Therefore, Kyushu Electric Power will continue to actively pursue these initiatives.

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