Efforts to Enhance the PRA Model at Tokai No.2 Power Station

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Abstract:
The Japan Atomic Power Company (JAPC) is working to enhance the Probabilistic Risk Assessment (PRA) model at the Tokai No.2 Power Station. This initiative aims to continuously improve the safety of nuclear power plants by utilizing plant-specific risk information. This paper outlines the main modification from the PRA model developed during the conformity assessment of the new regulatory standards for Tokai No.2 Power Station, focusing on the frequency of initiating events, system reliability analysis, parameter development, and human reliability analysis for the Level 1 PRA model for internal events during power operation.

Keywords: Probabilistic Risk Assessment, Level 1, Internal Events.

1. INTRODUCTION

In response to the accident at Tokyo Electric Power Company's Fukushima Daiichi Nuclear Power Plant, the Japanese Nuclear Regulation Authority published the "Operational Guide for Safety Improvement Evaluation of Commercial Nuclear Reactors"[1]. Consequently, Japanese nuclear operators are required to periodically evaluate the safety of nuclear power plants. Against this background, The Japan Atomic Power Company (JAPC) is enhancing the PRA model at the Tokai No.2 Power Station to continuously improve plant safety using risk information.

This paper explains the main changes to the Level 1 PRA model for internal events during power operation.

The enhanced Level 1 PRA model is based on the model developed during the conformity assessment for the new regulatory standards at Tokai No.2 Power Station[2], which is in conformity to the Atomic Energy Society of Japan (AESJ) Level 1 PRA Standard (2008)[3]. This model has been improved by incorporating revised PRA evaluation methods based on the AESJ Level 1 PRA Standard (2013)[4], as well as evaluation cases from domestic and international nuclear plants, and reflecting the latest design information and operational data.

2. PLANT CONFIGURATION OF TOKAI NO.2 POWER STATION

Figure 1 shows a schematic diagram of the Tokai No.2 Power Station (3293MWt). It is a BWR-5 plant with a Mark-II type containment manufactured by GE. Following the Fukushima Daiichi accident, severe accident countermeasures have been expanded.

Specifically, in addition to permanent alternative water injection equipment and portable water injection equipment, a pump for circulating cooling inside the containment vessel and a backup seawater pump have been added. Furthermore, a filtered vent system has been added to strengthen the heat removal function of the containment vessel. Additionally, permanent backup emergency power supply system and portable equipment have been added to strengthen power supply functions.

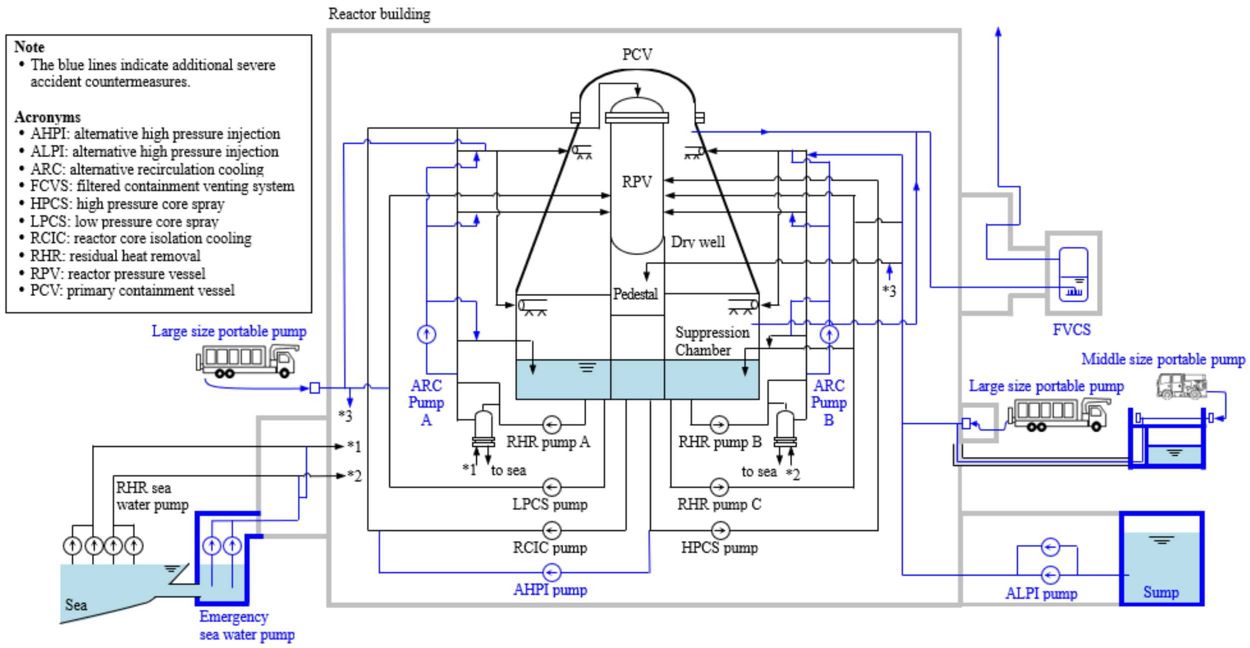

Figure 1. Schematic Diagram of the Plant System

3. MAIN MODIFICATIONS ACCOMPANYING PRA MODEL ENHANCEMENT

3.1 Selection and Evaluation of Initiating Event Frequencies

3.1.1 Selection of Initiating Events In the Previous Model

In the previous model, initiating events were selected based on literature surveys of domestic and international PRA documents, such as NUREG-1150[5] and EPRI-2230[6], and reviews of domestic nuclear plant trouble cases. In the enhanced model, initiating events are selected through a re-survey of domestic and international literature, as well as using Failure Mode and Effects Analysis (FMEA), a systematic method for analysing initiating events as exemplified in the AESJ Level 1 PRA Standard (2013)[4]. Table 1 compares the selected initiating events before and after enhancement.

(1) General Transient
Declining reactor water level and spurious trip via instrumentation are integrated into other initiating event groups considering the similarity in event progression and expected mitigation systems.

(2) Support System Initiating Events (SSIE)
For support system initiating events (SSIE) that affect the unavailability of mitigation equipment, new events such as loss of electrical room cooling system and loss of instrument air system have been selected through FMEA.

(3) LOCA inside PCV
The previous model considered only three cases of pipe break size (large, medium, and small). In the enhanced model, the break locations are categorized according to their impact on mitigation facilities and similarity in the success criteria, and initiating events are divided into more than 20 categories. It also includes additional events such as Excessive LOCA, Very Small LOCA, and Multiple Safety Relief Valves (SRV) Spurious opening due to failure of the automatic depressurization system.

(4) LOCA outside PCV
In addition to IS-LOCA, which was considered in the previous model, High-Energy Line Breaks (HELB) has been newly selected for the enhanced model.

Table 1. Comparison of selected initiating events : Previous Model vs. Enhanced Model

Category	Previous Model	Enhanced Model	Changes
General Transient	Non-isolation Event*	Non-isolation Event*	
	Isolation Event**	Isolation Event**	
	Loss of Feedwater	Loss of Feedwater	
	Declining Reactor Water Level	Loss of Feedwater	Integration
	Spurious Trip via Instrumentation, RPS Fault	Non-isolation Event*	Integration
	Loss of Offsite Power	Loss of Offsite Power	
	Inadvertent Opening of SRV	Inadvertent Opening of SRV	
Support System Initiating Events	Loss of RHR Seawater System Function	Loss of RHR Seawater System Function	
	Loss of AC Power	Loss of AC Power	
	Loss of DC Power	Loss of DC Power	
	Turbine Support System Failure	Loss of Auxiliary Cooling Water System	
	N/A	Loss of Instrument Air System	Addition
	N/A	Loss of Electrical Room Cooling System	Addition
LOCA inside PCV	N/A	Excessive LOCA	Addition
	N/A	Multiple SRV Spurious Opening	Addition
	Large LOCA (1case)	Large LOCA (7case)	Specification
	Medium LOCA (1case)	Medium LOCA (7case)	Specification
	Small LOCA (1case)	Small LOCA (8case)	Specification
	N/A	Very Small LOCA	Addition
LOCA outside PCV	IS-LOCA	IS-LOCA	
	N/A	HELB	Addition
Controlled Shutdown	Unplanned Shutdown	Unplanned Shutdown	

* This is an event where the reactor automatically scrams due to a turbine trip or similar occurrence, and the reactor and turbine sides are not isolated from each other, because the feedwater system remains available for use even after the event occurs.

** This is an event where the reactor automatically scrams due to the closure of the main steam isolation valve or similar occurrence, and the reactor and turbine sides are isolated from each other. To use the feedwater system, it is necessary to open the main steam isolation valve or similar operations.

3.1.2 Evaluation of Initiating Event Frequencies

In the enhanced model, updated actual data is used along with evaluation methods that can consider the detailed design of individual plants. Table 2 shows a comparison of the initiating event frequencies before and after enhancement.

(1) General Transient and Unplanned Shutdown

In the previous model, initiating event frequencies were calculated based on actual data from the operation start of all 32 domestic BWR plants until 2008. The enhanced model calculates initiating event frequencies for individual plants by Bayesian updating using actual data from Tokai No.2 Power Station, based on the actual data of domestic BWR plants from the operation start until 2014.

(2) SSIE

In the previous model, initiating event frequencies were calculated through statistical evaluation from the total operation period due to the lack of actual occurrence experience. In the enhanced model, initiating event frequencies are calculated considering the characteristics of individual plants, referencing EPRI TU-1016741 [7], and modeled as single basic events using point-estimate fault tree methods to avoid complicating the PRA model.

(3) LOCA inside PCV
In the previous model, the initiating event frequency was estimated for one case each for large, medium, and small LOCA from the literature data of NUREG/CR-5750[8] and NUREG-1829[9]. In the improved model, a more detailed assessment is performed by comprehensively extracting all piping connected to the reactor pressure vessel and apportioning the LOCA frequency in the containment vessel of NUREG-1829[9] according to the length and diameter of the pipe subject to evaluation. For the Excessive LOCA and Very Small LOCA, the literature data from NUREG-1829[9] and NUREG/CR-6928[10] are cited, respectively. Furthermore, for Multiple SRV Spurious Opening, the initiating event frequency is calculated by fault tree analysis.

(4) LOCA outside PCV
The evaluation method for IS-LOCA remains unchanged, using the calculation method from NUREG/CR-5862[11]. The added HELB is evaluated using pipe break probabilities from EPRI TR-3002000079[12] and fault tree analysis.

Table 2. Comparison of the initiating event Initiating frequency
: Previous Model vs. Enhanced Model

Initiating Event		Initiating Event Frequency (/reactor year)		Changes Δ%
		Previous Model	Enhanced Model	
General Transient	Non-isolation Event	1.7E-01	2.8E-01	65%
	Isolation Event	2.7E-02	1.9E-02	-30%
	Loss of Feedwater	3.7E-02	4.7E-02	27%
	Loss of Offsite Power	4.2E-03	5.2E-03	24%
	Inadvertent Opening of SRV	1.0E-03	9.4E-04	-6%
Support System Initiating Events	Loss of RHR Seawater System Function	1.4E-03	4.9E-03	243%
	Loss of AC Power	3.0E-04	4.2E-03	1300%
	Loss of DC Power	5.6E-04	3.0E-03	436%
	Loss of Auxiliary Cooling Water System	7.2E-04	4.7E-03	553%
	Loss of Instrument Air System	N/A	4.8E-03	N/A
	Loss of Electrical Room Cooling System	N\A	4.0E-03	N\A
LOCA inside PCV	Excessive LOCA	N/A	1.1E-08	N/A
	Multiple SRV Spurious Opening	N/A	2.2E-07	N/A
	Large LOCA	2.0E-05	1.6E-05	-21%
	Medium LOCA	2.0E-04	1.1E-04	-44%
	Small LOCA	3.0E-04	5.7E-04	90%
	Very Small LOCA	N/A	3.4E-03	N/A
LOCA outside PCV	IS-LOCA	8.3E-10	1.5E-08	1682%
	HELB	N/A	1.9E-03	N/A
Controlled Shutdown	Unplanned Shutdown	4.3E-02	9.2E-02	114%

3.2 System Reliability Analysis

The enhanced model expands the system equipment modeled to include severe accident countermeasures implemented post-Fukushima. Table 3 provides examples of newly modeled systems.

Table 3. Examples of Newly Modeled Systems

Function	Newly Modelled Systems	Remarks
Reactor Injection	Alternative High Pressure Injection System (AHPI)	Consideration of additional severe accident countermeasures
	Alternative High Pressure Injection System (ALPI)	
Containment Heat Removal	Alternative Recirculation Cooling System (ARC)	
	Filtered Containment Venting System (FCVS)	
Support System	Emergency Sea Water System (ESW)	
	Alternative AC Power Supply System	
	Alternative DC Power Supply System	
	SRV Nitrogen Supply System	
	Electrical Room Cooling System	Model added due to additional initiating events
	Instrument Air System	

3.3 Parameter Development

(1) Equipment Failure Rates
In the previous model, the general equipment failure data for domestic use published by Japan Nuclear Technology Institute[13] was used as is. The enhanced model references the latest data, the general equipment failure data for domestic nuclear plants published by the Central Research Institute of Electric Power Industry [14], and uses Bayesian updating with the equipment failure data from the Tokai No. 2 Nuclear Power Plant to estimate the equipment failure rate for individual plants.

(2) Common Cause Failure Parameters
The previous model used multiple U.S. literature sources due to the lack of Japanese domestic parameters. The enhanced model uses CCF Parameter Estimations 2015[15] as a more recent and systematic source of U.S. literature data.

3.4 Human Reliability Analysis

The previous model evaluated human error probabilities using the THERP method from NUREG/CR-1278[16]. The enhanced model applies EPRI's HRA Calculator Ver. 5.2[17], considering more detailed Performance Shaping Factors (PSF), procedural cues, response times, and time margins.

Additionally, the enhanced model evaluates Human Failure Events (HFE) related to initiating events, identified through plant information surveys, accident sequence analysis, and system reliability analysis. Human Error Probabilities (HEP) for each HFE are calculated using screening values or methods such as HCR/ORE, CBDTM, and THERP, considering their impact on system unavailability or accident sequence frequencies.

The consistency of obtained HEPs is verified by analyzing PSF characteristics and dependencies between multiple HFEs within the same accident sequence.

4. IMPACT ON QUANTIFICATION OF ACCIDENT SEQUENCES AND FUTURE EFFORTS

In the enhanced model, the newly added model for the loss of electrical room cooling system significantly impacts the quantification results of accident sequences. This is due to the conservative assumption that safety-related equipment connected to the emergency power bus is not expected to function immediately after the loss of electrical room cooling system.

However, in reality, there is a time delay in the rise of room temperature after the loss of electrical room air cooling system, and it is possible to avoid the loss of function of safety-related equipment by taking measures such as opening doors to mitigate the rise in room temperature.

Therefore, in the future, we aim to conduct a detailed room temperature evaluation during the loss of electrical room cooling system using analysis codes, and to consider the application of room temperature rise mitigation measures such as opening doors in actual operations, aiming to construct a more realistic model.

5. CONCLUSION

JAPC is working to enhance the PRA model of Tokai No.2 Power Station to improve safety.

The main changes in the enhanced PRA model include the selection and evaluation of initiating events, system reliability analysis, parameter development, and human reliability analysis.

As a future challenge, further consideration is required since the model for the loss of electrical room cooling system significantly impacts the quantification of accident sequences.

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