Implementation of risk-informed component isolation prior to periodic inspection at nuclear power plants

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Abstract:

Based on the lessons learned from the severe accident caused by the 2011 off the Pacific coast of Tohoku Earthquake, Japanese nuclear utilities have implemented various countermeasures against internal and external events. For the plants that have resumed operation, PRA models have been developed and the effectiveness of the countermeasures has been quantitatively evaluated in safety improvement assessments. In addition, the Risk-Informed Decision-Making (RIDM) approach has been introduced to the plant management [*1]. RIDM enables risk management and the selection of appropriate countermeasures based on risk information, even when inspections of safety-important equipment are required while the plant is operating. As an example, a case in which risk information was utilized in the isolation of the standby seawater pump prior to the periodic inspection at Kansai Electric Power's Ohi Nuclear Power Station Unit 3 in 2020 is illustrated here.

Keywords: PRA, RIDM

1. Utilization of PRA for nuclear power plants

Based on the provisions of the new regulatory requirements established in light of the severe accident caused by the 2011 off the Pacific coast of Tohoku Earthquake, various facilities have been installed and enhanced education/training has been conducted at Japan's nuclear power plants to deal with internal and external events. For the plants that have resumed operation after meeting the new regulatory requirements, probabilistic risk assessments (PRAs) are being conducted as part of the safety improvement assessments, and effective measures for further safety improvements are being extracted by analyzing the PRA results. Based on these evaluation results, the Japanese nuclear utilities is working to install additional facilities and improve training and education programs.

At Kansai Electric Power's nuclear power plants, we perform PRAs before carrying out work on safety-important equipment, and use the PRA results to determine whether or not the work can be carried out and to select compensatory measures. In this way, we are pursuing the optimized operation of our plants by introducing the risk-informed decision making approach to the management of our plants [1].

One of the problems in the current operation of nuclear power plants is that the number of equipment subject to inspection has increased at plants that have resumed operation after meeting the new regulatory requirements, and there is concern about the deterioration of construction quality due to the congestion of on-site work during the periodic inspection. One solution to this problem is to level the workload by introducing on-line maintenance. However, under the current regulatory framework, workload leveling can only be partially implemented. The reason for this is that the current regulatory system does not permit the planned shifting of the alignment of safety-important equipment, which is subject to the limiting conditions for operation specified in the Tech. Specs., to the outside of the limiting conditions for operation, except for the purpose of preventive maintenance. The current regulatory regime poses a challenge for the Japanese nuclear utilities in actively enhancing the plant safety by utilizing risk information as in the U.S. in the future. To address this challenge, it is necessary for the Japanese nuclear utilities to continuously demonstrate that they can perform on-line maintenance on equipment other than that subject to the Tech. Specs. utilizing risk information and implement maintenance management of their plants while ensuring safety. As an example, the construction work conducted at Kansai Electric Power's Ohi unit 3 in 2020, in which the standby seawater pumps were isolated prior to the periodic inspection, is presented in the next section and thereafter.

2. Example of utilization of risk information at a nuclear power plant

2.1 Operation of seawater system and seawater pumps and outline of construction work at Ohi NPP

Ohi units 3 and 4 have three seawater pumps (hereafter referred to as SWPs) ; A-SWP, B-SWP, and C-SWP. During plant operation, one SWP is in operation and two other SWPs are on standby. In the normal operating configuration, A-SWP or B-SWP supplies seawater to A-train in the seawater system (hereafter SWS), while B-SWP or C-SWP supplies seawater to B-train in the SWS. The train in the SWS that is not in use during normal operation is closed by a stop valve. In the event that the train in the SWS that is used during normal operation loses its function due to an accident, etc., it is necessary to manually open the stop valve to supply seawater to the train that is not used.

A construction project was planned for 2020 to integrate the rotary screen and raked bar screen installed in the seawater pump room of Ohi unit 3 (to be built in a single unit). Figure 1 shows an overview of this modification work. During conducting the modification work, it is necessary to isolate the seawater pump on the downstream side. However, if this work were carried out during the periodic inspection, there would be a concern that the quality of the work would deteriorate due to work congestion. Therefore, the work was planned to be carried out prior to the periodic inspection with the B-SWP isolated. Figure 2 shows an overview of the configuration of the seawater system during the modification work.

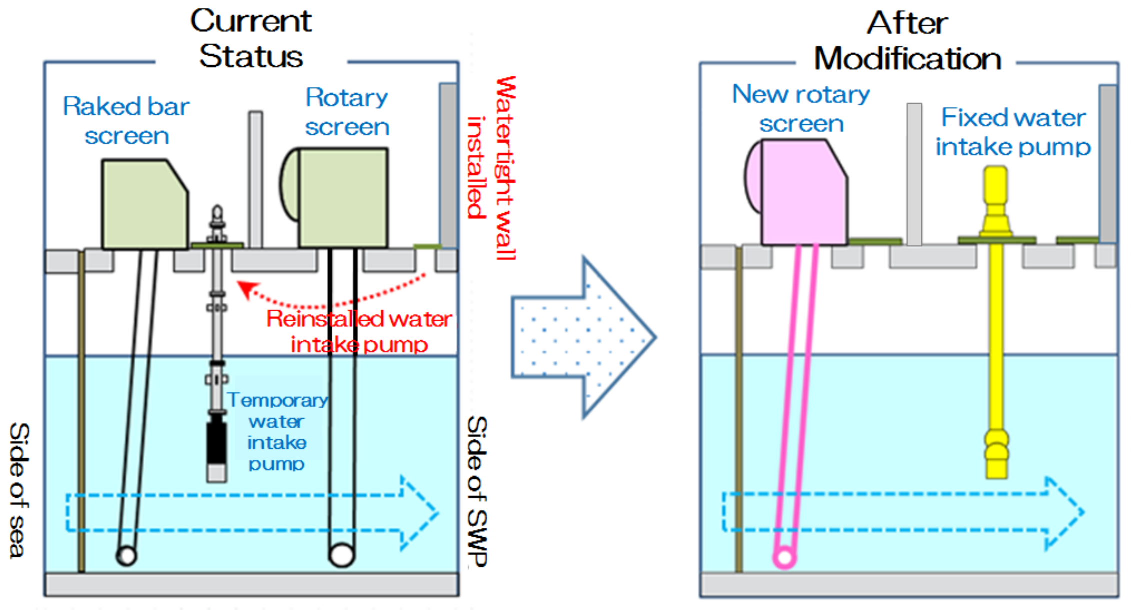

Figure 1 Overview of modification work

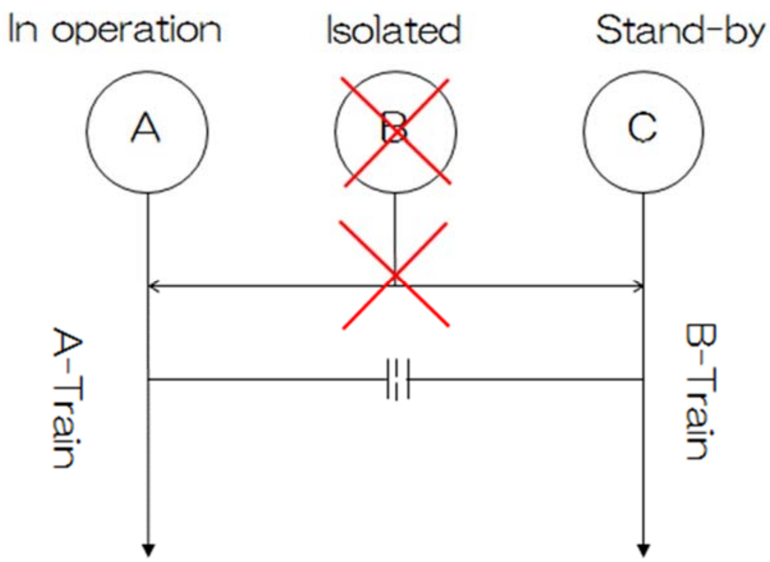

Figure 2 Configuration of SWS during the modification work

2.2 Risk assessment results for the configuration with a SWP isolated prior to periodic inspection

The SWPs are the important support system equipment for the safety-related component cooling water system. After determining that the isolation of one SWP would have a significant impact on the plant risk because there would be only one SWP on standby, the risk assessment on the period of construction work was conducted.

Table 1 shows the results of the PRA conducted on both the Ohi unit 3 B-SWP isolated and non-isolated (base case) scenarios. The incremental CDF was calculated assuming a 50-day construction period. As shown in Table 1, the impact due to an increase in the frequency of occurrence of the initiating event, “Total loss of component cooling seawater system” was dominant.

Table 1 PRA results

		① Base case/reactor year	② SWP isolated /reactor year	③ −① Incremental CDF/50 days
CDF for each initiating event	Total CDF	2.09E-06	3.72E-06	2.22E-07
	Total loss of SWS system	3.49E-07	1.92E-06	2.15E-07
Initiating event frequency	Total loss of SWS system	1.18E-05	6.22E-05	5.05E-03

2.3 Examination of compensatory measures

2.3.1 Extraction of candidate compensatory measures

Since many of the basic events with a higher Fussell-Vesely (hereafter FV) importance in B-SWP isolation are related to the frequency of “total loss of component coolant seawater system”, compensatory measures to reduce the frequency of this initiating event that were considered effective were examined. As shown in Table 2, the basic events considered for compensatory measures were those related to human errors (ranked 2nd, 8th, and 10th in FV) associated with seawater pump switching operations among the “equipment failure” and “human error” events with higher FV. These operations are related to the human error in which an operator fails to activate the C-SWP and open the B-train stop valve to switch the seawater system and component cooling seawater system to the B-train in the event of a failure of A-train of the seawater system or A-train of the component cooling seawater system. The dominant probability of this human error is the cognitive error probability, and we investigated methods to reduce this cognitive error probability as a compensatory measure.

Table 2 Human error related to SWP switchover operation

FV importance	Basic event	Cognitive error probability	Operator action error probability	Human error probability
2nd	Starting C-SWP/Opening B-CCW stop valve/Failure to start C, D-CCWPs (in case of failure of A, B-SWPs)	6.1E-03	1.7E-03	7.9E-03
8th	Starting C-SWP/ Failure to open B-CCW stop valve (in case of failure of A, B-CCWPs and success in automatic startup of C, D-CCWPs)	6.1E-03	1.2E-03	7.3E-03
10th	Starting C-SWP/Opening B-CCW stop valve/Failure to start C, D-CCWPs (in case of clogging of CCW A-train)	6.1E-03	1.7E-03	7.9E-03

2.3.2 Selection of compensatory measures

Cognitive error probabilities are calculated using the Cause-Based Decision Tree Method (hereafter CBDTM). According to the CBDTM, error in the operator-information interface and error in the operator-procedure interface are each divided into some components, as shown in Table 3, and then the probability of cognitive error for each component $p_c a$ to $p_c h$ is calculated according to the decision tree[2].

The CBDTM for evaluating the probability of human error (ranked 2nd 8th, and 10th in FV) in the seawater pump switching operation, which is a candidate compensatory measure, is characterized by the evaluation result of P_cf. The decision tree of P_cf is shown in Figure 3. Since the answer to the interview item, "Are the instructions in the procedure manual clear and contain all the necessary information?" was "No", the blue line on the decision tree is followed, resulting in the cognitive error probability of (f) 6.0E-3. The factor that made the answer to the interview item "No" was the lack of operating instructions that clearly specified the time to initiate the seawater system switching operation and the switching procedure. Accordingly, as a compensatory measure, the preparation of an operating procedure for the implementation of the operation to switch to the standby B-train (i.e., starting the B-train SWP, opening the stop valve of the B-train component cooling seawater system, and starting the B-train component cooling water pump) in the event of loss of functions of the seawater system and the component cooling water system of the A-train, which is in operation, (e.g., running failure of the pump) during the period in which the B-seawater pump is isolated while the reactor is in at-power operation, was selected. As a result, the answer to the interview item, "Are the instructions in the procedure manual clear and contain all the necessary information?", which is the description of the designator P_cf, turns to "Yes" and the probability of cognitive error results in (a)neg.

Table 3 CBDTM failure mechanisms[2]

High-Level Failure Mode	Designator	Description
Failures in the operator-information interface	p_c a	Data not available
	p_c b	Data not attended to
	p_c c	Data misread or miscommunicated
	p_c d	Information misleading
Failures in the operator-procedure interface	p_c e	Relevant step in procedure missed
	p_c f	Misinterpret instruction
	p_c g	Error in interpreting logic
	p_c h	Deliberate violation

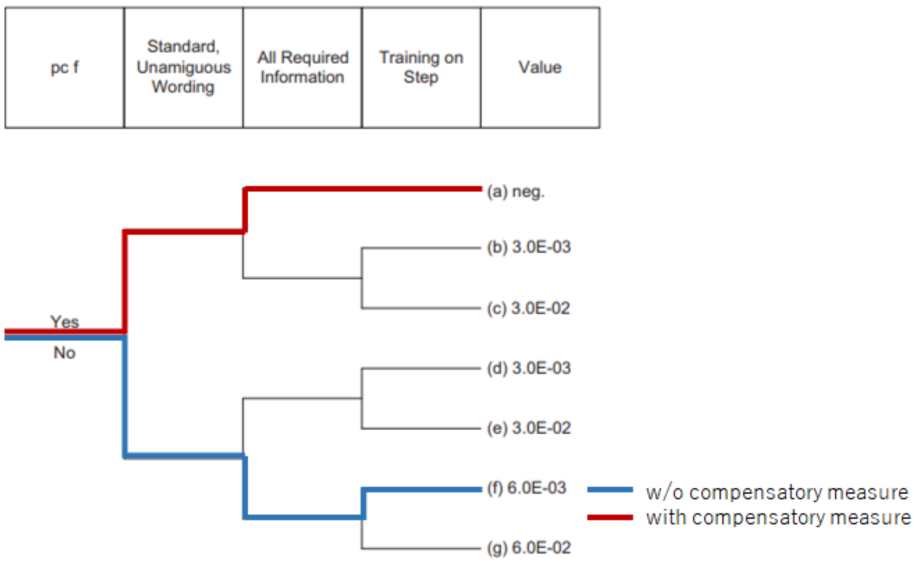

Figure 3 Decision tree for p_cf: misinterpret instruction[2]

2.3.3 Confirmation of the effectiveness of the compensatory measure

The effectiveness of the compensatory measure discussed in section 2.3.2 was confirmed. A sensitivity analysis was performed by applying the compensatory measure, changing the cognitive error probabilities of three operations, and conservatively setting the human error probabilities as shown in Table 4. As a result of the sensitivity analysis, the incremental CDF assuming 50 days of seawater pump isolation was 3.6E-08, which was about one-sixth of that in the case without the compensatory measure. After confirming that the incremental total CDF was sufficiently low with the implementation of the compensatory measure and that there were no problems with the contents of the compensatory measure, the modification work was implemented.

Table 4 Human error probabilities after application of the compensatory measure

FV importance	Basic event	Cognitive error probability	Operator action error probability	Human error probability*
2nd	Starting C-SWP/Opening B-CCW stop valve/Failure to start C, D-CCWPs (in case of failure of A, B-SWPs)	1.4E-04	1.7E-03	2.0E-03
8th	Starting C-SWP/ Failure to open B-CCW stop valve (in case of failure of A, B-CCWPs and success in automatic startup of C, D-CCWPs)	1.4E-04	1.2E-03	1.5E-03
10th	Starting C-SWP/Opening B-CCW stop valve/Failure to start C, D-CCWPs (in case of clogging of CCW A-train)	1.4E-04	1.7E-03	2.0E-03

*Conservative values are adopted.

Table 5 Sensitivity analysis results after implementing the compensatory measure

Period of SWP isolation	ΔCDF (after implementing compensatory measure)	ΔCDF (without compensatory measure)
50 days	3.55E-08	2.22E-07

3. Conclusion

As an example of the use of risk information at nuclear power plants in Japan, a case of modification work at Ohi Nuclear Power Station was introduced here; when work involving isolation of the seawater pump, which is safety-important equipment, became necessary while the plant was in at-power operation, the work was implemented by extracting risk-important operations utilizing PRA and selecting an appropriate compensatory measure. In the future, it is planned to introduce on-line maintenance in a more extensive manner by proactively utilizing risk information, including that obtained from PRAs in order to prevent deterioration in the quality of on-site work by leveling the workload while ensuring safety, and to demonstrate the ability of Japanese nuclear utilities to inspect equipment subject to the limiting conditions for operation specified in the Tech. Specs. while ensuring safety, as is the case in the United States.

4. References

[1] “Nuclear Operators’ Initiatives for Strategic and Action Plans for Implementation of Risk Information Utilization at Nuclear Power Plants” Federation of Electric Power Companies of Japan, December 2023
[2] EPRI/NRC-RES Fire Human Reliability Analysis Guidelines, Final Report, EPRI ,NRC