UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

REGULATORY AUDIT REPORT

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

IN SUPPORT OF THE REVIEW OF TECHNICAL REPORT 3002028673,

"LOSS-OF-COOLANT-ACCIDENT-INDUCED FUEL

FRAGMENTATION, RELOCATION AND DISPERSAL

WITH LEAK-BEFORE-BREAK CREDIT," APRIL 2024

ELECTRIC POWER RESEARCH INSTITUTE, INC (EPRI).

DOCKET NO. 99902021

1.0 BACKGROUND

By letter dated April 26, 2024 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML24121A203), EPRI submitted (1) EPRI Report 3002028673, "Loss-of-Coolant-Accident-Induced Fuel Fragmentation, Relocation, and Dispersal with Leak-Before-Break Credit – Alternative Licensing Strategy [ALS]," (2) EPRI Reports 3002028675 (NP)/3002028674 (P), "LOCA [Loss-of-Coolant-Accident] Analysis of Fuel Fragmentation, Relocation, and Dispersal [FFRD] for Westinghouse 2-Loop, 3-Loop and 4-Loop Plants – Proprietary, Evaluation of Cladding Rupture in High Burnup Fuel Rods Susceptible to Fine Fragmentation" (collectively called LOCA Analysis of FFRD), and (3) EPRI Report 3002023895, "Materials Reliability Program: xLPR [Extremely Low Probability of Rupture] Estimation of PWR [Pressurized Water Reactor] Loss-of-Coolant Accident Frequencies (MRP-480)" to the U.S. Nuclear Regulatory Commission (NRC) for review and approval. The primary objective of ALS is to provide a methodology to evaluate the probability of an instantaneous RCS piping rupture to justify not considering FFRD in 10 CFR 50.46 compliance analyses.

By email dated June 25, 2024 (ADAMS Accession No. ML24170A812), the NRC accepted the EPRI topical reports (TRs) for review and approval. This audit report discusses the audit related to EPRI Report 3002028673, "Loss-of-Coolant-Accident-Induced Fuel Fragmentation, Relocation and Dispersal with Leak-Before-Break Credit."

The NRC staff determined that a regulatory audit was needed to increase the efficiency, facilitate discussion, and clarify issues identified during the NRC staff's initial review and conducted a virtual regulatory audit beginning with the entrance meeting on May 2, 2025, until the exit meeting on June 27, 2025, based on the audit plan issued on April 22, 2025 (ADAMS Package Accession No. ML25107A065). The audit was held in accordance with the NRC Office of Nuclear Reactor Regulation procedure as described in LIC-111, "Regulatory Audits," and under the guidance provided in LIC-500, Revision 9, "Topical Report Process." Based on the results of the audit, the NRC issued its request for additional information (RAI) via email dated July 31, 2025 (ADAMS Accession No. ML25212A244).

2.0 REGULATORY AUDIT BASES

Title 10 of the *Code of Federal Regulations* Part 50 (10 CFR 50), "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 4, "Environmental and dynamic effects design bases," specifies, in part, that when analyses approved by the NRC demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping, dynamic effects associated with postulated pipe ruptures may be excluded from the design basis.

The ALS leverages the approved leak-before-break (LBB) analyses for the reactor coolant system main loop piping of pressurized water reactors (PWRs) that were performed to meet 10 CFR Part 50, Appendix A, Criterion 4. LBB is used in ALS to confirm that there is sufficient time between the initiation of a leak and the occurrence of a LOCA. This in turn is used to ensure that a plant with RCS leakage is safely shut down in accordance with the technical specifications. The results of these analyses are used to determine if FFRD must be considered in the 10 CFR 50.46 compliance analyses. Therefore, the regulatory basis for the audit is related to (1) justifying the use of LBB in 10 CFR 50.46 analyses for the reactor coolant system main loop piping of PWRs, and (2) determining whether or not FFRD must be considered in 10 CFR 50.46 compliance analyses.

3.0 AUDIT DISCUSSION

Each item below corresponds to the discussion topics (numbers 1 to 9) described in Section 4.1.2 of the audit plan.

1) The NRC staff discussed whether there is a potential contradiction between the deterministic LBB analyses (ALS TR Section 1.7) and the risk-informed LOCA frequency estimations in NUREG-1829 (ALS TR Section 2.3.2). The NRC staff noted that there is no contradiction because the results of the deterministic LBB analyses involving sufficient conservatisms demonstrate extremely low probabilities of LOCAs, consistent with the LOCA frequencies estimated in NUREG-1829. The NRC staff determined that no RAI was necessary.

2) The NRC staff discussed the applicability of the LBB technology to the ALS that is not directly related to the dynamic effects of a LOCA described in General Design Criterion 4 (GDC 4) in 10 CFR Part 50, Appendix A. The NRC staff noted that the safety and economic benefits of the ALS (e.g., reduced occupational dose due to fewer outages and improved economics due to the use of high burnup fuel) and past precedence provide the justification to evaluate the applicability of the LBB technology to the ALS. The NRC staff found a need to get the information regarding this topic docketed through the RAI process.

3) The NRC staff discussed the potential impact of primary water stress corrosion cracking (PWSCC) on the LBB evaluations that are credited in the ALS. The NRC staff noted that the potential impact of PWSCC on the LBB evaluations is addressed by using the extremely Low Probability of Rupture (xLPR) code jointly developed by EPRI and NRC. The NRC staff also noted that the following reports indicate extremely low probabilities of

LOCAs in the PWR main loop piping due to PWSCC based on probabilistic fracture mechanics analyses: (1) TLR-RES/DE/REB-2021-09, "Probabilistic Leak-Before-Break Evaluation of Westinghouse Four-Loop Pressurized-Water Reactor Primary Coolant Loop Piping using the Extremely Low Probability of Rupture Code," August 13, 2021; and (2) TLR-RES/DE/REB-2021-14-R1, "Probabilistic Leak-Before-Break Evaluations of Pressurized-Water Reactor Piping Systems Using the Extremely Low Probability of Rupture Code," April 2022. The NRC staff found a need to get the information regarding this topic docketed through the RAI process.

4) The NRC staff discussed the operating experience cases where plant personnel misdiagnosed abnormal leakage, which are addressed in ALS TR, Section 4.4.7. The NRC staff noted that in these cases, the location of the leaks was initially misdiagnosed but then identified properly at the plant startup. The NRC staff also noted that none of these cases had the potential to cause a LOCA. The NRC staff found a need to get the information regarding this topic docketed through the RAI process.

5) The NRC staff discussed the bounding nature of the pressurizer surge line and accumulator line in terms of the break size for the reactor coolant pressure boundary (RCPB) piping excluding the main loop piping. However, the NRC staff noted the possibility that plant-specific maximum break sizes may be different from those analyzed in the ALS due to the potential variations in plant-specific pipe sizes. Therefore, the NRC staff will consider specifying a condition regarding the use of the ALS to ensure that the plant-specific maximum break sizes of the RCPB piping excluding the main loop piping are bounded by the maximum break sizes analyzed in the LOCA analysis for the ALS. The LOCA analysis is described in EPRI TR 3002028675, "Loss-of-Coolant-Accident Analysis of Fuel Fragmentation, Relocation, and Dispersal for Westinghouse 2-Loop, 3-Loop, and 4-Loop Plants – Non-Proprietary: Evaluation of Cladding Rupture in High Burnup Fuel Rods Susceptible to Fine Fragmentation." The NRC staff determined that no RAI was necessary.

6) The NRC staff discussed the estimation of the mean LOCA frequency for the 27-inch piping that is described in ALS TR, Section 5.2. The NRC staff noted that the approach for the LOCA frequency estimation is reasonable because the pipe size and LOCA frequency data in NUREG-1829, Volumes 1 and 2, Table 1 are used to estimate the LOCA frequency for the 27-inch piping in a power curve fitting. The NRC staff determined that no RAI was necessary.

7) The NRC staff discussed the evaluation of potential non-piping component rupture. The NRC staff noted that in accordance with 10 CFR 50.46(c)(1), the design basis LOCAs for emergency core cooling system (ECCS) performance evaluation do not include a non-piping LOCA. The NRC staff also noted that the integrity of non-piping components is ensured as part of the original plant design, for which the American Society of Mechanical Engineering (ASME) Code includes design criteria specific to non-piping components. In addition, the potential for degradation of the non-piping components is managed by using aging management programs. The NRC staff found a need to get the information regarding this topic docketed through the RAI process.

8) The NRC staff discussed the effectiveness of the risk-informed inservice inspection (RI-ISI) for similar metal welds and the seismic effects on the potential failure of the main loop piping. EPRI indicated that it planned to discuss these topics at a workshop regarding accident tolerant fuel (ATF) FFRD consequences scheduled for the fall of

2025. At the workshop held on September 18 and 19, 2025, the NRC staff noted that the RI-ISI selects inspection items based on the relevant risk insights (e.g., degradation mechanism, failure probability and failure consequence). The NRC staff also noted that the seismic effects on piping failures are included in the LBB evaluations discussed in the ALS TR and the existing seismic probabilistic risk assessments for the main loop piping. The NRC staff found a need to get the information regarding these topics docketed through the RAI process.

9) The NRC staff discussed a lack of clarity regarding which leak detection mechanisms were required and the maximum possible unidentified leakage with consideration for additional required mass balance checks, technical specifications, trip setpoints, etc. The discussion pointed out that each plant has their own inspection requirements per their plant-specific probabilistic risk assessment (PRA), their own trip set points, and other requirements which make it difficult to generically identify the leak detection mechanisms and maximum possible unidentified leakage. The NRC staff found a need to get the information regarding this topic docketed through the RAI process.

During the audit, the NRC staff reviewed the following documents provided by EPRI in the audit portal or documents available in ADAMS referenced by EPRI during the audit.

List of Documents and Files Reviewed by the NRC Staff
EPRI Report NP-5769, “Degradation and Failure of Bolting in Nuclear Power Plants,” April 1988 (ML003727113).
Pressurized Water Reactor Owners Group, PWROG-17033-NP-A, Revision 1, “Update for Subsequent License Renewal: WCAP-13045, ‘Compliance to ASME Code Case N-481 of the Primary Loop Pump Casings of Westinghouse Type Nuclear Steam Supply Systems,’” November 2019 (ADAMS Accession No. ML19319A188).
Nuclear Energy Institute (NEI) 03-08, “Guideline for the Management of Materials Issues,” Revision 4, October 2020 (ML20315A536).
54 FR 18649, Federal Register, Vol. 54, No. 83, “10 CFR Part 50, Policy Statement on Additional Applications of Leak-Before-Break Technology,” May 2, 1989.
NUREG-1829, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process," Volumes 1 and 2, April 2008 (ML082250436 and ML081060300).
Technical Letter Report (TLR) TLR-RES/DE/REB-2021-09, "Probabilistic Leak-Before-Break Evaluation of Westinghouse Four-Loop Pressurized-Water Reactor Primary Coolant Loop Piping using the Extremely Low Probability of Rupture Code," August 13, 2021 (ML21217A088).
TLR-RES/DE/REB-2021-14-R1, “Probabilistic Leak-Before-Break Evaluations of Pressurized-Water Reactor Piping Systems Using the Extremely Low Probability of Rupture Code,” April 2022 (ML22088A006).

4.0 TEAM AND REVIEW ASSIGNMENTS

The audit team consisted of the following NRC staff:

NAME	ASSIGNMENT	DIVISION	BRANCH
Gordon Curran	Safety and Plant System Engineer	Division of Safety Systems (DSS)	Containment and Plant Systems Branch
James Delosreyes	Project Manager	Division of Operating Reactor Licensing (DORL)	Licensing Projects Branch (LLPB)
David Dijamco	Materials Engineer	Division of New and Renewed Licenses (DNRL)	Vessels and Internals Branch
Lois James	Senior Project Manager	DORL	LLPB
Seung Min	Materials Engineer	DNRL	Piping and Head Penetrations Branch (NPHP)
Eric Palmer	Materials Engineer	DNRL	NPHP
David Rudland	Senior Level Advisor	DNRL	N/A
Chris Van Wert	Senior Level Advisor	DSS	N/A
Brandon Wise	Reactor Systems Engineer	DSS	Nuclear Methods and Fuel Analysis

The EPRI audit team consisted of the following staff:

NAME	AFFILIATION
Sean Martin	EPRI
Kurshad Muftuoglu	EPRI
Fred Smith	EPRI
Valentina Angelici	MPR Associates, Inc.
Cecile Dame	MPR Associates, Inc.
Storm Kauffman	MPR Associates, Inc.
Jeffrey R. Kobelak	Westinghouse Electric Co.

5.0 CONCLUSION

The audit accomplished the objectives and goals listed in the audit plan by allowing direct interaction with EPRI technical experts. The NRC staff were able to obtain clarification on various topics, examine notes supporting EPRI's responses, and discuss differences in technical opinion. The clarifications and examination will allow the NRC staff to assess the need for RAIs and future RAI responses more efficiently.