Technical Advisory Committee of the Nuclear Risk Research Center
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SUBJECT: INTERIM REVIEW OF RISK-INFORMED CONTAINMENT VESSEL LEAK RATE TESTING GUIDELINE

Dear Dr. Apostolakis:

During the 21st meeting of the Technical Advisory Committee of the Nuclear Risk Research Center (NRRC), November 18-22, 2024, we met with representatives of the NRRC staff to discuss the current status of the guidance for implementation of risk-informed containment vessel leak rate testing programs at Japanese nuclear power plants. This letter report documents our review of the August 2024 version of the "Feasibility Study on Risk-Informed Reactor Containment Vessels Test Interval Extension in Japan," and additional information about the guidance that was discussed during our meeting.

This is a revision of our November 25, 2024 letter report on this topic. It clarifies our understanding of one item that was identified by your staff after our original letter report was issued.

CONCLUSIONS AND RECOMMENDATIONS

1. The Guideline contains comprehensive guidance that addresses all of the fundamental principles of risk-informed decision-making.

2. Use of containment failure frequency as the primary risk metric for these evaluations is appropriate. The applied risk acceptance criteria are consistent with Japanese standards and criteria that are used by the Nuclear Regulation Authority in the Reactor Oversight Process. Use of containment failure frequency in these evaluations, rather than large early release frequency, is more conservative than the U.S. practice.

3. The methodology for quantifying the change in risk due to extension of the Type A containment vessel leak rate test interval is consistent with methods and

guidance that are used in the United States and endorsed by the U.S. Nuclear Regulatory Commission.

4. The Guideline appropriately indicates that the analyses should account for the risk from all internal and external hazards during all plant operating modes when containment integrity is required.

5. The following enhancements should be made before the Guideline is issued for inclusion in Japan Electric Association Guideline JEAC4203:

   - The Guideline should provide guidance that addresses the provisional performance objective to confirm that the frequency of a release of more than 100 terabecquerels (TBq) of cesium-137 (Cs-137) remains below 10^{-6} event per year.

   - The Guideline should provide methods for evaluating uncertainties related to the completeness and technical quality of the applied PRA models, or it should recommend the use of relevant methods from internationally accepted guidance.

   - The technical bases for some numerical values and the supporting calculations for some intermediate results should be confirmed.

   - The Guideline should provide guidance for the identification and implementation of robust and effective compensatory measures which are directly related to reducing the potential impact of extending the test interval.

BACKGROUND

This Guideline provides the methods and guidance to support a risk-informed alternative to the Type A containment vessel leak rate testing (CVLRT) intervals that are specified in Japan Electric Association (JEA) Guideline JEAC4203-2017. These methods will be included in a planned update to the JEA guidance. That guidance must be approved by the Nuclear Regulation Authority (NRA) before it can be used by the utilities to change the test intervals.

Option B in Appendix J to Part 50 of Title 10 of the U.S. Code of Federal Regulations (10 CFR 50, Appendix J) allows nuclear power plant licensees to implement a risk-informed, performance-based containment leak rate testing program. Regulatory guidance for risk-informed decisions that affect a plant's licensing basis is contained in U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.174. The Atomic Energy Society of Japan (AESJ) Standard AESJ-SC-S012E:2019 provides guidance for integrated risk-informed decision-making in Japan, based on the experience in the U.S. and other countries.

The U.S. industry guidance and detailed technical methods to support risk-informed changes to containment leak rate testing programs are contained in Nuclear Energy Institute (NEI) Report NEI 94-01 and Electric Power Research Institute (EPRI)

Report 1018243. These methods have been endorsed by the U.S. NRC in Regulatory Guide 1.163.

Utilities in the U.S. have used these risk-informed methods to justify extensions of their Type A integrated containment leak rate testing intervals to 15 years. Those extensions provide numerous benefits, including reductions in personnel radiation doses, reductions in plant outage times, and reductions in testing-related stresses that are applied to the containment structures. The risk-informed, performance-based programs provide assurance that these benefits are achieved while maintaining a very high level of containment reliability to mitigate the consequences of potential accidents.

The methodology and guidance in this Guideline are derived from the methods that are used in the U.S. and endorsed by the U.S. NRC. Our May 24, 2023 letter report on "Risk-Informed Changes to Containment Vessel Leak Rate Testing Interval" provided our conclusions and recommendations about a preliminary version of the proposed methodology. This report summarizes our review of the current Guideline.

DISCUSSION

The following sections summarize our comments and the technical bases for each item in our Conclusions and Recommendations.

Principles of Risk-Informed Decision-Making

Regulatory Guide 1.174 describes the following five fundamental principles of integrated risk-informed decision-making:

- Principle 1: The proposed licensing basis change meets the current regulations unless it is explicitly related to a requested exemption.
- Principle 2: The proposed licensing basis change is consistent with the defense-in-depth philosophy.
- Principle 3: The proposed licensing basis change maintains sufficient safety margins.
- Principle 4: When proposed licensing basis changes result in an increase in risk, the increases should be small and consistent with the intent of the Commission's policy statement on safety goals for the operations of nuclear power plants.
- Principle 5: The impact of the proposed licensing basis change should be monitored using performance measurement strategies.

The Guideline contains an excellent discussion of these fundamental principles, including supplemental information from the AESJ Standard and other international references, such as IAEA TECDOC-1909. The Guideline provides guidance and examples of how the risk-informed decision to extend the Type A testing interval should account for the integrated consideration of all five principles.

Use of Containment Failure Frequency as the Primary Risk Metric

Regulatory Guide 1.174 and Regulatory Guide 1.163 use large early release frequency (LERF) as the primary risk metric for accidents that result in a release of radionuclides from the reactor containment. The Guideline uses containment failure frequency (CFF) as that metric. The CFF metric includes all sizes and timings of containment releases (i.e., small to large, and early to late). Thus, the CFF is an upper bound for the LERF. Use of the CFF is consistent with the risk acceptance guidance in the AESJ Standard. The CFF is also used as a risk metric in the NRA Reactor Oversight Process (ROP).

Experience from full-scope Level 2 PRAs has shown that the types of containment failure modes which are discovered during Type A leak tests typically account for a small fraction of the total LERF. Therefore, changes to the Type A test interval typically have a very small effect on the overall plant LERF. Changes to the Type A test interval have a larger effect on the CFF, due to the predominance of smaller leaks that are discovered during those tests.

Use of the CFF metric in these risk-informed analyses is appropriate, considering the current Japanese standards and regulatory framework. However, the change in CFF significantly over-estimates the change in LERF. Therefore, the risk acceptance criteria for this particular application are conservative, compared to the U.S. practice.

Methodology to Evaluate the Change in Risk

The methodology for quantitative evaluation of the change in risk due to extension of the Type A leak rate test interval is consistent with methods and guidance that are used in the U.S. and endorsed by the NRC. The methodology uses a time-dependent standby failure model to estimate the change in the conditional containment failure probability (ΔCCFP) when the test interval is extended. The overall change in plant risk is evaluated by combining the ΔCCFP with the total core damage frequency (CDF). The time-dependent model provides a conservative estimate for the change in risk, compared to other models and analysis techniques that distinguish among various possible containment failure mechanisms.

The method that is used to estimate the baseline CCFP for the current test interval is consistent with the methodology that is applied in the U.S. The baseline CCFP is derived from the operating experience and data from all Japanese plants, based on their historical Type A test intervals. A Jeffreys non-informative prior model is used to account for the uncertainty in the data. The test data account for all causes for containment leakage, including incipient conditions that develop over time and other failure mechanisms that do not depend directly on the latent period between tests. In practice, only the risk from incipient time-dependent failure mechanisms is affected by extending the Type A test interval. Therefore, this combined treatment of all leakage causes will provide an upper-bound estimate for the change in risk. Unfortunately, currently available methods to examine and analyze the detailed contributors to the leakage data have limited capabilities to distinguish among some specific causes for the observed experience. Therefore, although we know that the recommended methodology provides a conservative estimate for the change in risk, it is difficult to quantify the amount of that conservatism.

Risk Assessment Scope

The Guideline appropriately indicates that the analyses should account for the risk from all internal and external hazards during all plant operating modes when containment integrity is required. Those hazards include internal events (transients, LOCAs, support system failures, etc.), internal hazards (fires, floods, etc.), and external events (earthquakes, tsunamis, high winds, etc.). This scope is consistent with the guidance in Regulatory Guide 1.174, Regulatory Guide 1.163, and the AESJ Standard.

The numerical examples in the Guideline focus primarily on the contributions from internal events, based on the results from readily-available PRAs. However, the guidance clearly indicates that the risk-informed decisions must account for the contributions from all hazards. Quantitative estimates of each hazard's contribution from the current PRA models provide the best information about the plant-specific risk. In some cases, conservatively bounding estimates may be used to estimate the contributions from hazards that are not yet explicitly included in the PRA. In other cases, qualitative evaluations may justify why the risk contribution from specific hazards does not have a significant effect on the decision to extend the test interval. The Guideline discusses these techniques, and it refers to international references that provide more detailed methods and guidance, such as NUREG-1855.

Extension of the Type A test interval affects the containment failure probability during all plant operating modes when containment integrity is required. Those modes include full power operation, low power operation, and several shutdown plant operating states. The Guideline appropriately indicates that the analyses must account for the total change in risk during all applicable operating modes.

Some nuclear power plants require that the containment must maintain a slightly positive pressure to ensure that pumps which are aligned to the containment sump have adequate suction. If the containment leakage is large enough to vent that pressure, the resulting cavitation can damage the pumps and prevent adequate cooling of the reactor core. Therefore, at some plants, extension of the Type A test interval can also affect the estimated CDF, in addition to the CFF. The Guideline specifies that the risk-informed analyses for those plants must account for both contributions to the change in risk. This is consistent with the U.S. guidance and methods.

Evaluation of Cesium-137 Release Frequency

The NRA has reached a consensus on a provisional performance objective that the frequency of a release of more than 100 terabecquerels (TBq) of cesium-137 (Cs-137) should remain below 10^{-6} event per year. Extension of the Type A test interval will affect the conditional containment failure probability (CCFP), the containment failure frequency (CFF), and the associated frequency of Cs-137 releases. The Guideline does not alert analysts to this issue.

The applied risk acceptance criteria for the change in CFF (i.e., ΔCFF) do not directly provide justification that the frequency of Cs-137 releases remains below 10^{-6} event per year. In practice, if the risk analysis results show that the ΔCFF is less than 10^{-7} event per year (i.e., in risk acceptance criteria Region III), it is very likely that the Cs-137 release frequency will remain below 10^{-6} event per year. However, that may not always be the case if the plant-specific baseline release frequency is very close to that limit. If the risk analysis results show that the ΔCFF is between 10^{-7} and 10^{-6} event per year (i.e., in Region II), additional evaluations may be needed to confirm that the Cs-137 release frequency will remain below 10^{-6} event per year.

The Guideline should provide guidance that addresses the provisional performance objective to confirm that the frequency of a release of more than 100 TBq of Cs-137 remains below 10^{-6} event per year.

Guidance for Evaluation of Uncertainties

Regulatory Guide 1.174, the AESJ Standard, and other contemporary international references emphasize the need for a comprehensive evaluation of uncertainties in the analyses that are performed to support an integrated risk-informed decision. Combinations of quantitative and qualitative assessments are typically performed to identify potentially important sources of uncertainty and to evaluate their effects on the risk information that is used by the decision-makers. In practice, the uncertainties improve the decision-makers' understanding of the numerical results and the available margins to the risk acceptance criteria. In some cases, the uncertainties may affect decisions to implement a proposed option. Decision-makers may also apply supplemental compensatory measures for enhanced management of the risk from issues that involve significant uncertainties.

Contemporary references, such as NUREG-1855, typically address three general types of uncertainty: parameter uncertainty, model uncertainty, and completeness uncertainty. The Guideline contains general recommendations for how analysts should quantify the combined effects from parameter uncertainties in the industry Type A leak rate testing data and parameter uncertainties in the results from their plant-specific Level 1 PRA models. Those recommendations are consistent with contemporary practices for the treatment of parameter uncertainties.

Depending on the specific risk-informed application and the maturity of the supporting PRA models, experience has shown that modeling and completeness uncertainties can have a more significant influence on the risk-informed decision than the parameter uncertainties. The Guideline does not contain any guidance, methods, or examples for how utility analysts should identify, evaluate, and document the effects from uncertainties related to the scope, applied methods, and technical quality of the PRA models that are used to support these analyses. The Guideline should be expanded to describe how the assessments of model uncertainty and completeness uncertainty should be performed. If that guidance is not included in this report, the Guideline should explicitly recommend that utility analysts should use the guidance and methods that are described in other internationally accepted references, such as NUREG-1855.

Technical Bases for Specific Numerical Values and Intermediate Results

During our review of the Guideline, we identified a few specific numerical values that need better justification and calculations of intermediate results that should be confirmed. It is not practical to discuss those detailed calculations or the specific numerical values in this summary report. We have alerted the NRRC research team to our specific areas of concern, and we have described the technical issues that should be re-examined.

Risk Management Compensatory Measures

If the assessment results fall within Region II of the applied risk acceptance criteria, the Guideline indicates that supplemental compensatory measures are needed to provide increased confidence that the risk will be managed effectively. The Guideline should provide guidance for the identification and implementation of robust and effective compensatory measures which are directly related to reducing the potential impact of extending the test interval. Examples of a variety of such measures that have been considered internationally include:

- Enhanced monitoring and surveillance to regularly assess the environmental conditions and structural integrity of the containment

- Use of data analytics to track and predict potential leak paths

- Enhanced Type B and Type C testing data collection to allow for the total containment leakage

- Enhanced visual inspections and non-destructive testing methods to assess for wear or deterioration of vulnerable penetrations

We would like to review the final version of the Guideline before it is submitted to the JEA for technical committee approval and endorsement in JEAC4203.

Sincerely,

John W. Stetkar

John W. Stetkar
Chairman

REFERENCES

1. United States Code of Federal Regulations, 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors, Option B – Performance-Based Requirements."

2. Electric Power Research Institute, EPRI 1018243, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," Revision 2-A of EPRI 1009325, October 2008.

3. Nuclear Energy Institute, NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 3-A, July 2012.

4. Japan Electric Association Nuclear Standards Committee, JEAC4203-2017, "Implementation Guidelines for Containment Vessel Leak Rate Test," 2017.

5. United States Nuclear Regulatory Commission, NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking", Revision 1, March 2017.

6. United States Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 3, January 2018.

7. International Atomic Energy Agency, IAEA-TECDOC-1909, "Considerations on Performing Integrated Risk Informed Decision Making," 2020.

8. Atomic Energy Society of Japan, Standard AESJ-SC-S012E:2019, "Implementation Standard Concerning Integrated Risk-Informed Decision Making for the Continuous Safety Improvements in Nuclear Power Plants: 2019," March 2022.

9. United States Nuclear Regulatory Commission, Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program", Revision 1, June 2023.

10. Nuclear Risk Research Center, "Draft Guidelines for Containment Vessel Leak Rate Test," Presentation to NRRC Technical Advisory Committee, May 2023, Proprietary.

11. Technical Advisory Committee of the Nuclear Risk Research Center, "Risk-Informed Changes to Containment Vessel Leak Rate Testing Interval," May 24, 2023.

12. Nuclear Risk Research Center, "Risk-Informed Containment Vessel Leak Rate Test (CVLRT) Program," Presentation to NRRC Technical Advisory Committee, November 2023, Proprietary.

13. Nuclear Risk Research Center, Study Report SE00000, "Feasibility Study on Risk-Informed Reactor Containment Vessels Test Interval Extension in Japan," February 2024, Proprietary.

14. Stetkar, J. W., "Comments and Questions on Feasibility Study on Risk-Informed Reactor Containment Vessels Test Interval Extension in Japan, Draft Study Report SE00000," February 27, 2024, Confidential.

15. Nuclear Risk Research Center, "Draft Guidelines for Risk-Informed Containment Vessel Leak Rate Testing (CVLRT)," Presentation to NRRC Technical Advisory Committee, May 2024, Proprietary.

16. Nuclear Risk Research Center, Study Report NR24002, "Feasibility Study on Risk-Informed Reactor Containment Vessels Test Interval Extension in Japan," August 2024, Proprietary.

17. Nuclear Risk Research Center, "Draft Report for Risk-Informed Containment Vessel Leak Rate Testing (CVLRT)," Presentation to NRRC Technical Advisory Committee, November 2024, Proprietary.