PRA and Additional measures in 3rd safety improvement evaluation of the IKATA Unit3

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Abstract: In Japan, the plant status is confirmed and evaluated by ourselves with a standpoint of nuclear safety within six months from the end of each outage, and safety improvement activities are planned as additional measures based on the evaluation results. It is mandatory requirement to report to Nuclear Regulation Authority (NRA) and make it public. Regarding Ikata Unit 3, we conducted the third safety improvement evaluation for the plant status at the end of the 16th refuelling outage and submitted it to NRA on December 19, 2023 as the latest evaluation. This paper shows the results of Probabilistic Risk Assessment (PRA) and the details of the consideration of additional safety measures.
A PRA model should be updated every 5 years from the 1st evaluation as a requirement of the safety improvement evaluation guideline. An internal event PRA model during operation is the basis of other PRAs and would be updated regarding plant design changes and operation experiences on a priority basis. The sophisticated PRA model was created by referring advancement technology obtained through the latest information, parameter updates, and reviews by international experts.
The model provided Core Damage Frequency (CDF), Containment Failure Frequency (CFF), the risk evaluation values for each accident sequence group, and each mode of containment loss function. We also calculated each risk evaluation value and contribution ratio to the whole. These data showed important accident sequence groups and containment loss modes. We also identified areas for improvement in design and operation, and extracted additional safety improvement measures. Additionally, it was confirmed that CFF was reduced by more than 50% due to the effect of installing specified serious accident response facilities, which is a mitigation measure to prevent pressurized damage to the containment vessel.

Keywords: PRA, RIDM

1. INTRODUCTION

The Nuclear Reactor Regulation Law (Article 43-3-29) was revised in December 2013, introducing the Safety Improvement Evaluation System for nuclear plants that have resumed operation after undergoing a new regulatory compliance review. This system requires the operator to conduct self-checks and evaluations of the plant's status during periodic inspections, hereinafter referred to as "Periodic Inspections". The results of these evaluations must be reported to the Nuclear Regulatory Authority and opened to the public within six months of the inspection's completion.
Following the completion of the 16th periodic inspection of Ikata Unit 3 on June 20, 2023, a safety improvement evaluation was conducted for the third time. The results of this evaluation were submitted to the Nuclear Regulatory Authority on December 19, 2023, and opened to the public[1]. This paper provides an overview of the Probabilistic Risk Assessment (PRA) in the Third Safety Improvement Assessment Report for Ikata Unit 3, as well as an outline of the additional measures implemented.

2. PRA and Additional measures in 3rd safety improvement evaluation

2.1. Outline

In the third safety improvement evaluation, the PRA was conducted with updated Level 1 and Level 2 PRA models, which serve as the foundation for other PRAs. These updates were made following a five-year interval since the first safety improvement evaluation. The new PRA models incorporated the latest information on plant design and operation, parameter updates, as well as insights obtained from reviews by international experts.

2.2. PRA Model Update Details

The PRA model underwent significant updates, incorporating the Specialized Safety Facility, batteries (third system), and the emergency gas turbine generator, all of which are now operational. Additionally, in order to improve equipment failure rate accuracy, specific plant failure rate data for Ikata was generated using domestic failure rate data as a prior distribution. Furthermore, other relevant findings obtained through the Ikata-3 project were taken into account during the PRA model upgrade. Table 1 provides a summary of the updated PRA contents.

Table 1. Summary of the updated PRA contents

classification		1st safety improvement evaluation	3rd safety improvement evaluation
Plant Information update	**Design information/ Operational information**	equipment for severe accidents	Specialized Safety Facility, batteries (3rd system), and emergency gas turbine generators Plant information updated due to construction etc. after restart.
Parameter update	**equipment failure rate**	Domestic general failure rate data	Individual plant failure rate data
Model advancement	**HRA**	THERP method	HRA Calculator
	Method for estimating the frequency of initiating events	Estimation method using maximum likelihood estimation Plant availability factor not considered	Estimation method using average value Consider plant availability factor
	Modeling alternating operation	×	○
	Other advancements	-	Other information obtained through the Ikata-3 project, etc. Reflecting knowledge related to PRA model advancement

2.3. PRA model analysis results

The advanced PRA model was utilized to assess both the core damage frequency (CDF) and containment failure frequency (CFF). As detailed in Table 2, the adoption of new methods for human reliability, the re-evaluation of success criteria, and the incorporation of newly installed equipment have collectively led to increase in CDF and decrease in CFF.

Table 2. Transition of PRA model sophistication

		Base case	**Sensitivity analysis**	**Case1**	**Case2**	**Case2'**	**Case3**	**Case4**	**Case5**	**Case6**
CDF		1.8E-6	4.2E-6	4.1E-6	3.8E-6	3.8E-6	3.7E-6	3.5E-6	2.8E-6	2.8E-6
CFF		5.7E-7	-	9.3E-7	1.1E-6	5.3E-7	1.1E-6	9.2E-7	6.7E-7	2.8E-7
Analysis conditions	**Survey period for occurrence frequency of causative events**	~2016. 3.31	←	~2018. 3.31	←	←	~2022. 3.31	←	←	←
	Equipment failure rate	29 years of data	←	29 years of data + individual plant data	Domestic general equipment failure rate	←	←	Individual plant failure rates	←	←
	HRA	THERP	HRA Calculator	←	←	←	←	←	←	←
	Frequency of occurrence of initiating event	Maximum likelihood estimate Plant availability factor not considered	←	Average value Consider Plant availability factor	←	←	Plant specific frequency Review of LOOP occurrence frequency	←	←	←
	Modeling alternating operation	×	←	←	←	←	←	←	○	←
	Emergency gas turbine generator	×	←	←	○	←	←	←	←	←
	Specialized Safety Facilities, storage batteries (3rd system)	×	←	←	←	○※	×	←	←	○

※ Partially

2.4. Additional measures extracted from analysis results

Based on 2.3. PRA model analysis results, the Level 1 PRA for internal events identified "loss of ECCS recirculation function" and "loss of reactor component cooling function" as accident sequences with significant risk. To enhance safety, additional measures were identified, including the utilization of risk information in the development of education and training programs, the implementation of automated equipment for ECCS recirculation switching operations, and the introduction of a RCP shutdown seal. Figure 1 illustrates the concept of automated ECCS recirculation switching operations, while Figure 2 provides an illustration of the RCP shutdown seal.

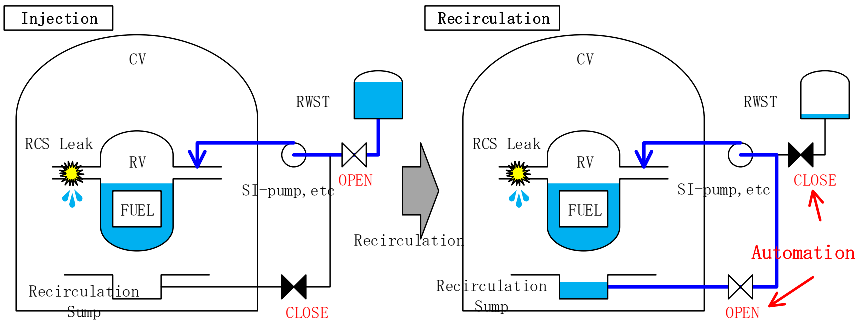

Figure 1. Illustration of automated ECCS recirculation switching operation

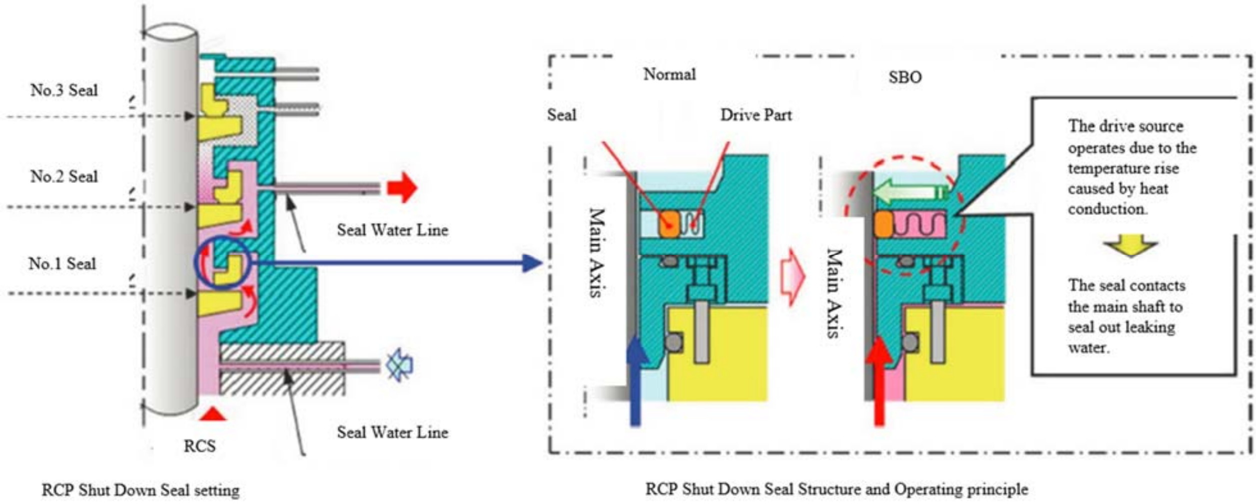

Figure 2. Illustration of RCP shutdown seal

Regarding the Level 2 PRA for internal events, significant containment vessel loss modes were identified as "containment vessel isolation failure" and "steam generator tube rupture". To enhance safety, additional measures were identified, including ensuring diversity in feed-and-bleed operations for the primary coolant system and incorporating risk information into education and training programs. Figure 3 provides an overview of how this information is utilized in the development of education and training programs.

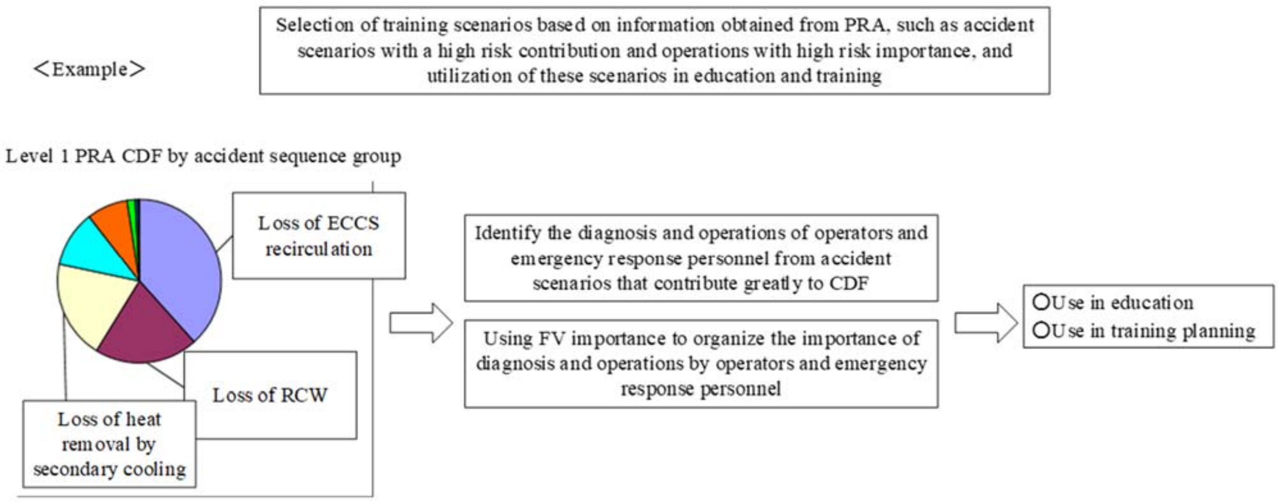

Figure 3. Overview of use in formulating education and training programs

Furthermore, during the evaluation process, it was identified that failure to close the RWST return valve of the residual heat removal pump after its regular operation was a crucial factor affecting the containment failure frequency (CFF). To address this, verification of the closed state of the valve was included as a check item during patrol inspections, and measures were implemented to ensure the valve remained closed. These additional measures are outlined in Table 3. Moving forward, further evaluation will be conducted to consider the introduction of additional measures that may involve equipment modifications.

Table 3. Additional measures extracted

No.	classification	Additional measures	Expected effect	Events
1	RIDM	Utilization in developing education and training programs for operators and emergency response personnel	It can raise the awareness of operators and emergency response personnel and improve their ability to respond to accidents.	L1PRA L2PRA
2	Equipment measures/ Operational measures	Improved reliability of feed-and-bleed operations in primary cooling systems	Regarding feed-and-bleed, which is effective even in scenarios that lead to containment isolation failure, reliability can be expected to improve by establishing procedures for operating multiple operating methods and ensuring operational diversity.	L2PRA
3		Enhanced monitoring of residual heat removal pump RWST return valve closed status during patrol inspections	This prevents forgetting to close the return line valve from the residual heat removal pump to the fuel exchange water tank, which is an important risk factor.	L2PRA
4		Introduction of ECCS recirculation switching automation equipment	It is expected that the reliability of ECCS recirculation switching operation will be improved.	L1PRA
5		Introduction of RCP shutdown seal	This can be expected to reduce the frequency of primary coolant pump seal LOCA when the reactor auxiliary cooling function is lost.	L1PRA

4. CONCLUSION

Through the sophistication of the PRA, we have identified additional measures that can lead to further safety improvements. To ensure the effective utilization of risk information, we remain committed to enhancing the sophistication of the PRA model as part of the Ikata-3 project. This involves incorporating knowledge gained from overseas expert reviews and the research conducted by the Nuclear Regulatory Research Committee (NRRC). Our aim is to continuously enhance the accuracy and reliability of the PRA model for the benefit of overall safety.

References

[1] Shikoku Electric Power Co., Ltd. Homepage,
https://www.yonden.co.jp/energy/atom/safety/safety_improvement/20231219_assessment.html
（Accessed,May 26,2024）