Technical Advisory Committee of the Nuclear Risk Research Center
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SUBJECT: RISK-INFORMED CHANGES TO CONTAINMENT VESSEL LEAK RATE TESTING INTERVAL

Dear Dr. Apostolakis:

During the 18th meeting of the Technical Advisory Committee of the Nuclear Risk Research Center (NRRC), May 15-19, 2023, we met with representatives of the NRRC staff to discuss a proposed methodology to evaluate the risk from changes to the containment vessel integrated leak rate testing intervals at Japanese nuclear power plants.

CONCLUSIONS AND RECOMMENDATIONS

1. The proposed methodology and simplified computation framework are very useful because they can be applied by every Japanese plant now, without the need to develop a detailed Level 2 probabilistic risk assessment (PRA).

2. For this risk-informed application, the change in containment failure frequency (ΔCFF) provides an appropriately conservative bound for the change in large early release frequency (ΔLERF).

3. The change in risk can be appropriately approximated using the core damage frequency (CDF) derived from the full-power PRA model for internal initiating events, when such an estimate is supplemented by considerations of additional contributors such as internal hazards, external events, and events which can occur during low power and shutdown modes when containment integrity must be maintained.

4. The total industry exposure time should not be used to derive estimates for the containment standby failure rate.

5. The proposed methodology should contain guidance for quantification of uncertainties in the conditional containment failure probability (CCFP). Those

uncertainties should be combined with the CDF uncertainties and supplemented, as needed, by additional quantitative or qualitative assessments to evaluate the overall uncertainties in the analysis results.

BACKGROUND

The NRRC is developing methods and guidance that can be used by Japanese utilities to justify extensions to their containment leak rate testing intervals, while maintaining a high level of overall plant safety. Operating nuclear power plants in the United States have used risk-informed, performance-based methodologies to meet the requirements described in Option B of 10 CFR 50, Appendix J to extend their Type A containment integrated leak rate testing intervals to 15 years.

U.S. Nuclear Regulatory Commission Regulatory Guide 1.174 describes the fundamental principles of a risk-informed integrated decision-making process. It also provides guidance for the types of information and supporting analyses which are needed to implement that process. Regulatory Guide 1.163 contains additional guidance that is focused specifically on implementation of a risk-informed, performance-based containment leak rate testing program. The current draft Revision 1 to Regulatory Guide 1.163 endorses the analytical methods that are described in Nuclear Energy Institute report NEI 94-01, Revision 3-A and Electric Power Research Institute report EPRI 1018243.

DISCUSSION

Risk-informed, performance-based changes to a plant's licensing basis should be justified by integrated decisions which implement all five of the fundamental principles that are described in Regulatory Guide 1.174. Our discussions with the NRRC staff about the proposed methodology were focused primarily on the analyses which would be performed to support a conclusion that the change in overall plant risk is acceptably small. We were not briefed on details of the additional guidance and methods that will be used to support the other four elements of the integrated decision process and the plant-specific licensing submittal. Therefore, our conclusions and recommendations in this letter report pertain only to the proposed risk evaluation methodology.

Regulatory Guide 1.174 provides guidelines for acceptable changes to the core damage frequency (ΔCDF) and the large early release frequency (ΔLERF) which may result from a proposed change to a plant's licensing basis. Every Japanese nuclear power plant has a Level 1 probabilistic risk assessment (PRA) that quantifies the CDF. Those PRAs are currently developed to varying levels of completeness and technical quality. Some plants have Level 2 PRAs that can be used to estimate the LERF. However, there is more variability in the scope and fidelity of those PRAs, compared to the Level 1 PRAs.

The proposed methodology is tailored to the current status of the Japanese industry PRAs. In particular, it does not require explicit quantification of the baseline LERF that is associated with the existing containment leak rate testing interval or the LERF change which results from an increase in that interval. It relies rather on an estimate of the change in the total conditional containment failure probability (ΔCCFP) due to all conditions which are identified by the Type A integrated leak rate tests. The change in the containment failure frequency (ΔCFF) is then used as a surrogate metric for ΔLERF. It is quantified by:

$$\Delta CFF = CDF * \Delta CCFP$$

The baseline CCFP and its change are derived from a time-based incipient degradation model which uses containment testing experience data. The methods and guidance in NEI 94-01, Revision 3-A and EPRI 1018243 use a conceptually similar standby failure rate model to estimate the change in LERF. However, in contrast to much of the detailed guidance for the U.S. plant submittals, the NRRC simplified computation framework is very useful because it can be applied by every Japanese plant now, without the need to develop a detailed Level 2 PRA.

The proposed methodology quantifies a conservative bound for the actual ΔLERF. The analysis accounts for the total probability of all containment failure conditions which result in a release of any size that exceeds the Type A testing acceptance criteria. Use of those leakage criteria in these analyses ensures that the actual LERF will, at most, be equal to the CFF. In practice, experience from full-scope Level 2 PRAs and numerous integrated leak rate tests shows that the LERF is typically much smaller than the CFF, because most containment flaws will not result in a large release. Therefore, use of ΔCFF in the proposed computational framework provides an appropriately conservative bound for the ΔLERF that is associated with an increased leak rate testing interval.

Evaluation of Core Damage Frequency

The guidance in Regulatory Guide 1.174 emphasizes that risk-informed decisions should account for how the overall plant risk is affected by the proposed change. This requirement has two implications for the scope of the Level 1 PRA models and other supporting analyses that are used to estimate the CDF.

The first implication is that the analyses should account for the total CDF from all initiating events (i.e., internal events, internal hazards, and external events) which may occur during all reactor operating modes when containment integrity must be maintained (i.e., full power, low power, and several shutdown plant operating states). If the scope of the available plant-specific PRA models does not include all of these contributions, the licensing submittal must describe how the missing elements are considered and evaluated to reach the risk-informed conclusion. In some cases, it may be possible to use appropriately conservative quantitative estimates for some missing contributors. In other cases, combinations of quantitative and qualitative justifications may suffice. In practice, these issues of PRA completeness are often addressed by an integrated decision-making panel of experts who are familiar with the available plant-specific PRA, its results, risk assessments for similar plants, and analytical techniques that are most effective to supplement the lacking information.

A second possible implication regarding the CDF evaluation may apply for specific plants. The licensing basis accident analyses for some U.S. plants include credit for pressure in the containment (so-called "containment overpressure" or "containment accident pressure") to maintain adequate net positive suction head (NPSH) for pumps that can be aligned to the containment sump. For those plants, the analyses should evaluate how the extended testing interval affects the probability of containment leakage conditions which can functionally disable those pumps when they are needed for core recirculation cooling. The risk-informed decision must then account for how the proposed change affects ΔCDF, as well as ΔLERF. This issue is addressed in the guidance in draft Revision 1 of Regulatory Guide 1.163 and EPRI 1018243.

Estimation of Containment Standby Failure Rate

The proposed methodology relies on the use of a standby failure rate model to estimate the ΔCCFP due to the increased testing interval. This type of model is endorsed by the methods and guidance in NEI 94-01, Revision 3-A and EPRI 1018243, and it has been used in the U.S. plant analyses.

The proposed NRRC formulation of the quantification method inappropriately uses the total industry exposure time to derive an estimate for the containment standby failure rate. This process results in a numerical estimate which is smaller than the failure rate which can be justified by the available operating experience. When that failure rate is applied over the extended testing interval, the calculation will underestimate the ΔCCFP.

These analyses use a time-based standby failure rate model for the accumulation of incipient failures and testing restoration. Therefore, it is essential that the supporting data must correctly account for the testing intervals during which the relevant operating experience was compiled. This is a subtle, but very important issue. It is best illustrated conceptually by a simple example.

Suppose that the available operating experience includes 10 units. Each unit has operated for 10 years. During that 10-year period, each unit has performed four Type A leak rate tests. Suppose also that no failures have occurred during any of those tests. Thus, the available industry experience is that no failures have occurred in a total of 40 tests, which were performed with an average interval of 2.5 years between tests. Application of the proposed methodology would incorrectly combine the total industry operating time to infer that no failures have occurred in 100 consecutive years. The standby failure rate that is derived from that 100-year exposure period is smaller than the failure rate which would be derived from the actual 2.5-year testing interval data.

The total industry exposure time should not be used to derive estimates for the containment standby failure rate. Other methods should be used to derive the baseline CCFP and the appropriate containment standby failure rate. The guidance in EPRI 1018243 summarizes a calculation method that has been used for the U.S. plant analyses.

Evaluation of Uncertainties

The guidance in Regulatory Guide 1.174 emphasizes that risk-informed decisions should account for quantitative and qualitative uncertainties in the supporting analyses and results. It specifically refers to the extensive guidance on identification and treatment of uncertainties that is provided in NUREG-1855.

In practice, the available Level 1 PRA models that are used to quantify the plant-specific CDF should quantify the corresponding uncertainty in that parameter.

There are also uncertainties in the CCFP. The proposed methodology should contain guidance for quantification of those uncertainties. The CCFP uncertainties should be combined with the CDF uncertainties to quantify the composite uncertainty in the resulting estimate for ΔCFF.

In practice, it may also be necessary to supplement these explicitly quantified uncertainties by additional quantitative or qualitative assessments of other sources of uncertainty. For example, the integrated decision-making expert panel may evaluate the effects from uncertainties associated with their assessments of the completeness and technical quality of the applied PRA models.

We look forward to continuing our review of this important NRRC project and its benchmark application for practical and effective risk-informed decision-making.

Sincerely,

John W. Stetkar

John W. Stetkar
Chairman

REFERENCES

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2. U.S. 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.

3. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," Proposed Revision 1, Draft Regulatory Guide DG-1391, November 2022.

4. 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.

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