NRA Nuclear Regulation Authority

Current Status of Confirming the Appropriateness of the Licensees' PRA Models and the Expectations of the Regulatory Side

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*1: “Industry means NRRC at the CRIEPI and ATENA in this document.

1. Introduction

○ It has been established that nuclear regulatory inspections are to use risk information obtained from PRA.

○ Following discussions with licensees, the Nuclear Regulation Authority (“NRA”) has decided to use the licensees’ PRA models in nuclear regulatory inspections after reviewing its content.

○ Models for 12 PWRs and one ABWR have been reviewed, starting with the appropriateness confirmation of the Level 1 PRA model for Ikata Unit 3 in April 2017.

○ Today’s presentation will address the state of appropriateness confirmations, its challenges, and regulatory expectations concerning these confirmations.

2. Use of PRA in nuclear regulatory inspections

○ The nuclear regulatory inspection system is based almost entirely on the US ROP system. It is utilized to quantitatively evaluate the importance of inspection findings that may be beyond green.

○ Measures of risk importance for each component obtained in Level 1 PRA (FV, RAW) are used in the processes below.

- Selection of components on which to conduct walkdowns in routine inspections
- Selection of periodic operator inspections to be inspected in supervising periodic operator inspections
- Selection of components to be inspected in Comprehensive Engineering Team Inspections (“CETI”)

○ Inspections that focus on risk-important accident sequences are being considered for CETI.

3. Background to the decision to use licensees' PRA Models

- If both the regulator and licensees have their own PRA model, the difference in models could lead to differences in important evaluations of inspection findings, which cannot proceed to the discussion of importance evaluation.

- It is difficult for the regulatory side to create and maintain PRA models for multiple plants as we cannot obtain detailed and up-to-date information about operations and components in a timely manner.

3. Background to the decision to use licensees' PRA Models

○ In the discussions with licensees on the introduction of the new inspection system, the operators proposed to provide their PRA models.
The NRA decided to accept this proposal, as this helps the regulatory and licensees start at the same page in discussions on evaluating importance when using the same PRA model, and to confirm the appropriateness of the model for itself.

○ The NRA has created and is using a guide for confirming the appropriateness of licensees' Level 1 and Level 1.5[*3] PRA models referencing the US ASME/ANS Standard[*1] and the Atomic Energy Society of Japan's PRA Standard[*2].

*1: ASME/ANS, Addenda to ASME/ANS RA-S-2008—Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, ASME/ANS RA-Sb-2013, The American Society of Mechanical Engineers, 2013

*2: Atomic Energy Society of Japan, "Implementation Standard for Probabilistic Risk Assessment of the Nuclear Power Plant's Operating State (ver. Level 1 PRA) :2013", AESJ-SC-P008, August 2014 and "Implementation Standard for Probabilistic Risk Assessment of the Nuclear Power Plant's Operating State (ver. Level 2 PRA): 2016, AESJ-SC-P009, June 2016

*3: Assessment up to the frequency of containment vessel function loss

4. Record of appropriateness confirmation

○ The confirmation of Level 1 PRA models of plants that have conformed to the new regulatory standard are prioritized. Issues identified in the verification are compiled into a list of mid-to-long term areas for improvement.

Plant	Level 1 (Date reported to the NRA)	Level1.5 (Date reported to the NRA)
Ikata Unit 3 (PWR)	Completed (March 2020)	Completed (July 2021)
Ohi Units 3 and 4 (PWR)	Completed (Feb. 2021)	Being confirmed
Genkai Units 3 and 4 (PWR)	Completed (Feb. 2021)	Being confirmed
Takahama Units 3 and 4 (PWR)	Completed (July 2022)	Being confirmed
Sendai Units 1 and 2 (PWR)	Completed (July 2022)	Being confirmed
Takahama Units 1 and 2 (PWR)	Completed (Dec. 2023)	To be confirmed
Mihama Unit 3 (PWR)	Completed (Dec. 2023)	To be confirmed
Kashiwazaki-Kariwa Unit 7 (BWR)	Completed (Sept. 2024)	To be confirmed
Onagawa Unit 2 (BWR)	Being confirmed	To be confirmed

5. Mid-to-long term areas of improvement that need to be discussed with the industry

Mid-to-long term areas of improvement and the NRA plans[*1] are as follows.

No.	Mid-to-long term areas of improvement	Specifics	NRA plans going forward
1	Domestic component failure data (Started discussions)	①Kashiwazaki-Kariwa Unit 7 is using data compiled by JANSI in 2016. ②Licensees are supposed use the most up-to-date data[*2] (Published by the NRRC in September 2021) The following improvements are being made by the industry. ①The NRA reviewed the data collection guide published by the NRRC in May 2023 and presented the NRRC with 10 findings[*3]. ②The NRRC started revising the data collection guide based on ① above. ③In the meeting with ATENA and NRRC on June 20, 2024[*4], they reported that the data collection guide will be revised over a period of three years to address NRA findings and Technical Advisory Committee of NRRC findings, and have it reviewed by the industry, as part of efforts to continuously improve the guide and its operation. Data collection using the revised guide is expected to start in FY2027.	Confirm the state of the NRRC and ATENA discussions.

*1:Excerpts from the appropriateness confirmation of the Kashiwazaki-Kariwa NPS Unit 7 Level 1 PRA model reported to the NRA on September 18, 2024, were modified and corrected.

*2:Estimates of generic component reliability parameters for PRA in domestic nuclear power plants, Research Report: NR21002, September 2021, CRIEPI

*3:Scope of failure mode data collection, handling of human error, period subject to collection, handling of startup failure, handling of external factors, exposure data collection methods, summaries of component failure cases, component grouping, failure mode selection cases from Addendum D, others

*4:https://www.da.nra.go.jp/view/NRA100003431?contents=NRA100003431-002-002#pdf=NRA100003431-002-002

5. Mid-to-long term areas of improvement that need to be discussed with the industry

No.	Mid-to-long term areas of improvement	Specifics	NRA plans going forward
2	Review of loss of off-site power frequency and power restoration failure probability (Started discussions)	①Kashiwazaki-Kariwa Unit 7 evaluated loss of off-site power (LOOP) frequency using only BWR operating history when it should have included PWR operating history. ②In the Kashiwazaki-Kariwa Unit 7, the probability of restoration failure up to 24 hours differs by one order of magnitude due to the difference between the PWR to used data and the evaluation methods for the probability of restoration failure after loss of off-site power. ③Licensees have stated they will use NRRC data in the model updates they are currently performing. Improvements made in the industry are as follows. ①The NRRC created a report that calculated the LOOP frequency from off-site power loss history in domestic PWRs and BWRs created in March 2024. But we have heard that even if all external transmission lines are lost, as long as house load operation is successful it is not considered to be LOOP, and in that case, the LOOP frequency will be low for plants that do not include house load operation in their designs. ②A method for evaluating the probability of off-site power restoration failure is currently developing by NRRC.	①Revise and discuss methods of calculating frequency considering the availability of house load operation. ②Confirm the state of discussions at the NRRC.

5. Mid-to-long term areas of improvement that need to be discussed with the industry

Mid-to-long term areas of improvement	Specifics	NRA plans going forward
Improvements on using domestic component failure data (Clarifying the approach)	①Kashiwazaki-Kariwa Unit 7 uses the failure probability of components that are similar in structure or configuration to substitute for the failure rates for components for which they have no data. For example, the emergency diesel generator failure rate is used as the air-cooled gas turbine generator failure rate. ②The calculation method for common cause failure of basic software and application software for digital safety protection systems and the source for digital component failure data differ between PWRs and BWRs. ③ The grouping of common-cause failures between independent systems is based on the commonalities in inspection and maintenance. However, since maintenance workers are sometimes selected for each type of equipment and sometimes for each system, it is desirable to set common-cause failure groups based on this difference. ④There is a difference in the rates used for component startup failure probability between PWRs and BWRs; demand failure probability is used in PWRs while time failure rate is used in BWRs.	The following will be required of the industry and progress will be confirmed as needed. ①Clarify and articulate the approach to take when substituting data from different components from the perspective of reliability that takes into consideration environmental factors and maintenance management. ②Review the approach as there is a difference in the approach taken in PWRs and BWRs. ③Organize in ② above. ④Review the approach as there is a difference in the approach taken in PWRs and BWRs.

5. Mid-to-long term areas of improvement that need to be discussed with the industry

No.	Mid-to-long term areas of improvement	Specifics	NRA plans going forward
4	Clarify the evaluation scope of internal event PRA and internal inundation PRA (Clarifying the approach)	The Internal Inundation PRA Standard*1 of the Atomic Energy Society of Japan ("AESJ") cites steam blowout due to high energy pipe failure as an example of inundation mode. PWR internal event PRA has not evaluated the effects of steam blowout on other components assuming it will be confirmed in internal inundation PRA. However, the Level 1 PRA model of Kashiwazaki-Kariwa Unit 7 evaluates the impact of steam blowout due to high energy pipe failure of the main steam pipes on other components.	Request the industry organize which PRA is going to assess the effects of steam blowout on other components, as it shouldn't depend on reactor type.
5	Definition of unavailability and discussion of the scope of modeling (Clarifying the approach)	The 2022 AESJ Internal Event PRA Standard*2 defines unavailability as "probability that a SSCs will be unable to perform its function due to causes other than failure during the assessment period." The Kashiwazaki-Kariwa Unit 7 Level 1 PRA model, however, includes failure in unavailability based on the AESJ definition of terms related to risk assessment (2018)*3. It also takes into a consideration initiating events during shutdown operation based on the technical specifications. This is inconsistent with the latest standards and expands the scope of modeling.	The definition of unavailability in the Kashiwazaki-Kariwa Unit 7 model is inconsistent with the latest AESJ standards and expands the modeling scope. The NRA requests that the industry discuss these issues at AESJ.

*1: Implementation Standard for Probabilistic Risk Assessment Initiated by Internal Inundation in a Nuclear Power Plant: 2012 (AESJ-SC-RK005: 2012)
*2: Standard for Probabilistic Risk Assessment Initiated by Internal Event in a Nuclear Power Plant (ver. Level1 PRA): 2022 (AESJ-SC-RK010:2022)
*3: Definition of Terms Used in Nuclear Facilities Risk Assessment Standards: 2018 (AESJ-SC-RK003:2018). This Standard defines unavailability as "probability that a SSCs will be unable to perform its function during the assessment period. Note, this may include component failure rates."

5. Mid-to-long term areas of improvement that need to be discussed with the industry

No.	Mid-to-long term areas of improvement	Specifics	NRA plans going forward
6	Development of human error assessment method for touch panel control panels (To be discussed)	Kashiwazaki-Kariwa Unit 7 is using human error rates for existing analog control panels as the human error probability for touch panel control panels. It is desirable to collect and analyze operator data from touch panel control panels and evaluate the probability of human error.	This was also identified as a mid-to-long term area of improvement in the appropriateness confirmation of Mihama Unit 3 and Takahama Units 1 and 2 Level 1 PRA model. We will develop common understanding in discussion with the NRRC and ATENA, and request that these issues be addressed in order of priority.
7	Development of human error probability using Japanese operator training data (To be discussed)	The HRA Calculator that licensees are using to calculate human error is based on US operator data. Because differences in culture between the U.S. and Japan could affect operator human error probability, Japanese operator data should be gathered, analyzed, and checked for HRA calculator applicability. If the calculator based on US data is not a good fit with Japanese data, more Japanese data should be gathered as needed.	We will develop common understanding in discussion with the NRRC and ATENA, and request that these issues be addressed in order of priority.

5. Mid-to-long term areas of improvement that need to be discussed with the industry

No.	Mid-to-long term areas of improvement	Specifics	NRA plans going forward
8	Development of a method for assessing human error when continuing operation during mission time (To be discussed)	There currently is no method for evaluating human error probability for operators where pump startup and shutdown need to be repeated when manually starting up or shutting down backup pumps or valves. It is desirable to be able to evaluate the probability of human error even for such repeated operations.	We will develop common understanding in discussion with the NRRC and ATENA and request that these issues be addressed in order of priority.
9	Development of maintenance-related parameter (To be discussed)	Out-of-standby probability in maintenance work at Kashiwazaki-Kariwa Unit 7 is calculated by multiplying the failure rate by 10 and the average out-of-standby time, based on US practices. It is preferable to calculate this based on actual work time.	
10	Development of a failure probability assessment method for intermittent operation components (To be discussed)	It is desirable to develop a method for intermittent operation components, such as the main steam relief safety valve that automatically depressurizes. The current challenge is that if using demand failure probability, the failure probability increases when the component actuates many times, and if the time failure rate is used, the time interval after the first time becomes shorter and can be ignored, so the failure probability becomes too small.	

6. Challenges in appropriateness confirmation

(1) Workload for confirmation

Confirmation of the first PWR PRA model confirmed Ikata Unit 3 involved 15 meetings. This confirmation process has been streamlined by focusing on the differences with already confirmed plants. Only 4 meetings were required for Mihama Unit 3 and Takahama Units 1 and 2, PWRs whose models were confirmed recently.

At the same time, the number of items that require discussion as an industry, such as the component failure rate and off-site power loss frequency, is increasing in individual plant confirmation, and the workload for appropriateness confirmation overall has not decreased.

(2) Differences in the perception of the level of detail required in a model

The approach to the modeling scope has been a target of discussion, as licensees will not make changes to the model if it has a small impact on CDF, but they will model things if they deem them necessary, even if they have a small impact on CDF. There is a discrepancy in the licensees' and regulatory's sense of what should be modeled using what measurement.

6. Challenges in appropriateness confirmation

(3) Differences in practices between PWR and BWR

Matters that do not depend on reactor type need to be standardized between PWRs and BWRs and codified, such as whether to assess the impact of steam blowout from high energy pipe failure in an internal event PRA or internal inundation PRA.

(4) Catching up with changes to the PRA model

It is our understanding that the licensees update the PRA model as needed when components that may affect the PRA model are built or modified, such as the specialized safety facility or the replacement of the seawater pump with a model that does not need lubricants. There is no mechanism in place for the regulatory side to confirm those model changes in a timely and efficient manner.

(5) Not enough PRA engineers

Even as experienced staff retires and is reappointed, skills and knowledge are not being smoothly passed onto younger staff. Appropriateness confirmation, especially for Level 1.5 PRA, is understaffed.

(1) Promoting the further use of PRA results in the field (from PRA to "show" to a PRA to "use")

In the review stage, PRA was used to extract accident sequence groups that could affect the plant.

In the operating stage, staff's sensitivity to risk can be fostered by actively using the PRA results in improving the plant, for example, reflecting information such as component risk importance and risk significant accident sequences on plant maintenance and operator training programs.

Furthermore, the incorporation of experience gained through application in the field into the model will allow for refinement of the model so that it is more effective.

7. Expectations of the regulatory side

(2) Building a mechanism to voluntarily and continuously improve the model

It is expected that licensees will utilize the results of reviews by overseas experts (including follow-up reviews) and their experience of applying the model in the field to continuously improve the model and data and establish a process for voluntarily improving the model and ensuring data quality.

Furthermore, when the NRA is confirming the appropriateness of the Licensees' PRA Models, it is desirable to show a model that overseas experts have reviewed.

(3) NRRC’s active involvement in solving challenges

There should be a forum for the NRRC to take the lead and discuss with the NRA and licensees key discussion points using results from overseas experts’ reviews of individual plants. This could help the NRA and licensees develop a shared sense of the level of detail required in a PRA model and resolve challenges.

Moreover, sharing industry and regulatory information with the AESJ and having them discuss issues may also help resolve challenges and ensure the active use of PRA.

7. Expectations of the regulatory side

(4) Engaging actively in resolving common challenges

Component failure rates are being drastically improved, by revising the data collection guide while reviewing it as an industry.

Furthermore, as the probability of human error is generally greater than the probability of component failure, it is desirable to improve the probability of human error in the future for digital central control panels, etc.

8. For PRA to gain trust widely

Only a limited number of people know the assumptions made in PRA. Trust may be hard to gain unless the limits of these assumptions are widely recognized.

In addition, NRA faces the challenge of having very few PRA engineers available to utilize risk information (especially Level 1.5.)

○ Explanations that include assumptions (limits)

The current PRA only presents users with results (e.g., CDF), without explaining the assumptions made in the assessment, turning it into a black box. First of all, it is desirable to explain the limitations (assumptions) according to the purpose of use to PRA users.

○ Development of PRA engineers at the NRA

The number of staff in charge of PRA has been increased and experienced staff from manufacturing backgrounds are conducting OJT on them. Skills and knowledge are being passed down to other staff from the staff in charge of the appropriateness confirmation of Ikata Unit 3 Level 1.5 PRA.

9. Summary

- PRA information whose quality has been ensured is needed to conduct inspections using risk information.

- Performance Category II, as licensees have confirmed the conformity through an overseas expert review, is an acceptable level of detail for most applications in the US Regulatory Commission's RG1.200*1.

- Licensees will be expected to voluntarily and continuously improve the PRA quality, including their models, by updating the models, for example. The NRA wishes to efficiently confirm the appropriateness of the models by focusing on individual items after checking on the progress made in the improvement process.

- The NRA wishes to cooperate with the industry and AESJ to resolve mid-to-long term areas for improvement.

*1: C.2.1 of the "ACCETABILITY OF PROBABILISTIC RISK ASSESSMENT RESULTS FOR RISK-INFORMED ACTIVITIES" states that, "In general, the NRC staff anticipates that performance category II is the level of detail that is acceptable for the majority of applications."