Overview of Level 1 PRA (Shutdown State) for Safety Improvement Assessment on Sendai Nuclear Power Station

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Abstract: Sendai NPS Unit 1 is the first plant in Japan to have completed the "new Regulatory Requirements Conformity Review" after the Fukushima Daiichi NPS accident and returned to commercial operation in September 2015. In accordance with domestic laws and regulations, restarted nuclear power plants are required to conduct Safety Improvement Assessment within six months on completion of periodical inspection, and then report to the Nuclear Regulation Authority without delay. Nishimu, a member of the Kyuden Group, constructed the Level 1 PRA model (Shutdown State) on the latest plant condition (e.g., using Special Facilities for Severe Accidents Management, etc.) in preparation for the 6th Safety Assessment Report for Sendai NPS Unit 1. Nishimu analysed the scenarios and factors leading to core damage based on the PRA results and extracted further safety improvement measures that should be taken in the future. This paper reports an overview of Level 1 PRA (Shutdown State) conducted by Nishimu for the Safety Improvement Assessment on Sendai NPS Unit 1.
Keywords: PRA, Level 1 PRA, Shutdown State, PWR

1. INTRODUCTION

Considering the lessons learned from the accident at Fukushima Daiichi NPS, the “new regulatory requirements” were enforced on July 8, 2013. These include countermeasures against severe accidents, in addition to the enhancement of the conventional requirements at the design basis. Furthermore, nuclear power plant operators that have passed the pre-service inspections and have been restarted are required to carry out an evaluation for safety improvement. This must be done within six months after the completion of the periodic facility inspection. They must also notify the Nuclear Regulation Authority without delay. In accordance with the Safety Improvement Evaluation Notification System, the operators voluntarily and continuously engage in safety improvement efforts, including conducting evaluations such as Probabilistic Risk Assessment (PRA) to reduce risks and enhance safety.
Kyushu Electric Power Co., Inc.’s Sendai NPS Unit 1 is the first plant in Japan to pass the new regulatory requirement conformity review after the Fukushima accident. It has been in operation since 2017, and Kyushu EPCO has submitted six reports of evaluation for safety improvement so far. In 2020, Sendai NPS Unit 1 started operating the Specified Safety Facilities for Severe Accidents Management, and then the 6th Safety Assessment Report reflecting the latest plant condition was notified on November 20, 2023. This paper presents the Level 1 PRA model during shutdown state, as constructed by Nishimu Electronics Industry.

2. Overview of Level 1 PRA during Shutdown

The Level 1 PRA model during shutdown was constructed by referencing the “Standard for Procedures of Probabilistic Risk Assessment of Nuclear Power Plants during Shutdown State (Level 1 PRA):2019” issued by the Atomic Energy Society of Japan. The flow of construction of the Shutdown Level 1 PRA model is shown in Figure 1. The key considerations in the construction of the Level 1 PRA model are noted as below.

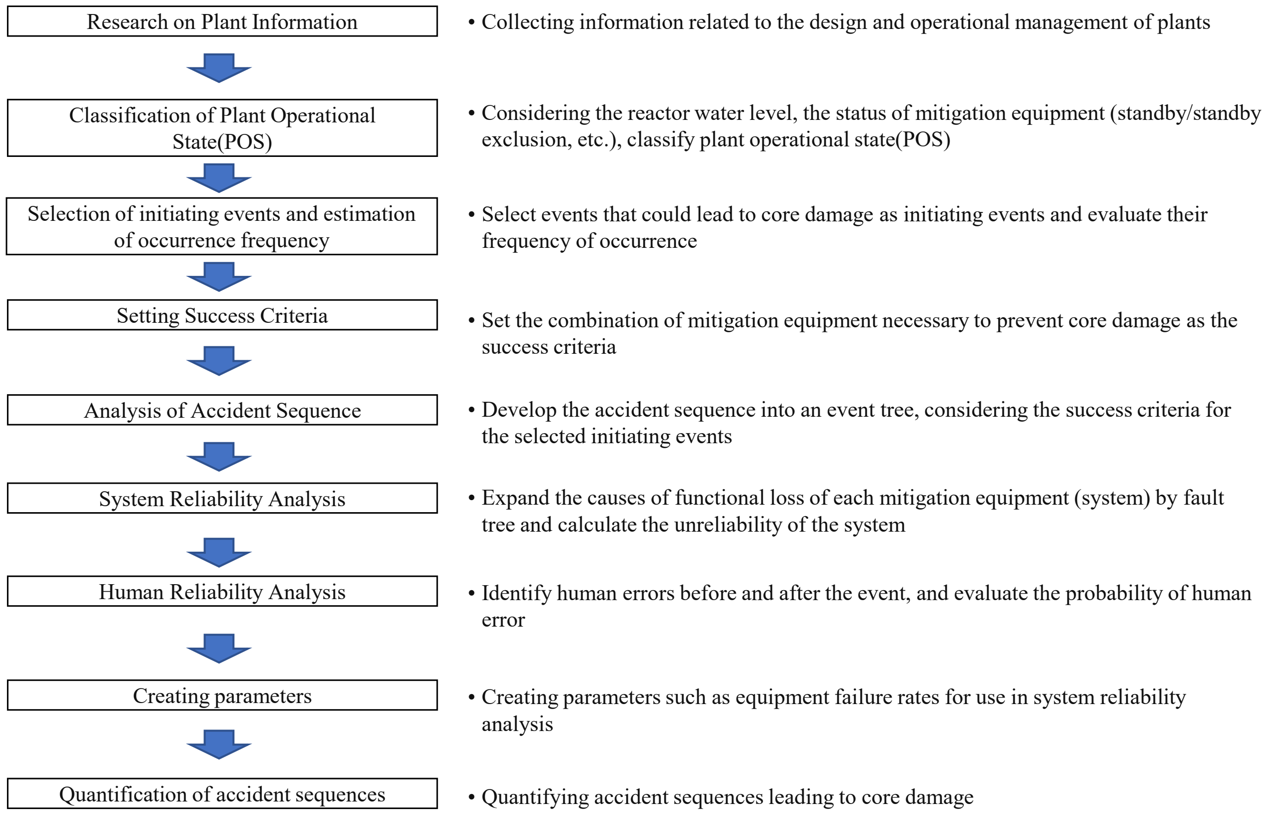


Figure 1. The flow of construction of the Shutdown Level 1 PRA model

2.1. Classification of Plant Operational State (POS)

During the periodic facility inspection, which is the target period for the Level 1 PRA during shutdown, the plant state changes variously due to operator's actions associated with plant shutdown/startup, water level adjustments in the reactor coolant system accompanying maintenance, and exclusion of equipment. As these change, the condition and parameters of equipment related to decay heat removal also fluctuate. Therefore, the Level 1 PRA during shutdown appropriately classifies the plant operational state (POS) by considering the following items.

- Reactor coolant inventory (water level)
- Temperature and pressure of reactor coolant
- Decay heat level
- Condition of mitigation equipment (operation / standby / maintenance (isolation))
- Condition of support system equipment (operation / standby / maintenance (isolation))

The image of the periodic facility inspection process, which is the base of this evaluation, is shown in Figure 2, and the POS classification is shown in Table 1. The characteristics of each POS in the construction of the Level 1 PRA model during shutdown are as follows.

- Referring to the Standard, the period from start of output drop to arrival of rated output is classified into 15 POSs. In this Level 1 PRA during shutdown state, evaluation period is set from "block of automatic start signal of emergency core cooling equipment" (POS 3) to "block release" (POS 13). This is particularly different from Level 1 PRA during power operation in terms of the state of mitigation equipment. Also, in POS 14, only the initiating event "reactivity misinsertion" is modeled.
- Note that the fuel removal state (POS 7) and the full water state of the reactor cavity (POS 6, 8) are not modeled.
- The mitigation equipment for each POS is set based on the planned work schedule of the 27th periodic facility inspection. Since the change of the operating state of the mitigation equipment (such as the isolation of B line of seawater cooling system) was planned during the period of POS 5, it is subdivided into POS 5-1 and POS 5-2.

Table 1. POS classification

POS	Contents of POS	
3	High temperature shutdown state	From the block of ECCS operation signal to the start of cooling state by RHR system
4	State of cooling by RHR system	From the start of cooling state by RHR system to the removal of pressurizer safety valve
5-1	Middle loop operation state	From the removal of the pressurizer safety valve to the isolation of the seawater system
5-2	Middle loop operation state	From the isolation of the seawater system to the completion of cavity water filling
6 (Outside of evaluation)	Upper reactor cavity full of water	From the cavity full of water to the completion of fuel removal
7 (Outside of evaluation)	Fuel removal state	From the completion of fuel removal to the start of fuel loading (state where there is no fuel in the core)
8 (Outside of evaluation)	Upper reactor cavity full of water	From the start of fuel loading to the beginning of cavity water drainage
9	Middle loop operation state	From the start of cavity water drainage to the completion of RCS water filling
10	State of cooling by RHR system	From the completion of RCS water filling to the isolation of the RHR system
11	Primary coolant leakage test	From the isolation of the RHR system to the resumption of cooling state by the RHR system
12	State of cooling by RHR system	From the resumption of cooling state by the RHR system to the isolation of the RHR system
13	High temperature shutdown state	From the isolation of the RHR system to the unblocking of the ECCS operation signal
14	High temperature shutdown state	From the unblocking of the ECCS operation signal to the reactor criticality

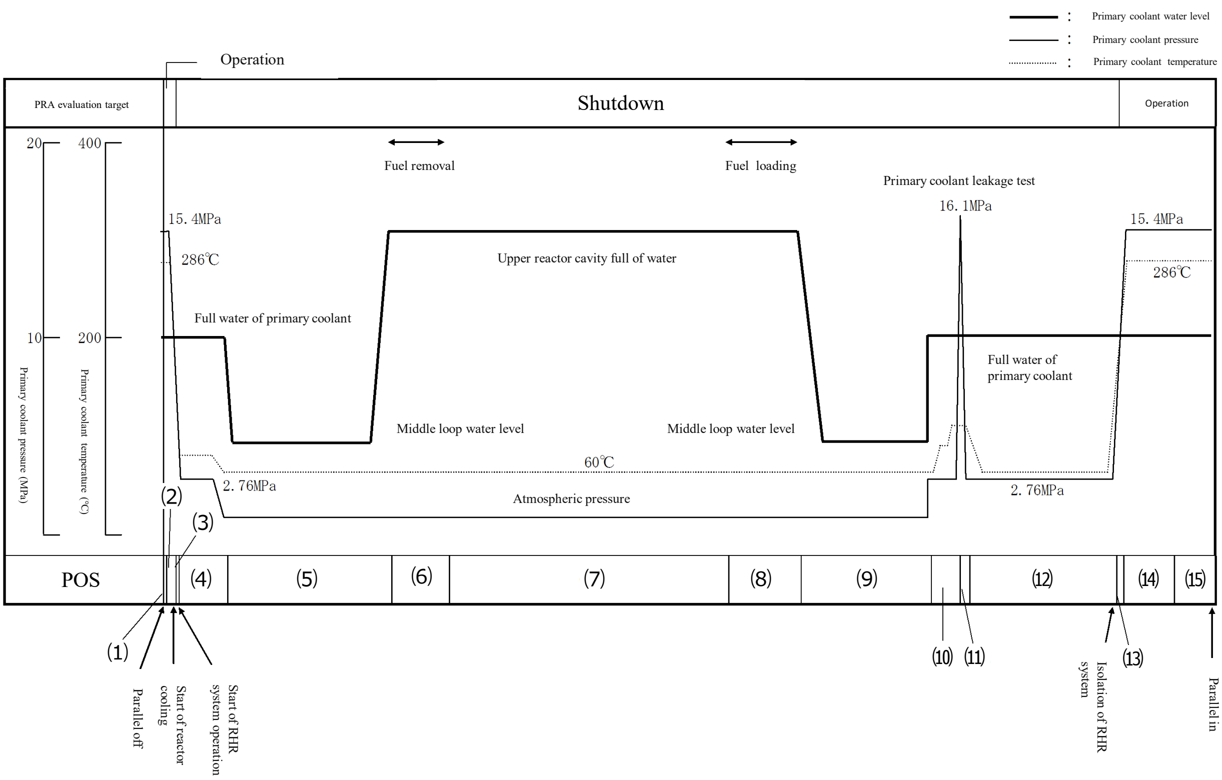


Figure 2. Image of the periodic facility inspection process

2.2. Selection of Initiating Events

The initiating events to be evaluated in the Level 1 PRA during shutdown state are selected based on the review of trouble cases of domestic PRA plants, and are grouped into the following 21 items:

- LOCA of pressurizer relief valve or safety valve
- Loss of Main Feedwater Flow
- Loss of Reactor Coolant Pressure Boundary Function
- Loss of Residual Heat Removal Function
- Overdrain
- Failure to Maintain Water Level
- Loss of Offsite Power
- Partial loss of Control Air System
- Total Loss of Control Air System
- Partial Loss of Component Water System (Loss of A or B header)
- Partial Loss of Component Water System (Loss of C header)
- Total Loss of Component Water System
- Partial Loss of Sea Water System
- Total Loss of Sea Water System
- Partial Loss of Safety-Related High Voltage AC bus
- Total Loss of Safety-Related High Voltage AC bus
- Partial Loss of Safety-Related Low Voltage AC bus
- Total Loss of Safety-Related Low Voltage AC bus
- Partial Loss of Safety-Related DC bus
- Total Loss of Safety-Related DC bus
- Misinsertion of Reactivity

In addition, the specific initiating events to Level 1 PRA during shutdown state are as follows:

- Loss of Residual Heat Removal Function
  An event where all systems in operation of the residual heat removal system lose function due to failure of valves or pumps.

- Overdrain
  An event where the operation to stop water drainage fails during RCS water drainage operation, and the water level continues to decrease.
- Failure to Maintain Water Level
  An event where the RCS water level decreases and continues to decrease due to an imbalance between the filling flow rate and the extraction flow rate caused by a failure of the chemical volume control system during Mid-loop Operation.
- Misinsertion of Reactivity
  An event where abnormal dilution is performed uncontrollably due to human error during reactor startup. (Evaluated as POS 14)

2.3. Analysis of Accident Sequences

For the selected initiating events, the event tree method is used to evaluate accident sequences leading to core damage. The final state of the accident sequence is classified into either a core damage state or a success state, and the accident sequences leading to core damage are classified into the following groups:
(1) Accident sequence group in POS 3, 11, 13
- Loss of Decay Heat Removal Function from Secondary Cooling System
- Loss of all AC Power
- Loss of Reactor Auxiliary Cooling Function
- Loss of Decay Heat Removal Function of Reactor Containment
- Loss of Reactor Shutdown Function
- Loss of ECCS Injection Function
- Loss of ECCS Recirculation Function
- Containment Bypass

(2) Accident sequence group in POS 4, 5, 9, 10, 12, 14
- Loss of Decay Heat Removal Function
- Loss of all AC Power
- Leakage of Reactor Coolant
- Misinsertion of Reactivity

3. Evaluation of Core Damage Frequency

The Level 1 PRA during shutdown state was modelled based on the planned work schedule of the recent 27th periodic facility inspection (February 16, 2023 - April 23, 2023), which served as a base case. In addition, utilizing the shutdown risk monitor, analysed scenarios and factors that could lead to core damage and confirmed the effectiveness of risk reduction, adjustments were made to shift the isolation of the B line of the seawater cooling system, among other things, out of the evaluation target period. The actual process was evaluated as a sensitivity analysis case. The CDF by POS for both the base case and sensitivity analysis case is shown in Figure 3, while the CDF by initiating event and the CDF by accident sequence are shown in Figure 4 and 5, respectively.
The following discussion presents the evaluation results of the base case.
- In the CDF by POS, the CDF of POS 5-2 is dominant.
- In the CDF by initiating event, the Total Loss of Reactor Auxiliary Cooling Water System is dominant.
- In the CDF by accident sequence, the Loss of Decay Heat Removal Function is dominant.

In the base case scenario in POS 5-2 (during the Mid-loop period), which the B line of the seawater cooling system and other equipment are isolated, the failure of the A line of seawater cooling system in operation leads to the failure of all reactors auxiliary cooling systems, resulting in the loss of the decay heat removal function is significant importance.
Also, when comparing the base case and the sensitivity analysis case, the total CDF of the base case is 2.1E-05 (/reactor year), while that of the sensitivity analysis case is 1.2E-06 (/reactor year), which is an order of magnitude lower than the total CDF of the base case. In the sensitivity analysis case, the proportion of POS 4 with high decay heat, POS 5 with Mid-loop water level, and POS 9 have increased. The decrease in the total CDF value in the sensitivity analysis case is thought to be due to the impact of the increase in equipment that can be expected to serve as mitigation equipment, resulting from the delay in the isolation of the B line seawater cooling system in POS 5-2 by adjusting the procedure using the risk monitor.

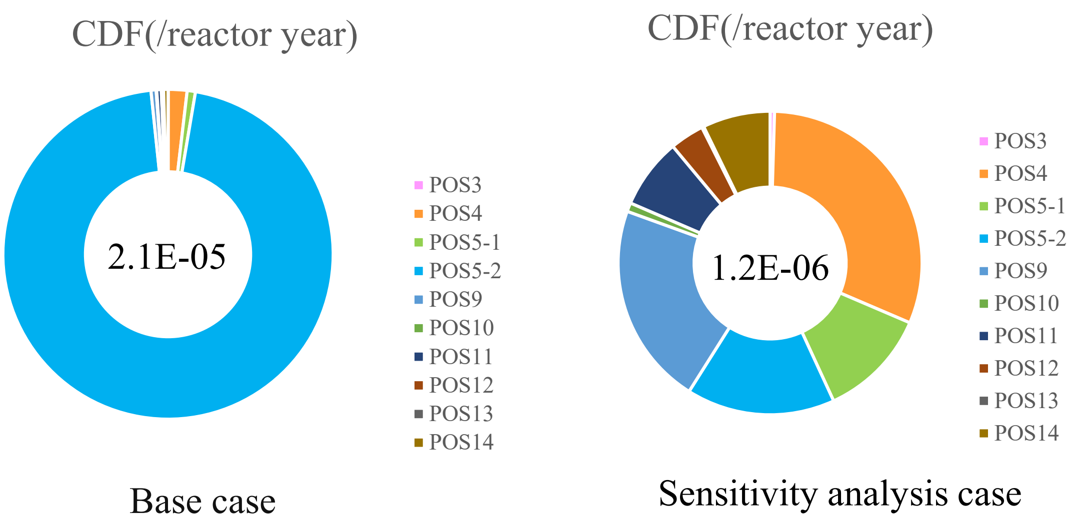
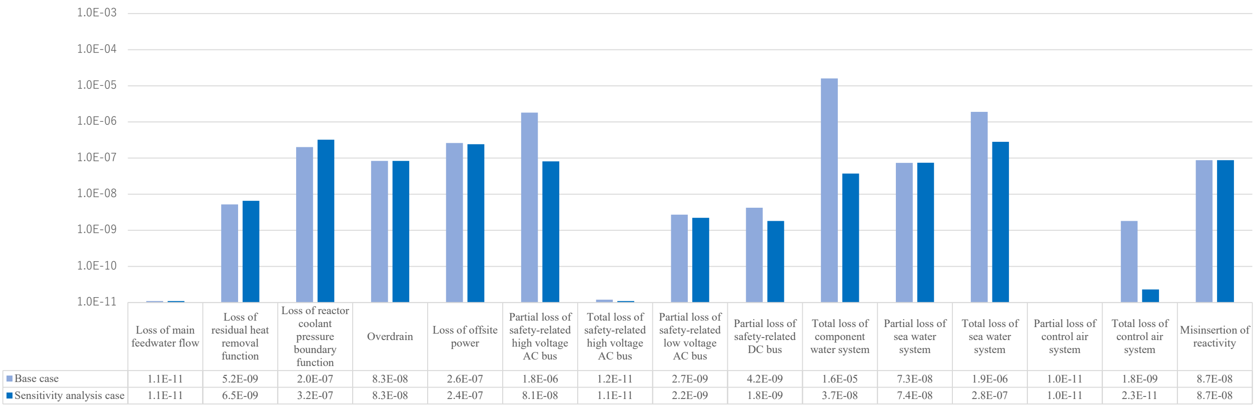


Figure 3. CDF by POS



	Loss of main feedwater flow	Loss of residual heat removal function	Loss of reactor coolant pressure boundary function	Overdrain	Loss of offsite power	Partial loss of safety-related high voltage AC bus	Total loss of safety-related high voltage AC bus	Partial loss of safety-related low voltage AC bus	Partial loss of safety-related DC bus	Total loss of component water system	Partial loss of sea water system	Total loss of sea water system	Partial loss of control air system	Total loss of control air system	Misinsertion of reactivity
Base case	1.1E-11	5.2E-09	2.0E-07	8.3E-08	2.6E-07	1.8E-06	1.2E-11	2.7E-09	4.2E-09	1.6E-05	7.3E-08	1.9E-06	1.0E-11	1.8E-09	8.7E-08
Sensitivity analysis case	1.1E-11	6.5E-09	3.2E-07	8.3E-08	2.4E-07	8.1E-08	1.1E-11	2.2E-09	1.8E-09	3.7E-08	7.4E-08	2.8E-07	1.0E-11	2.3E-11	8.7E-08

Figure 4. CDF by initiating event

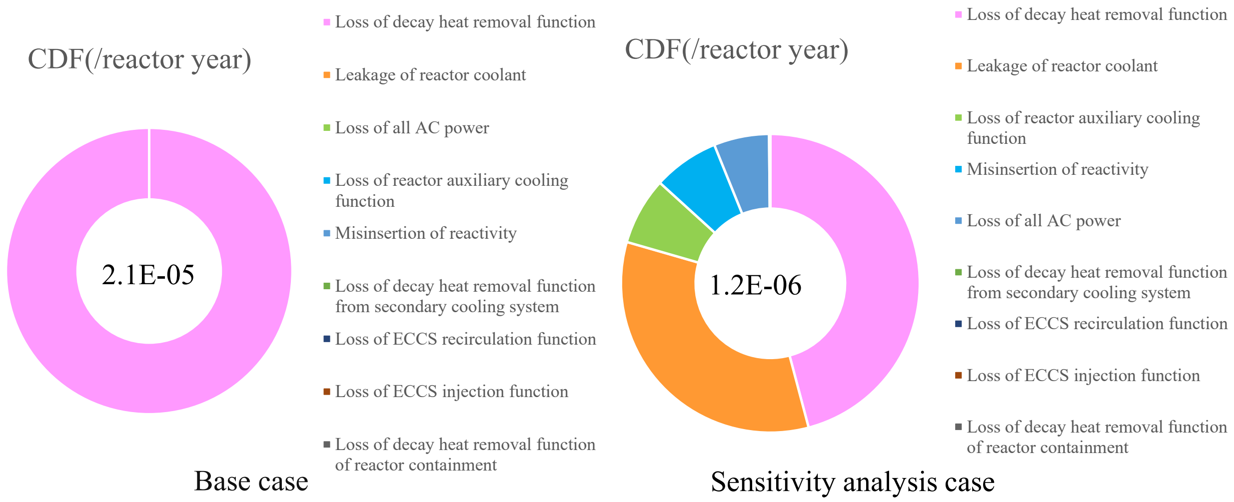


Figure 5.CDF by accident sequence

4. Consideration of Additional Measures

Based on the evaluation results of the base case and sensitivity analysis case, additional measures for improving safety were considered.

4.1. Additional Measures for the Base Case

In the base case, the scenario in which "during the Mid-loop period, the seawater cooling system and other equipment are isolated, and a failure of the operating seawater cooling system leads to a total loss of the reactor auxiliary cooling system, resulting in the loss of the decay heat removal function" is considered significant. In this scenario, by utilizing the shutdown risk monitor, implementing process adjustments to mitigate risk. From the result of the sensitivity analysis, we confirmed that shifting the isolation period of equipment such as the seawater cooling system during the Mid-loop period leads to risk reduction. As the additional measures, we propose to persist in using the shutdown risk monitor, to devise a process, and to implement risk reduction measures that are reasonably achievable.

4.2. Additional Measures for the Sensitivity Analysis Case

In the sensitivity analysis case, the scenario in which "during the cooling of the primary cooling system by the decay heat removal system, one Reactor Auxiliary cooling water pump fails, and if the load limit operation by the flow rate adjustment operation of the Reactor Auxiliary cooling water system is not timely, another Reactor Auxiliary cooling water pump loses its function, leading to a complete failure of the Reactor Auxiliary cooling water system and the loss of the decay heat removal function" is considered significant. For this scenario, the following has been identified as a further measure for CDF reduction.

- Consideration of operation related to load limit of Reactor Auxiliary Cooling Function

5. Conclusion

For the 6th Safety Assessment Report (notified on November 20, 2023) at Sendai NPS Unit 1 of Kyushu EPCO, we constructed a Level 1 PRA model during shutdown state. This model was based on the planned procedure of the most recent periodic facility inspection and the latest plant condition, which includes the Specialized Safety Facilities for Severe Accidents Management. Furthermore, we conducted a sensitivity analysis on process adjustments using the shutdown risk monitor, analysed scenarios and factors that could lead to core damage and confirmed the effectiveness of risk reduction. We have identified further safety improvement measures as part of our ongoing strategy for further CDF reduction. As a member of the Kyuden Group, our company is committed to continuously and voluntarily utilizing PRA in our efforts to further enhance the safety of nuclear power plants.