Extraction of Additional Measures based on Safety Improvement Assessment

Kentaro Kawazu[a*], Daigo Hiratsuka[a], Masato Nishimura[a], Yoshihiro Ide[a], Tomomi Uchida[a], Kouichi Ohta[b], Osamu Saramoto[b], Shoichi Nose[b]

[a]Kyushu Electric Power Co.,Inc, 1-82, Watanabe-dori 2-Chome, Chuo-ku, Fukuoka, Japan
[b]Nishimu Electronics Industries Co.,Ltd, 1-2-1, Minoshima, Hakata-ku, Fukuoka, Japan

Abstract: To improve the safety of nuclear power plants, it is useful to develop quantitative understanding through PRA, the role of equipment functions and operation management that assure safety of the nuclear power plant. In accordance with Japanese law and regulations, nuclear power plants that have restarted operations after the year of great east Japan earthquake are required to conduct a safety improvement evaluation within six months after the completion of each periodic facility inspection, and report to the Nuclear Regulation Authority by due date. As part of the safety improvement evaluation for Sendai nuclear power station unit 1, internal event PRA (At-power operating mode and shutdown mode), seismic event PRA (At-power operating mode), and tsunami PRA (At-power operating mode) were developed based on the latest plant design and operation, with consideration of facilities for dealing with specific severe accidents, etc. Based on the results of the PRA, dominant accident scenarios and factors leading to core damage and loss of containment function were analysed to derive additional measures to improve safety.

Keywords: Safety Improvement Assessment, Safety Improvement Measures

1. INTRODUCTION

Probabilistic risk assessment (PRA) of nuclear power plants take into consideration of abnormal events that can possibly occur and evaluates the probability of subsequent progression of events based on plant specific system configuration and reliability of the system structures and components (SSCs) as well as human actions. Understanding the roles of system functions and operations that assure safety of nuclear power plants, using quantitative insights gained through PRA, has an important role in improving safety.

Regarding past safety improvement assessment of Sendai Nuclear Power Station Unit 1 (hereinafter referred to as “Sendai 1”) that has been reported to the regulatory body, the fifth safety improvement assessment incorporated updated plant design and operational data into the PRA model for internal events at-power. In addition, the sixth safety improvement assessment, which involved the update of the internal events at-power PRA, included an evaluation of internal event shutdown PRA (Level 1), seismic events at-power PRA (Levels 1 and 2), and tsunami at-power PRA (Levels 1 and 2). These assessment results of the sixth safety improvement assessment were used to analyse dominant accident scenarios and risk important factors contributing to core damage and containment failure to investigate additional safety improvement measures to further improve safety of Sendai 1.

The PRA results were organized to prioritize additional safety measures effective to mitigate accident scenarios with high contribution to risk. With regards to Level 1 PRA, the core damage frequency (CDF) contribution of each accident sequence group within each hazard groups and plant operating mode (internal events at-power, internal events during shutdown, seismic events at-power, and tsunami at-power), and the risk contribution of each hazard group and operating mode to the overall CDF were analysed. And with regards to Level 2 PRA, the containment failure frequency (CFF) contribution of each containment failure mode in each hazard groups (internal events at-power, seismic events at-power, tsunami at-power) and the risk of contribution hazard groups to the overall CFF were analysed. Thus, important accident sequence groups and containment failure modes subjected to consideration for additional safety measures were selected. This selection process refers to the severe accident management standard [1] published by the Atomic Energy Society of Japan (hereinafter referred to as the “SAM Standard”).

2. INTERNAL AND EXTERNAL EVENT PRA

2.1. Internal events at-power PRA

Internal events at-power PRA was developed using the Atomic Energy Society of Japan (AESJ)'s Level 1 PRA standard [2]and Level 2 PRA standard [3] (hereinafter referred to as the "Level 2 PRA Society Standards"). The quantification results showed an overall CDF to be 3.0×10^{-6} (/reactor year [RY]). Figure 1 shows CDF contribution from each initiating event. The initiating event with the highest contribution was small break LOCA, which accounted for approximately 30% of the overall CDF, and the second and third largest contributors were respectively medium break LOCA and loss of offsite power.

Contribution of each containment failure modes to the at-power internal events CFF is shown in Figure 2. Of the overall CFF, 36.8% was from steam generator tube rupture (g mode), 30.3% from over-pressure rupture due to accumulation of steam/non-condensable gas (δ mode), 18.6% from containment isolation failure (β mode), 11.3% from interface system LOCA (v mode), and 2.2% from containment failure before core damage due to steam accumulation (θ mode). Contributions from all other containment failure modes were respectively approximately 1% or less.

2.2. Internal events shutdown PRA

The Level 1 internal events shutdown PRA was developed using the Atomic Energy Society of Japan (AESJ)'s Shutdown PRA standard [4]. This assessment was conducted using the shutdown evolution and plant configurations planned during the initial planning phase of the 27th refuelling outage of Sendai 1 as a base case. The results of accident sequence quantification showed that the overall CDF was 2.1×10^{-5} (/RY). Figure 3 presents CDF contribution from each initiating events.

Using the risk insights gained from the base case shutdown PRA, the shutdown evolution and plant configuration of the 27th refuelling outage of Sendai 1 was modified to reduce risk, and evaluated as sensitivity analysis case. The reduction in risk in the finalized shutdown evolution plan was confirmed. Compared to the overall CDF based on the initially planned shutdown evolution and plant configuration (2.1×10^{-5} (/RY)), the overall CDF evaluated of the finalized shutdown evolution plan was 1.2×10^{-6} (/RY), resulting in approximately 94% reduction of CDF. Details of the development of the Shutdown PRA model are presented in the Nishimu Electronics Industries co., LTD's paper[5].

2.3. Seismic events at-power PRA

Seismic events at-power PRA was developed using the seismic PRA standard [6] published by the Atomic Energy Society of Japan and Level 2 PRA Society Standards [3]. The overall CDF from at-power seismic events was 9.9×10^{-7} (/RY). Contribution of risk from each seismic initiating events to the overall at-power seismic CDF is shown in Figure 1. The results organized by initiating event showed that the total loss of component cooling water being most dominant initiating event, accounting for 62.6% of the overall at-power seismic CDF.

Contribution of each containment failure modes to the at-power seismic CFF is shown in Figure 2. The overall CFF was 5.5×10^{-7} (/RY). Of the containment failure modes, over-pressure rupture due to accumulation of steam/non-condensable gas (δ mode) and containment isolation failure (β mode) were the dominant risk contributors with CFF of 3.0×10^{-7} (/RY) and 2.1×10^{-7} (/RY), respectively.

2.4 Tsunami at-power PRA

Tsunami events at-power PRA was developed using the tsunami PRA standard [7] published by the Atomic Energy Society of Japan and Level 2 PRA Society Standards [3]. The overall CDF from tsunami at-power was 7.5×10^{-9} (/RY). The contribution of risk from each initiating events to the overall tsunami CDF is shown in Figure 1. Of the overall CDF, approximately 62.0% was from losses of component seawater system and approximately 34.6% was from loss of offsite power. These two initiating events accounted for majority of the overall CDF.

Contribution of each containment failure modes to the at-power tsunami CFF is shown in Figure 2. The overall CFF was 2.5×10^{-9} (/RY). Of the containment failure modes, the CFF of over-pressure rupture due to accumulation of steam/non-condensable gas (δ mode) was dominant with CFF of 2.1×10^{-9} (/RY), contributing more than 80% of the overall tsunami CFF. The second dominating failure mode was containment isolation failure (β mode) with CFF of 3.3×10^{-10} (/RY), contributing approximately 13.3% of the overall tsunami CFF.

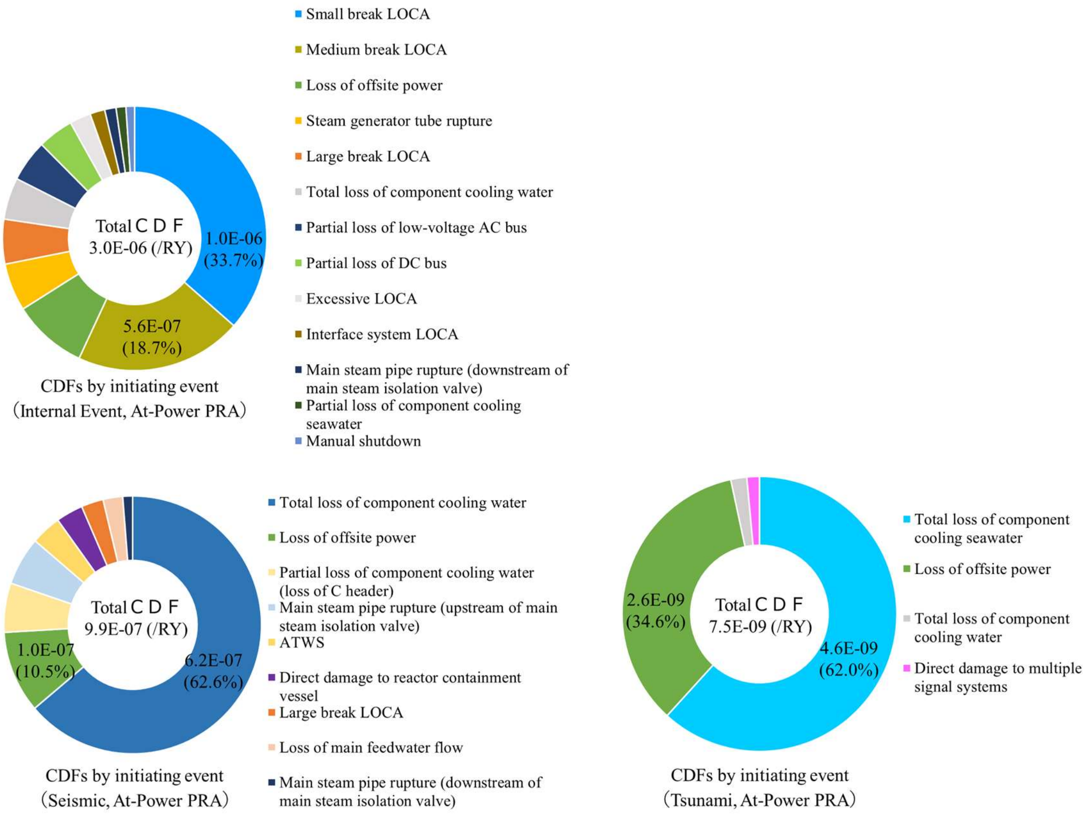

Figure 1. CDFs by initiating events (at-power PRA)

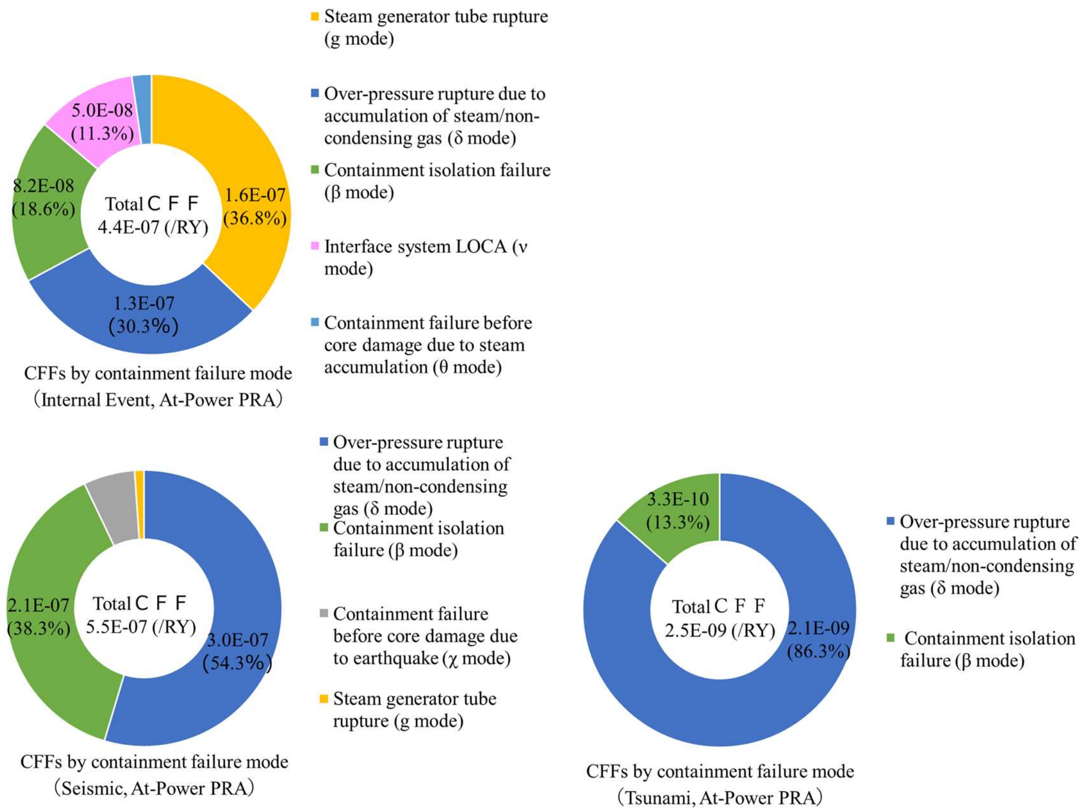

Figure 2. CFFs by containment failure modes (at-power PRA)

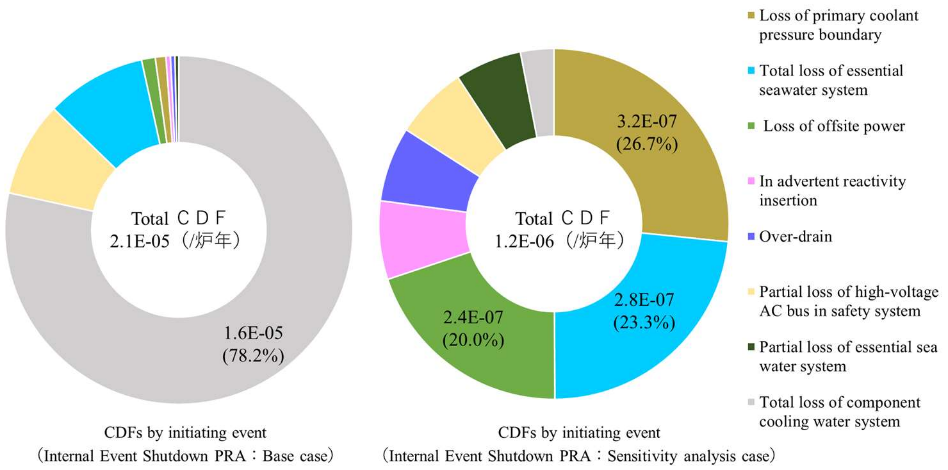

Figure 3. CDFs by initiating events (internal events shutdown PRA)

3. ADDITIONAL MEASURES SELECTED THROUGH PRA

Based on the accident sequence quantification of individual events, additional safety improvement measures were investigated through analysis of dominant accident scenarios and the factors involved therein leading to core damage and/or containment failure, for the purpose of improving safety. The process followed to investigate additional safety improvement measures is described below.

3.1. Analysis of Typical Accident Scenarios Leading to Core Damage and/or Containment Failure and Consideration of Additional Measures

An analysis was conducted on CDFs of each accident sequence group for each hazard group and plant operating mode (internal events at-power, internal events during shutdown, seismic events at-power, and tsunami at-power), and the risk contribution of each hazard group and operating mode to the overall CDF. With regards to CFF, CFF of each containment failure mode for each hazard group (internal events at-power, seismic events at-power, and tsunami at-power), and risk contribution of each hazard group to the overall CFF was analysed. Subsequently, using the criteria illustrated in Figure 4, accident sequence groups and containment failure modes to be considered for the evaluation of additional safety improvement measures were selected.

Using the SAM Standard as a reference, CDF for each accident sequence group and CFF for each containment failure mode selected for consideration of additional measures were classified either “High”, “Medium” or “Low” significance levels. Additional safety improvement measures potentially effective to reduce risk from the accident sequence groups and containment failure modes were evaluated according to their significance level classification. The classification results are shown in Tables 1 and 2. For seismic events at-power, analyses were conducted for each seismic acceleration categories using conditional containment failure probabilities as shown in Table 3. The additional safety improvement measures selected through this process are listed in Table 4 and 5.

Significance level	screening criteria
High	・10^{-4}/RY or greater（Occurrence frequency of accident sequence group） ・10^{-6}/RY or greater（Occurrence frequency of accident sequence group） and 50% or more of overall CDF
Medium	・The occurrence frequency of the accident sequence group is 1.0×10^{-5}/reactor years or more and less than 1.0×10^{-4}/reactor years, and the contribution ratio is less than 50% of the total CDF ・The occurrence frequency of the accident sequence group is 1.0×10^{-6}/reactor year or more and the contribution ratio is 20% or more and less than 50% of the total CDF
Low	・The occurrence frequency of the accident sequence group is 1.0×10^{-6}/reactor years or more and less than 1.0×10^{-5}/reactor years, and the contribution ratio is less than 20% of the total CDF ・The occurrence frequency of the accident sequence group is 1.0×10^{-7}/reactor years or more and less than 1.0×10^{-6}/reactor years, and the contribution ratio is 20% or more of the total CDF

*1：For containment failure modes, x-axis values are smaller by one order of magnitude.
*2：SAM Standards[1] published by the Atomic Energy Society of Japan lists operator’s responses for criticality levels “High,” “Medium,” and “Low.” Responses are made using the standards as a reference.

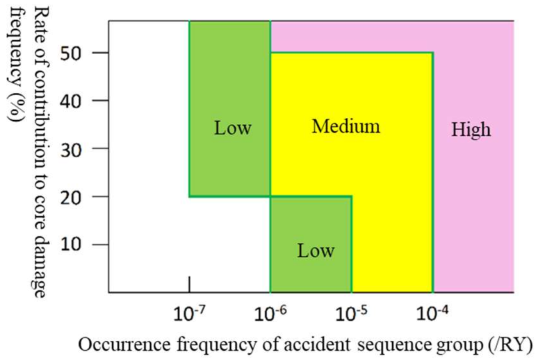

Figure 4. screening criteria

Table 1. CDF of each accident sequence group for each event

Accident sequence group	CDF (/RY)				
	At power	Shutdown (base case)	Shutdown (sensitivity analysis case)	Seismic	Tsunami
Loss of secondary heat removal	3.1E-07 (10.2%)	1.1E-09 (<0.1%)	1.1E-09 (0.1%)	1.1E-07 (11.0%)	8.7E-11 (1.2%)
Total loss of AC power	2.2E-07 (7.2%)	1.0E-07 (0.5%)	7.4E-08 (6.1%)	8.9E-08 (9.0%)	2.5E-09 (33.4%)
Loss of component cooling water	3.7E-07 (12.2%)	8.9E-08 (0.4%)	8.9E-08 (7.3%)	6.2E-07 (62.6%)	4.8E-09 (64.0%)
Loss of containment heat removal	1.3E-08 (0.4%)	ε (<0.1%)	ε (<0.1%)	6.2E-11 (<0.1%)	ε (<0.1%)
Loss of reactor shutdown capability	6.3E-10 (<0.1%)			3.7E-08 (3.8%)	
Loss of ECCS in the injection phase	1.4E-07 (4.5%)	9.5E-11 (<0.1%)	9.5E-11 (<0.1%)	8.1E-08 (8.2%)	ε (<0.1%)
Loss of ECCS in the recirculation phase	1.8E-06 (58.6%)	5.0E-10 (<0.1%)	5.0E-10 (<0.1%)	1.5E-08 (1.5%)	ε (<0.1%)
Containment vessel bypass	2.1E-07 (6.9%)				
Loss of decay heat removal		2.0E-05 (97.3%)	5.6E-07 (45.8%)		
Loss of primary coolant		2.8E-07 (1.4%)	4.1E-07 (33.4%)		
Inadvertent criticality		8.7E-08 (0.4%)	8.7E-08 (7.2%)		
Direct damage to reactor building				ε (<0.1%)	
Direct damage to reactor containment vessel				3.3E-08 (3.3%)	
Steam generator tube ruptures (Structural damage leading to multiple SG failures)				5.6E-09 (0.6%)	
Direct damage to multiple signal systems					1.1E-10 (1.4%)
Total	3.0E-06	2.1E-05	1.2E-06	9.9E-07	7.5E-09

Table 2. CFF of each containment failure mode for each event

Containment failure mode	CFF (/RY) (rate of contribution)		
	At power	Seismic	Tsunami
Steam explosion in reactor vessel	1.8E-10 (<0.1%)	ε (<0.1%)	ε (<0.1%)
Containment isolation failure	8.2E-08 (18.6%)	2.1E-07 (38.3%)	3.3E-10 (13.3%)
Hydrogen combustion (before reactor vessel damage)	ε (<0.1%)	ε (<0.1%)	ε (<0.1%)
Hydrogen combustion (immediately after reactor vessel damage)	ε (<0.1%)	ε (<0.1%)	ε (<0.1%)
Hydrogen combustion (long after reactor vessel damage)	ε (<0.1%)	ε (<0.1%)	ε (<0.1%)
Over-pressure rupture due to accumulation of steam/non-condensing gas	1.3E-07 (30.3%)	3.0E-07 (54.3%)	2.1E-09 (86.3%)
Base mat melt-through	1.1E-09 (0.3%)	1.1E-09 (0.2%)	5.8E-12 (0.2%)
Containment failure before core damage due to steam accumulation	9.8E-09 (2.2%)	9.6E-10 (0.2%)	4.3E-13 (<0.1%)
Steam explosion outside reactor vessel	2.0E-09 (0.5%)	1.2E-10 (<0.1%)	2.7E-12 (0.1%)
Direct heating of containment vessel atmosphere	ε (<0.1%)	ε (<0.1%)	ε (<0.1%)
Interface system LOCA	5.0E-08 (11.3%)		
Steam generator tube rupture	1.6E-07 (36.8%)	6.0E-09 (1.1%)	ε (<0.1%)
Containment vessel failure due to over-temperature	7.3E-11 (<0.1%)	2.1E-10 (<0.1%)	ε (<0.1%)
Direct contact of debris with containment vessel	ε (<0.1%)	ε (<0.1%)	ε (<0.1%)
Containment failure before core damage due to earthquake		3.3E-08 (6.0%)	
Total	4.4E-07	5.5E-07	2.5E-09

3.1.1 Internal events at-power

The core damage accident sequence group classified as significance level "High" was "Loss of ECCS in the recirculation phase". Before or after a low-water-level signal is sent from the refueling water storage tank, if the operator fails in performing a series of operation such as system lineup and pump startup operations required for switchover to recirculation mode, capability to recirculate containment sump water for long term core cooling will be lost. Therefore, multiple operations for recirculation mode switchover must be performed within a short period of time to prevent core damage. In order to reduce the probability of human error in such operations, it is effective to totally or partially automate the required actions for recirculation mode switchover operations (hereinafter referred to as the "ECCS recirculation automation"). Based on this concept, adoption of the ECCS recirculation automation was selected as an additional safety improvement measure. Other additional measures were selected including educating operators about the importance of ECCS recirculation switchover operation in the event of LOCA and continuous implementation of education and training on this operation as efforts to prevent the loss of ECCS recirculation capability. In this way, additional measures were selected from both perspectives of design solutions and enhanced education and training.

Significance classification was conducted on the CFF results based on containment failure modes. Containment failure modes classified into medium significance were "over-pressure rupture due to accumulation of steam/non-condensable gas" and "steam generator tube rupture". Regarding over-pressure rupture due to accumulation of steam/non-condensable gas, core damage sequences resulting to this containment failure mode is mostly from RCP seal LOCA followed by failure of severe accident countermeasures (e.g. core injection using permanent motor pump and mobile large-capacity pump truck). Containment failure occurs if the mobile large-capacity pump truck fails, which leads to loss of failure natural convection cooling within containment using sea water supply to the containment fan cooler units, and followed by failure of specific severe accident management facilities that provide containment filtered vent.

Specific severe accident management facilities are effective in reducing risks associated with containment failure caused by over-pressure. Maintenance and improvement efforts have been already made on the accident response capability by providing education with the aim of training operators to achieve proficiency of operations related to the specific severe accident management facilities as additional safety improvement measures. To achieve further enhancement in reliability, adding confirmation action (recovery step) in the operating procedure that affects filtered vent operation (closure of the airtight door) was selected as an additional safety improvement measure.

In the event of steam generator tube rupture event, the damaged steam generator is isolated and heat removal function is provided using the intact steam generators and the auxiliary feedwater system. Failure in isolating the damaged steam generator results in continuous leakage from the primary system to the secondary system, and when followed by subsequent failure of core damage prevention measures, the core will be damaged and with a containment bypass path. Additional measures selected to deal with steam generator tube rupture events were to provide education and training to operators to achieve proficiency of operations involved in isolation of the damaged steam generator and severe accident measures taken when isolation has failed.

3.1.2 Internal events during shutdown

The Level 1 internal events shutdown PRA was conducted based on the shutdown evolution and maintenance plan developed during the planning phase of the 27th refuelling outage of Sendai 1 (base case) and on the finalized plan after adjustment with the aim of risk reduction. Additional measures were investigated for each of the cases.

As a result of significance classification of core damage accident sequence groups in the base case, "Loss of decay heat removal" was classified as significance level "High". In the 27th refuelling outage of Sendai 1, configuration risk management was conducted in the planning phase using a shutdown risk monitor. The initial shutdown evolution and maintenance plan has been revisited, and by adjusting the evolution and maintenance plan to the extent currently reasonably achievable, shutdown risk was reduced through changes in the seawater system configuration during POS 5 (mid-loop operation). From this experience, continuous effort in performing shutdown risk assessment and management using shutdown risk monitors to develop shutdown evolutions plans with reduced risk to the extent currently reasonably achievable, and to implement risk reduction measures, has been selected as an additional mean for safety improvement.

In the assessment based on the finalized shutdown evolution plan, significance classification was conducted on the core damage accident sequence groups. “Loss of decay heat removal” and “Loss of primary coolant” were classified as significance level “Low”. In POS 4, which is early stage of cold shutdown with the reactor coolant system full, if a failure in one of the component cooling water (CCW) pump occurs when two residual heat removal pumps are in operation, the CCW system load must be restricted to prevent pump run-out of the remaining CCW pump. If load restriction by means of flow rate control on the CCW system is not completed within time, the remaining running CCW pump will also fail, resulting in total loss of CCW. In the event of total loss of CCW, subsequent failure of heat removal function via steam generators will result in core damage. It was assessed that introducing measures to avoid failures to restrict CCW system load that cause total loss of the CCW system, would be effective to prevent this accident scenario. In line with this assessment, additional safety improvement measures were investigated. Further investigation in operations related to CCW system load restriction was selected as additional measure to improve reliability of CCW load restriction.

3.1.3 Seismic events at-power

Significance classification was conducted on the core damage accident sequence groups, and “Loss of component cooling water” was classified as significant level “Low”. If an earthquake causes damage to the seawater intake line structure, which will result in a total failure of the seawater system, and loss of offsite power occurs simultaneously as a result of seismic event, total loss of the CCW and total loss of AC power will occur. Subsequently, if steam generator supply flow rate control by the turbine-driven auxiliary feedwater valve fails, auxiliary feedwater system will be lost and eventually the core will be damaged. In addition, if the earthquake cause structural damage on the non-seismically qualified section of the CCW system, which has a relatively higher damage probability than other SSCs, loss of component cooling capability will occur if isolation of those faulted section fail. Some SSCs that constitute the CCW system have a relatively higher damage probability than other SSCs. If such a components and structure are damaged, total loss of component cooling capability may occur. For this reason, incorporating features in the procedure to enhance CCW system monitoring that enables early detection and isolation of leakage from the CCW system in the event of seismic is effective to reduce risk. Education and training will be continuously provided in line with the measures incorporated in the procedure to increase reliability of early detection and isolation.

Significance classification of containment failure modes were performed based on the Level 2 seismic event at-power PRA, and “Over-pressure rupture due to accumulation of steam/non-condensable gas (δ mode)” was determined significance level “High” and “Containment isolation failure (β mode)” as significance level “Medium”. Regarding over-pressure rupture due to accumulation of steam/non-condensable gas (δ mode), this failure mode can occur when a seismic event causes loss of the offsite power and total loss of the component cooling seawater system triggered by seismic-damage of seawater intake line structure. Loss of component cooling water seawater system results in inoperability of emergency diesel generators that would lead to loss of all AC power. When this accident sequence is followed by failures to start the large-capacity air-cooled generator, the severe accident containment spray become inoperable and containment heat removal capability, including natural convection cooling in containment, will be lost and eventually the containment will be over-pressured. SSCs that constitute offsite power and the CCW system have a relatively higher damage probability than other SSCs. If such SSCs were to be damaged, loss of all AC power and total loss of component cooling water may occur. During such accident conditions, expectations are to establish large-capacity air-cooled generator and alternate power source of the specific severe accident management facilities (power generator). However, component reliability parameter used in the PRA for those additional power sources have large uncertainties due to lack of operational experience, and the PRA results regarding these accident sequences are uncertain. PRA model refinement was selected as a measure because risk analyses are expected to become more realistic through studies to reevaluate the equipment failure rates more realistically by collecting and incorporating operating history records.

Regarding containment isolation failure (β mode), this failure mode can occur when the auxiliary building is damaged. In such an event, various equipment in the building are damaged, causing monitoring instruments become unavailable and containment isolation function degrades. Although this containment failure mode becomes important in the high-acceleration categories, precise analysis of plant response in the event of auxiliary building failure is difficult because complete (catastrophic) damage of buildings is assumed in the PRA. Analyses were conducted for each acceleration category using conditional probability of containment failure shown in Table 3. The results revealed that in acceleration category 1, where there is a high chances of offsite power and class 1 AC power maintained operable is high, if containment isolation failure (β mode) were to occur, there is a high probability that the pressure in the containment at the time of core damage is still below the pressure limit to initiate containment isolation signal and transmission of the containment isolation signal has failed for other reasons. Therefore, to reduce risks from containment isolation failure (β mode), a decision was made to consider adding a procedure to instruct closure of the containment isolation valves at the time of non-transmission of the containment isolation signal.

Table 3. Seismic acceleration category based on seismic bins

Acceleration Category	Mean Frequency of Earthquake Occurrence (/year)	CDF (/RY)	CFF (/RY)	Conditional Probability of Containment Failure
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 4. The additional measures (Level 1 PRA)

Class	Additional Measure	Expected Effect	Event
Equipment/Operation Measures	• Installation of equipment for automatic switchover to ECCS recirculation mode	• Taking measures from both perspectives of equipment solutions and enhanced education and training is expected to be effective for reducing risks because the time allowed for switching to ECCS recirculation mode is short.	• Level 1 internal event, at-power PRA
	• Consideration concerning operation related to load restriction for the component cooling water system	• This measure is expected to reduce the frequency of occurrence possibly leading to a complete failure of the component cooling water system.	• Level 1 internal event shutdown PRA
	• Further safety improvements through continuous risk assessment and management using shutdown risk monitor. (Development of processes with the aim of reducing risks within a currently reasonably achievable extent and implementation of risk reduction measures)	• Use of shutdown risk monitor is expected to be effective in conducting refueling outages with the aim of reducing risks because in each refueling outages, the plant configuration is different.	
Enhanced Education/Training	• Continuous implementation of education and training on ECCS recirculation switchover operation	• Taking measures from both perspectives of equipment solutions and enhanced education and training is expected to be effective for reducing risks because the time allowed for switching to ECCS recirculation is short.	• Level 1 internal event, at-power PRA
	• Education for enhanced monitoring of the retained water quantity of the component cooling water system to prevent failure of the component cooling water system in the event of earthquakes	• This measure is expected to enable early detection of leakage of component cooling water due to earthquakes and reduce the frequency of occurrence leading to a complete failure of the component cooling water system.	• Level 1 seismic, at-power PRA

Table 5. The additional measures (Level 2 PRA)

Class	Additional Measure	Expected Effect	Event
Equipment/Operation Measures	• Regarding successful filtered vent operation (closure of the airtight door), an action of confirmation (recovery step) is added to the procedure	• Owing to the improved reliability of filtered vent operation, this measure is expected to reduce over-pressure rupture risks.	• Level 2 internal event, at-power PRA
	• Consideration concerning adding a procedure of closing the containment isolation valve at the time of non-transmission of the containment isolation signal	• This measure is expected to reduce risks of leading to containment isolation failure (β mode) in the event of non-transmission of the containment isolation signal.	• Level 2 seismic, at-power PRA
Enhanced Education/ Training	• Implementation of education and training on severe accident measures to be taken after isolation of a damaged steam generator or isolation failure	• By providing education and training with emphasis on highly important operations according to scenarios with high risk contribution, it is possible to raise the awareness and accident response capabilities of operators.	• Level 2 internal event, at-power PRA
PRA Model Refinement	• Improving the accuracy of equipment failure rates. (Particularly regarding the equipment that uses substitute parameters [specific severe accident management facilities (e.g. generators)] among failed equipment in important scenarios, continuously collect and incorporate operation history records)	• This measure is expected to reduce the uncertainty included in PRA and enable more realistic analysis of risks.	• Level 2 seismic, at-power PRA

4. CONCLUSION

Additional safety improvement measures, selected through PRA, are as follows: reducing the frequency of occurrence of initiating events, which is an operational solution; refuelling outages and reduction of risks that can lead to containment isolation failure in the event of non-transmission of the containment isolation signal with the aim of reducing risks to a currently reasonably achievable extent; improving systemic reliability by means of enhanced education and training; and improving risk analysis accuracy through PRA model refinement.

To utilize risk insights obtained from PRA in designing and operating nuclear facilities, it is necessary, among other actions, to work on analyses and research for more realistic assessment. Therefore, we will continue to make these efforts.

References

[1] Atomic Energy Society of Japan (AESJ). AESJ-SC-S005:2019. Implementation Standard Concerning Preparation, Maintenance and Improvement of Severe Accident Management in Nuclear Power Plants. Tokyo: AESJ; 2019. in Japanese

[2] Atomic Energy Society of Japan (AESJ). AESJ-SC-P008:2013. A Standard for Procedures of Probabilistic Risk Assessment of Nuclear Power Plants during Power Operation (Level 1 PRA). Tokyo: AESJ; 2013. in Japanese

[3] Atomic Energy Society of Japan (AESJ). AESJ-SC-P009:2021. Standard for Probabilistic Risk Assessment for Nuclear Power Plants At - Power (Level 2 PRA). Tokyo: AESJ; 2021. in Japanese

[4] Atomic Energy Society of Japan (AESJ). AESJ-SC-P001:2019. Standard for Procedures of Probabilistic Risk Assessment of Nuclear Power Plants during Shutdown State (Level 1 PRA). Tokyo: AESJ; 2019. in Japanese

[5] Hidekazu Suwa, Masaharu Inoue et al. Overview of Level 1 PRA (Shutdown State) for Safety Improvement Assessment on Sendai Nuclear Power Station. Nishimu Electronics Industries Co.,LTD (submitted)

[6] Atomic Energy Society of Japan (AESJ). AESJ-SC-P006:2015. A Standard for Procedure of Seismic Probabilistic Risk Assessment for Nuclear Power Plants. Tokyo: AESJ; 2015. in Japanese

[7] Atomic Energy Society of Japan (AESJ). AESJ-SC-RK004:2016. Implementation Standard Concerning the Tsunami Probabilistic Risk Assessment of Nuclear Power Plants. Tokyo: AESJ; 2016. in Japanese