Initiatives to Improve Safety of Ohma-Nuclear Power Plant by Using the Risk Information in the Construction Phase

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Abstract: It is important initiatives for maintaining and improving the safety of Nuclear Power Plant that analyzes the risk information from Probabilistic Risk Assessment (PRA), analyzes vulnerabilities from the risk information and take measures to compensate for the vulnerabilities. The voluntary and continuous improvement of safety is being tried by using the risk information from the current PRA models, and assessing the risks associated with equipment and operations of Nuclear Power Plant. In addition to the initiatives, the PRA models are being advanced to assess the risks of plant more accurately. Internal at-power Level 1 PRA model and Level 1.5 PRA model of Ohma-Nuclear Power plant have already been advanced, and these models are available. These advanced PRA models are based on the information of equipment designs and operation procedures which are assumed now. Although even if these PRA models are not based on finalized plants information, the risk information can be utilized for further designs and operations improvement in each construction phase by considering other information (e.g., Deterministic safety assessment results). From these viewpoints, the PRA models that be reflected the designs and operations at each phase of construction permission and fuel loading will be utilized, and the equipment and operations which should improve the reliability will be selected from PRA. In other words, Design and Operation Reviews will be conducted for further safety improvement of plant. Prior to the start of authentic Design and Operation Reviews, the trial of Design and Operation Reviews was conducted by using the internal at-power level 1 PRA model which has already been advanced. As a result, the workflow of Design and Operation Reviews was clarified, and the methods of selecting Weak Points using the risk information was considered. This paper describes the status of these initiatives at the Ohma-Nuclear Power plant in the construction phase.
Keywords: PRA，RIDM，Design and Operation Reviews

1. INTRODUCTION

At the Ohma-Nuclear Power plant, the equipment and operations to further improve reliability will be considered by using Probabilistic Risk Assessment (PRA) which reflected the design and operation information at each phase of construction permission and fuel loading. These initiatives are called "Design and Operation Reviews."

In the Design and Operation Reviews at the phases of construction permission and fuel loading, the facilities and operations are reviewed based on plants information at that time. At present, the plants information has not been finalized in Ohma-Nuclear Power plant, because construction permission has not been obtained. Although Internal at-power Level 1 PRA model and Level 1.5 PRA model[1] that be based on Design and Operation information at present phase have already been advanced. Therefore, the review based on design and operation information at present phase has become possible.

Even if the PRA models are not based on finalized plant information, the risk information can be utilized for further design and operation improvement depending on the accuracy of design and operation model in each construction phase by considering other information (e.g., Deterministic safety assessment results). In addition, by utilizing the knowledge obtained from the Design and Operation Reviews in the construction phase as training materials, and aim to improve our analysis skills for authentic Design and Operation Reviews.

In this study, the trial of Design and Operation Reviews using the Level 1 PRA model was conducted in order to clarify the workflow of the Design and Operation Reviews, and consider the methods to select the Weak Points using the risk information.

[1] The PRA model was created by referencing the design information until 2018, and operation information which based the procedures and Tech Spec of prior plant.

2. PROCESS OF DESIGN AND OPERATION REVIEWS

The workflow of the Design and Operation Reviews by PRA is shown in Figure 1.

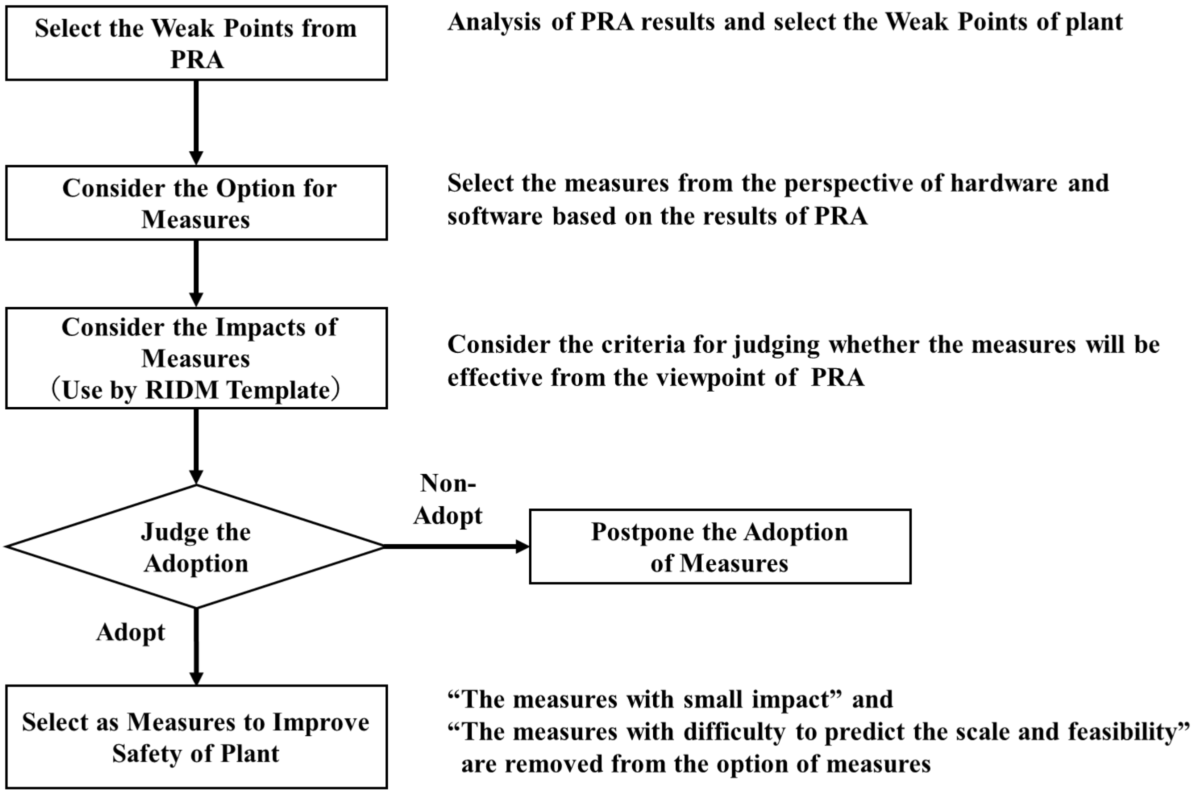

Figure 1. The Workflow of the Design and Operation Reviews by PRA

2.1 Selection of the Weak Points from PRA Results

Some Weak Points were selected from PRA, and the option of measures were considered. The Assumptions for Modeling should be considered, when analyzing and selecting the Weak Points of plant.

2.2 Consideration of the Option for Measures

The Weak Points of plant are selected as Basic Events in PRA. The Basic Events are discussed with design and operation department. The impacts of "The measures with small impacts" and "The measures with difficulty to predict the scale and feasibility" are not considered.

2.3 Consideration of the Impacts of Measures

"RIDM Template" has been introduced in Ohma-Nuclear Power plant for the purpose of systematic visualization on the impacts of measures and to perform Risk-informed decision-making (RIDM). The format of RIDM Template is shown in Figure 2.

Key Elements of the IRIDM Standard [1]	Perspectives to consider	Weight	OPTION 1		OPTION 2
			Impact induced by the option	Score	・・・
Key Element #1: Standards and Good practices			In light with the latest standards and / or good practices in domestic and overseas plants, confirm and evaluate to what extent the current situation will be improved or satisfied in case of implementing each option.		
Key Element #2: Operating Experience			Confirm and evaluate how it will affect the current design and operating status of the target plant in case of implementing each option.		
Key Element #3: Deterministic Considerations					
A. Defence In Depth (DID)					
Ensure balance and independence between DID levels			Confirm that the balance of each level of DID is not overly rely on one or two specific levels of DID and the independence between DID levels is adequately ensured in case of implementing each option.		
Avoid excessive reliance on management measures			Confirm that it does not excessively rely on management measures such as operations by the operators, tests, and inspections to ensure the reliability of each DID level in case of implementing each option.		
Ensure multiplicity or diversity and independence			Confirm that the multiplicity or diversity and independence of the Structures, Systems and Components(SSCs), which are responsible for countermeasures of each DID level, are ensured in case of implementing each option according to the frequency, consequence and their uncertainties of initiating events at the Nuclear Power Plant (NPP).		
Implementation of protective measures against common cause failures	**PERSPECTIVES TO CONSIDER AND THEIR WEIGHTING** Describe perspectives in a concrete and concise manner, associating them with the below individual key elements of the IRIDM standard [1], depending on the profile of the issue and the nature of the proposed options.		Confirm that appropriate protective measures are taken for common cause failures that are commonly related to multiple DID levels in case of implementing each option. And also, confirm that protective measures are taken for common cause failures that have a large impact on sustaining individual DID levels.		
Implementation of measures to prevent human error			Confirm that preventive measures are properly taken for human errors that are commonly related to multiple DID levels in case of implementing each option. And also, confirm that measures are taken to prevent human error, which have a large impact on sustaining individual DID levels.		
B. Safety Margins			Clarify whether there is a possibility that the licensing contents, such as installation permission, construction plan approval, and technical specification approval, will change in case of implementing each option, and confirm that there is no change in the licensing contents.		
			If there is a possibility that the licensing contents may be changed, evaluate all the loads such as temperature and pressure affected by the change in the safety assurance activities, and confirm that the regulatory acceptance criteria in the license are satisfied.		
Key Element #4: Probabilistic Considerations			Perform quantitative risk assessment such as PRA to identify accident scenarios that are affected in case of implementing each option. Identify evaluation assumptions and uncertainty factors related to those scenarios.		
Key Element #5: Organizational Considerations			Confirm and evaluate how it will affect the organization and management in case of implementing each option. And also, confirm and evaluate how it will affect the target certification, maintenance, inspection, and testing.		
Key Element #6: Security Considerations			Confirm and evaluate how it will affect the security measures for nuclear materials in case of implementing each option.		
Key Element #7: Other Considerations			The other considerations for prioritizing options is to estimate and evaluate the economic costs required to implement each option. In addition, confirm and evaluate how the exposure dose of workers will be affected in case of implementing each option. Moreover, identify and evaluate methods to monitor the impact of implementing each option.		
		Total Score			・・・
		Priority Ranking			・・・

Figure 2. The Format of RIDM Template

In this template, Key Elements from "The Implementation Standard Concerning Integrated Risk-Informed Decision Making for the Continuous Safety Improvements in Nuclear Power Plants [1] " are selected as perspectives to considered in order to assess the impacts of measures.

Depending on the characteristics of measures, the perspectives are detailed. Depending on the impacts of taking the measures, set a score between -2 and +2 for each perspective from Key Elements 1-7. The priorities for nuclear safety are clarified by calculating the total score of each measure.

Among these Key Elements, Viewpoint 4 “Probabilistic Considerations" is an element to quantitatively assess the risks from changes to the designs and operations. Therefore, the impacts of each measure are assessed by PRA. Specifically, each measure is modeled in advanced PRA model, and the impacts of risk reduction for total Core Damage Frequency (CDF) is assessed. In the case of risk reduction of more than 10% for total CDF by referring to domestic and foreign practice*, the measures were judged as effective and high priority.

*According to domestic and foreign practice that of "The Implementation Standard Concerning Integrated Risk-Informed Decision Making for the Continuous Safety Improvements in Nuclear Power Plants [1]" "Implementation Standard Concerning Preparation, Maintenance and Improvement of severe Accident Management in Nuclear Power Plants [2]" "RG 1.174, Revision 3 [3]", variations in the total CDF of more than 10% are referred to "variations that require compensating for measures" or "variations that are highly important for measures". Therefore, the criteria for judging the impacts of measures is "Risk Reduction of more than 10% for total CDF".

As a result, the workflow of Design and Operation Reviews were clarified, and the provisional criteria for judging the priority of measures based on PRA were determined.

3. CONSIDERATION OF METHOD FOR SELECTING THE BASIC EVENTS

The Basic Events were selected by method that combining the assessment method by Risk Importance Values based on "JEAC 4209-2021 [4]" and "NEI 00-04, Revision0 [5]" and the viewpoint of accident sequence based on "Implementation Standard Concerning Preparation, Maintenance and Improvement of Severe Accident Management in Nuclear Power Plants [2]" and "NEI 91-04, Revision 1 [6]". Fussell-Vesely (FV)[2] and Risk Achievement Worth (RAW)[3] were used for Risk Importance Values. The method of selecting the Basic Events is shown in Figure 3.

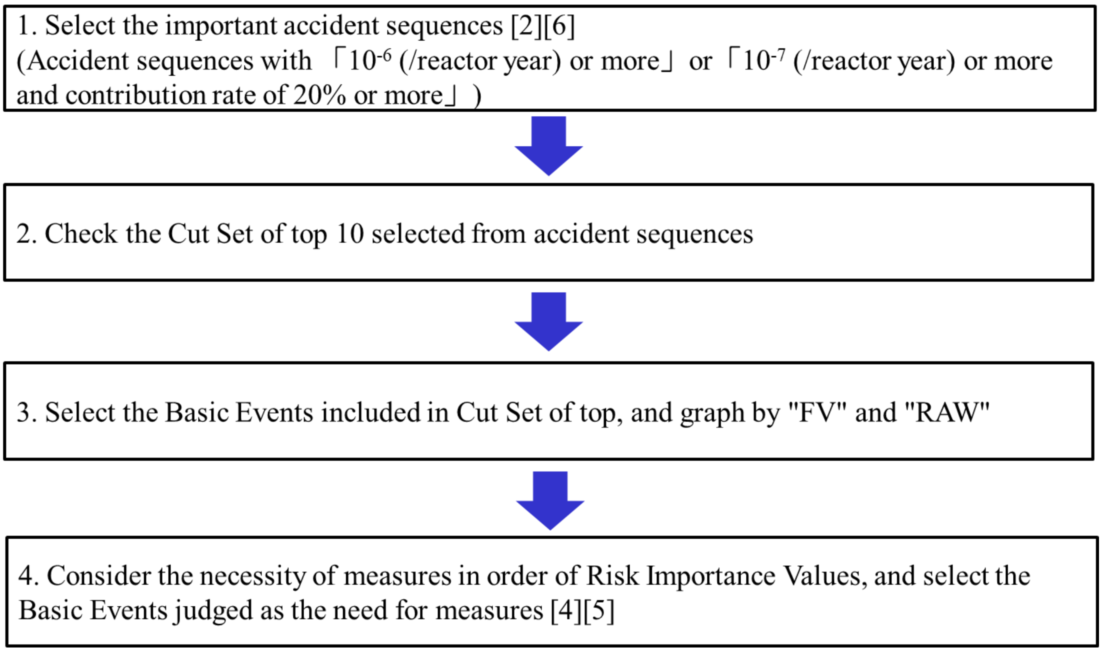

Figure 3. The Method of Selecting the Basic Events

4. SELECTED RESULTS OF THE BASIC EVENTS

Based on the method of selecting the Basic Events shown in Figure 3, the Basic Events that need to be considered for measures were selected. In the consideration of measures, the Risk Importance Value was selected as "FV" because it is possible to analyze the impacts of measures by assessing the CDF reduction. Additionally, "RAW " was also used as reference to judge the Risk Importance Values in case that "FV" is the same level or judge whether to reduce the "RAW" as well.

The Basic Events were selected from Cut Set of top 10 in the critical accident sequences, and graphed by "FV" and "RAW". The results are shown in Figure 4.

[2] The Fussell-Vesely (FV) Importance is the probability, given that a critical failure has occurred, that at least one minimal Cut Set containing a particular element contributed to that failure.
[3] The Risk Achievement Worth (RAW) represents how much the probability of critical failure increases when a particular element fails.

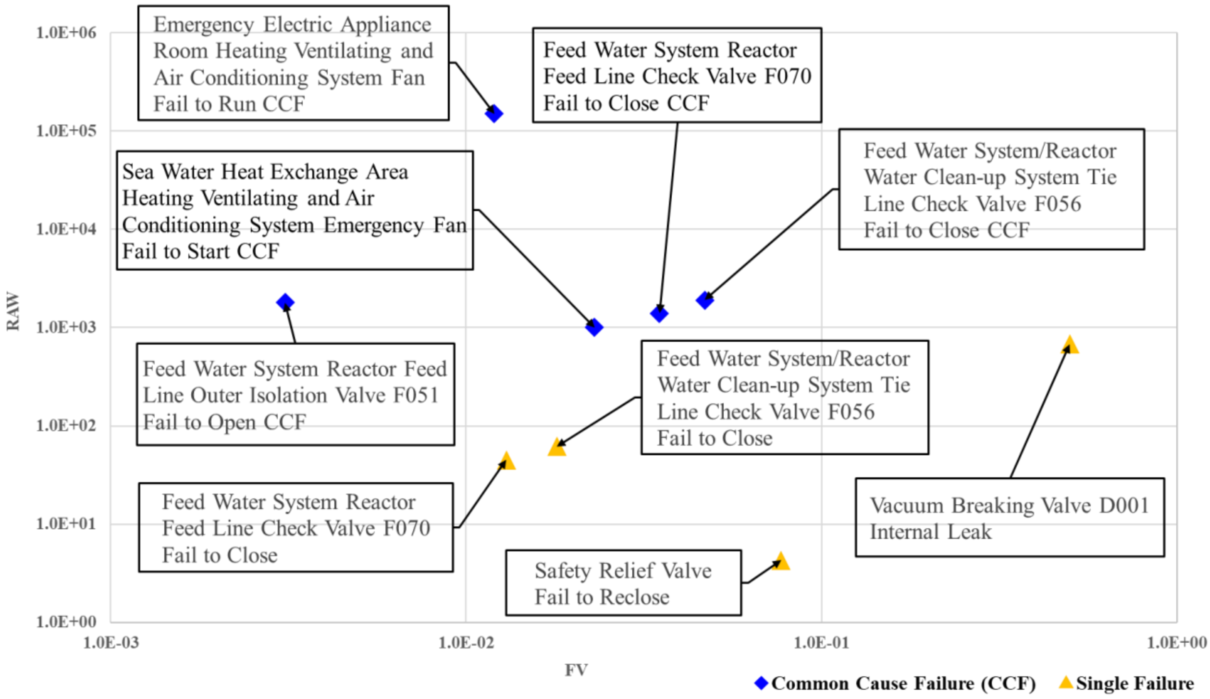

Figure 4. Primary Basic Events Included Top of Critical Accident Sequences

From the results of Figure 4, the Basic Events for considering the measures were selected.

As future initiatives, the measures will be considered for the Basic Events that are selected by discussions with design and operation department, and the impacts of measures will be analyzed by using "RIDM Template".

In this consideration, the Basic Events were assessed by using PRA models that are not based on the finalized designs and operations. Therefore, the PRA models will be updated when the designs and operations are advanced, or PRA assessment method and parameters are updated.

5. OTHER INITIATIVES

5.1 Development of PRA Training Materials

The PRA model was modified and quantified considering several measures (e.g., Multiplying, Diversifying and Improving Reliability) by focusing on a basic event with large "FV". The knowledge obtained from the series of analysis has been reflected as training materials for PRA analysts in 2023.

In preparation for the start of authentic Designs and Operations Reviews when the construction permission is obtained, the training for PRA analysts will be conducted, and the practical analysis will be conducted to confirm the impacts of safety improvements. After the analysis, the results will be reflected in training materials to improve our analysis skills.

5.2 Expansion to Internal at-Power Level 1.5 PRA

The selection of the Basic Events by using at-Power Level 1.5 PRA have been completed, as well as Level 1 PRA.

As future initiatives, the measures for the Basic Events will be considered, and the PRA model will be modified and quantified. The knowledge obtained will be reflected in training materials.

6. CONCLUSION

As the result of this consideration, the following findings were obtained.

・The workflow of Design and Operation Reviews was clarified.
・Provisional criteria for judging the priority of measures based on PRA was clarified.
・The measures were selected according to the purpose of Risk Importance Values, by combining the viewpoints of "FV and RAW" and "CDF of each accident sequence".
・The Basic Events were screened from the quantitative viewpoints based on domestic and foreign practice.

REFERENCES

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