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SUBJECT: STATUS OF IKATA PROBABILISTIC RISK ASSESSMENT MODELS AND RECOMMENDED RESEARCH

Dear Dr. Apostolakis:

During the fourth meeting of the Technical Advisory Committee of the Nuclear Risk Research Center (NRRC), October 26-30, 2015, we met with representatives of Shikoku Electric Power Company, Ltd. (Yonden) and their contractor, Mitsubishi Heavy Industries, Ltd. (MHI) to receive a briefing on the status of the probabilistic risk assessment (PRA) models for Ikata Nuclear Power Plant Unit 3.

Yonden has volunteered Ikata Unit 3 to be the lead pressurized water reactor (PWR) to focus NRRC support and guidance for the development of a plant-specific Level 2 PRA that initially evaluates the risk from internal initiating events, seismic events, and tsunamis during plant power operation. The eventual goal is to extend that PRA to include a full-scope evaluation of the risk from all internal events, all external hazards, and all major site radiological sources during all operating modes that is consistent with the current international state-of-practice in PRA methods, models, and technical quality.

CONCLUSION AND RECOMMENDATION

1. The scope, level of detail, and technical quality of the Ikata Unit 3 PRA event sequence models, initiating events, system fault trees, and plant-specific data for the analyses of internal initiating events have advanced substantially beyond the models that we examined one year ago. The PRA team is making excellent progress toward the development of a state-of-practice PRA for the evaluation of internal initiating events. Additionally, these models will also provide a solid basis for PRA models for other hazards such as internal flooding, fires, and external events.

2. The NRRC should work with the Japanese nuclear industry to develop research, methods, and guidance for improved treatment of the following phenomena in all

relevant PRA models. These topics are not associated uniquely with Ikata Unit 3 or its PRA.

- Consequential failures of steam generator tubes during certain severe accident scenarios
- Initiating events caused by failures of multiple tubes in a steam generator
- Leakage rates through isolation valves and low pressure systems exposed to reactor coolant system pressure
- Treatment of common cause failures in models for the quantification of plant-specific initiating event frequencies
- Treatment of national, regional, and site-specific data for the causes, frequency, and duration of offsite power failures

BACKGROUND

Our November 1, 2014, letter report on "Suitability of Models for Ikata Site Probabilistic Risk Assessment" concluded that the event sequence models that were developed to support the Ikata Unit 3 Periodic Safety Review process provided a good technical foundation for extension and eventual development of a full-scope Level 2 PRA. We also noted that we will perform periodic reviews of technical tasks in the PRA, its supporting analyses, and results, and report our conclusions and recommendations to the NRRC project team.

We have had extensive and very detailed ongoing technical exchanges with the Ikata PRA team over the last year, including full Technical Advisory Committee briefings in January and May. The topics of those exchanges have covered the Level 1 PRA event trees, the selection of generic and plant-specific initiating events, the fault tree models for a sample of frontline and support systems, and the compilation of plant-specific data. This letter report provides a brief summary of the current status of the Ikata PRA, and it contains recommendations for focused research activities that will support enhanced PRAs for other plants.

DISCUSSION

Status of Ikata Unit 3 PRA Models

The scope and level of detail in the Level 1 PRA event trees are consistent with current state-of-practice event sequence models. Event trees have been developed for a full range of initiating events caused by plant transients, losses of offsite power, support system failures, loss of coolant accidents (LOCAs), steam generator tube ruptures, and LOCAs through interfacing systems that bypass the containment. The models account explicitly for conditions such as failures of the reactor trip (ATWS) and turbine trip functions, consequential failures of electric power supplies, overcooling transients, failures of the reactor coolant pump seals, LOCAs through the pressurizer relief valves, and mismatches between reactor coolant system makeup and letdown flows. We have not yet examined the Level 2 PRA models for accident sequence progression and containment performance. Nevertheless, we have noted that the Level 1 event trees are structured to provide a direct interface to those models. The Level 1 event sequence models currently include limited credit for the additional equipment and enhanced accident mitigation capabilities that are being implemented at Ikata in response to the accidents at Fukushima Daiichi. Work is in progress to more fully examine a few detailed technical issues and to refine elements of the timing and success criteria for some event sequences, based on plant-specific best-estimate thermal-hydraulic analyses. Our interactions with the PRA team are continuing, and we do not anticipate any major technical impediments.

The PRA team is performing detailed failure modes and effects analyses (FMEAs) to identify a full scope of plant-specific initiating events. This effort is especially important for failures of support systems such as cooling water supplies, instrument air supplies, ventilation and room cooling, high voltage and low voltage AC power supplies, and DC power supplies. This work is currently in progress. The scope and level of detail in the FMEAs are consistent with current state-of-practice methods. The team will use the FMEAs to identify initiating events that are uniquely associated with specific features of the Ikata Unit 3 plant design, system configurations, and operating practices. We will continue to review the results of these analyses as the FMEAs are completed, and the final set of initiating event groups is developed.

Our examinations of the system fault tree models have involved a deep "vertical slice" focused review of the fault trees for a representative frontline system. That examination also includes the models for the associated support systems (e.g., AC power, DC power, actuation signals, cooling water, room cooling and ventilation, etc.) to check how those systems and dependencies are integrated into the PRA. This level of review is not intended to be comprehensive. However, it provides a good understanding of how the models are developed and their important supporting assumptions. Our technical exchanges with the PRA team in this area are in progress. We have not identified any significant concerns with the scope of the fault tree models, their level of detail, failure modes, and treatment of issues such as the unavailability of equipment due to testing and maintenance, personnel errors during those activities, and common cause failures. Our discussions have also addressed possible enhancements to the models for normally-operating systems to facilitate their future use for risk-informed applications. Those enhancements do not directly affect the capability of the current models to quantify the average plant risk. However, they are intended to improve use of the models as engineering tools for integrated plant risk management. Fault tree models are also being developed to quantify the frequency of initiating events that are caused by plant-specific support system failures. That approach is consistent with current state-of-practice guidance and methods.

The PRA team is using the Ikata Unit 3 Equipment Maintenance Information Database (EAM) to derive detailed plant-specific data for equipment failures and unavailability due to maintenance. The available data cover approximately seven years of operating experience for all plant systems and equipment. Other sources of information such as plant operating and testing records are being used to derive data for component demands, operating hours, and standby exposure periods, which are not directly available in EAM. This effort is in progress, and it is very encouraging. It is consistent with the best practices in contemporary PRAs. We will continue to follow the team's progress as the databases are developed and more information becomes available.

Needs for Focused Research, Methods, Guidance, and Data

During our discussions and technical exchanges with the Ikata PRA team, the following topics were identified as potential needs for focused research, methods, guidance, and supporting data. These topics are not associated uniquely with Ikata Unit 3 or its PRA. Therefore, the NRRC should work with the Japanese nuclear industry to develop the necessary information and guidance to facilitate their consistent treatment in all relevant PRA models.

Consequential Failures of Steam Generator Tubes

Research performed by the U.S. Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research has indicated that certain severe accident scenarios may result in conditions that increase the potential for thermally-induced failures of steam generator tubes. If they occur, these failures can result in a containment bypass event and a release of fission products to the environment. The main accident scenarios of interest are those that lead to core damage with high reactor pressure, a dry steam generator, and low steam generator pressure (so-called "high-dry-low" conditions).

Because these conditions are a consequence of only specific severe accident scenarios, they do not affect the frequency of core damage. However, they affect the scope and the logic structure of the Level 1 PRA models, because those models must be developed to consistently identify and quantify these scenarios for their further evaluation in the Level 2 PRA analyses.

Research is needed to examine the applicability of the U.S. NRC's analyses and conclusions to Japanese nuclear power plant steam generator designs, fabrication, and materials. Guidance is also needed to ensure that the integrated Level 1 and Level 2 PRA models are developed in a manner to ensure that this issue is evaluated consistently, as needed on a plant-specific basis.

Initiating Events Caused by Multiple Steam Generator Tube Failures

Contemporary international PRAs have evaluated the risk from initiating events that involve failures of multiple tubes in a single steam generator. The models for those events are logically and functionally very similar to those for single tube failures. However, the detailed success criteria and the time windows that are available for operator actions to mitigate the event are typically much more challenging.

There is substantial uncertainty about the frequency of these multiple tube rupture events. Estimates are available from studies that have been conducted in different countries over a period of several years. Research is needed to perform a survey of the international literature, the supporting technical analyses, and the available data to determine their applicability to the Japanese steam generator designs, operating practices, and inspection programs.

Enhanced Models for Interfacing System LOCA Events

Interfacing system LOCA events are typically not an important contribution to the frequency of core damage. However, experience from numerous PRAs has shown that they can account for a very important contribution to offsite releases. The equipment and operator actions that can be used to effectively mitigate the consequences from these events depend on both the size of the LOCA and its location.

Limited research has been performed to examine the likelihood of leakage or rupture of various types of isolation valves when they are exposed to large differential pressures. Limited research has also been performed to examine the vulnerabilities of low pressure piping, welds, equipment, seals, etc. to leakage or failure when they are exposed to pressures that are beyond their rated designs. This research has been used in contemporary state-of-practice PRAs to develop numerical correlations for the frequency and size of interfacing system LOCA events, the locations of low pressure system failures, and the consequential reactor coolant leakage rates. These correlations are then used to determine the functional success criteria and time windows that are available for operator actions to prevent core damage or to mitigate offsite releases.

Research is needed to perform a survey of the international literature, the supporting technical analyses, and the available data to determine their applicability to Japanese system designs. This research may also require re-examination of selected components, system material properties, or the development of focused probabilistic models to refine previous estimates or to better characterize the associated uncertainties.

Treatment of Common Cause Failures for Plant-Specific Initiating Events

As noted previously, fault tree models are being developed to quantify the frequencies of initiating events that are caused by plant-specific support system failures. Those fault trees are then integrated with the corresponding event sequence models for plant response to each initiating event. This approach correctly accounts for component failures that cause the initiating event and the unavailability of that equipment in the models for event mitigation. It also accounts for potential dependencies between human errors that may contribute to the initiating event and operator performance after the reactor trip occurs. This integrated approach is consistent with current state-of-practice guidance and methods.

Experience has shown that the treatment and quantification of common cause failures in the integrated models can have a very significant effect on the initiating event frequency and the conditional probability for failures of similar equipment that is needed for post-trip event mitigation. The issue is especially important for normally-operating equipment such as pumps, ventilation chillers, air compressors, fans, etc. This is a common challenge for all PRAs.

Contemporary PRAs have applied different methods, data, and assumptions for the quantification of these common cause failures. Guidance is needed to ensure that these failures are modeled and quantified consistently in all Japanese PRAs.

Data for the Frequency and Duration of Loss of Offsite Power Events

Loss of offsite power events are typically important contributors to overall plant risk. International operating experience has shown that the causes, frequencies, and durations of power failures are often affected significantly by national, regional, and local factors. For example, typical categories for the causes of power failures include grid-related events, weather-related events, and switchyard-related events[1]. Both the frequencies and the durations of offsite power failures are often correlated to these causes, and they may vary significantly from site-to-site. Therefore, it is essential that each PRA should consistently account for all available operating experience, design features, and data that affect the evaluation of loss and recovery of offsite power supplies at the particular site.

Research and coordinated data collection are needed to develop a comprehensive database for the causes, frequencies, and durations of offsite power failures that can be used in a consistent manner for site-specific PRAs throughout Japan. The first step in that process requires a clear definition of the scope of causes and analytical boundaries for each category of loss of offsite power initiating event.

The data for these analyses may not be readily available to each nuclear power plant or its operating company. For example, relevant data may be available from sources such as national grid reliability studies, regional power system operating data, and experience from other large municipal or industrial consumers near each nuclear power plant site. Therefore, such data should be compiled as a unified Japanese industry activity.

We look forward to continuing our review of this milestone NRRC project and its key technical tasks as it evolves toward a state-of-practice full-scope PRA.

Sincerely,

John W. Stetkar

John W. Stetkar
Chairman

[1] Some databases also cite frequencies for "plant-centered" events. However, experience has shown that the frequencies and consequences from those events are influenced very strongly by the plant-specific electric power system design (e.g., transformers, operating and standby bus alignments, automatic bus transfers, etc.). Therefore, state-of-practice PRAs often develop plant-specific models to quantify the frequencies of those plant-centered events, in a manner similar to initiating events caused by other support system failures.

REFERENCES

1. "Suitability of Models for Ikata Site Probabilistic Risk Assessment," Technical Advisory Committee letter report to Dr. George Apostolakis, November 1, 2014.

2. "Ikata Unit 3 Probabilistic Risk Assessment, Level 1 Event Trees," Shikoku Electric Power Co. Ltd., December 26, 2014, Proprietary.

3. "Ikata Unit 3 Probabilistic Risk Assessment, Fault Trees for Auxiliary Feedwater System, High Pressure Injection System, and Support Systems," Shikoku Electric Power Co. Ltd., December 26, 2014, Proprietary.

4. "Ikata Unit 3 Level 1 PRA for Internal Events at Power Event Trees," Shikoku Electric Power Co. Ltd. Presentation to NRRC Technical Advisory Committee, January 19, 2015, including Confidential materials.

5. "Selection of Initiating Events for Ikata Unit 3 PRA for Internal Event at Power," Shikoku Electric Power Co. Ltd. Presentation to NRRC Technical Advisory Committee, January 20, 2015.

6. "Ikata Unit 3 Level 1 PRA for Internal Events at Power Fault Trees," Shikoku Electric Power Co. Ltd. Presentation to NRRC Technical Advisory Committee, January 20, 2015, including Confidential materials.

7. "Comments and Questions on Ikata Unit 3 Level 1 PRA Event Tree Models," John W. Stetkar, January 25, 2015, Confidential.

8. "Responses to Comments and Questions on Ikata Unit 3 Level 1 PRA Event Tree Models," Shikoku Electric Power Co. Ltd., May 11, 2015, Confidential.

9. "Ikata 3 PRA Event Trees and Fault Trees," Shikoku Electric Power Co. Ltd. Presentation to NRRC Technical Advisory Committee Working Group 1, May 26, 2015, including Confidential materials.

10. "Ikata 3 PRA Data Analysis," Shikoku Electric Power Co. Ltd. Presentation to NRRC Technical Advisory Committee Working Group 1, May 27, 2015, including Confidential materials.

11. "Comments and Questions on Ikata Unit 3 Level 1 PRA Event Tree Models," John W. Stetkar, May 30, 2015, Confidential.

12. "Comments and Questions on Ikata Unit 3 PRA Auxiliary Feedwater Fault Tree Models," John W. Stetkar and James C. Lin, June 3, 2015, Confidential.

13. "Responses to Comments and Questions on Ikata Unit 3 Level 1 PRA Event Tree Models," Shikoku Electric Power Co. Ltd., September 8, 2015, Confidential.

14. "Responses to Comments and Questions on Ikata Unit 3 PRA Auxiliary Feedwater Fault Tree Models," Shikoku Electric Power Co. Ltd., September 20, 2015, Confidential.

15. "Comments and Questions on Ikata Unit 3 Level 1 PRA Event Tree Models," John W. Stetkar, October 14, 2015, Confidential.

16. "Ikata Unit 3 Internal Event Level 1 Full Power PRA Status Update," Shikoku Electric Power Co. Ltd. Presentation to NRRC Technical Advisory Committee, October 27, 2015, including Confidential materials.

17. "Ikata Unit 3 Internal Event Level 1 Full Power PRA Status Update, Upgrade Initiating Events," Shikoku Electric Power Co. Ltd. Presentation to NRRC Technical Advisory Committee, October 27, 2015, including Confidential materials.

18. "Ikata Unit 3 Internal Event Level 1 Full Power PRA Status Update, Upgrade PRA Parameters," Shikoku Electric Power Co. Ltd. Presentation to NRRC Technical Advisory Committee, October 27, 2015, including Confidential materials.

19. U.S. Nuclear Regulatory Commission, "Consequential SGTR Analysis for Westinghouse and Combustion Engineering Plants With Thermally-Treated Alloy 600 and 690 Steam Generator Tubes," Draft NUREG-YYYY, Revision 46, February 2015 (ML15028A283).

20. U.S. Nuclear Regulatory Commission, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process," NUREG-1829, Volumes 1 and 2, April 2008 (ML082250436, ML081060300).

21. Electric Power Research Institute, "Support System Initiating Events, Identification and Quantification Guideline," EPRI 1016741, December 2008.