REQUEST FOR ADDITIONAL INFORMATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REGARDING THE REVIEW OF TECHNICAL REPORT 3002028673,

"LOSS-OF-COOLANT-ACCIDENT-INDUCED FUEL FRAGMENTATION, RELOCATION AND

DISPERSAL WITH LEAK-BEFORE-BREAK CREDIT

ALTERNATIVE LICENSINGSTRATEGY,"

APRIL 2024

ELECTRIC POWER RESEARCH INSTITUTE, INC (EPRI).

DOCKET NO. 99902021

Background

By application dated April 26, 2024 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML24121A203), Electric Power Research Institute, Inc. (EPRI), submitted technical report EPRI Report 3002028673, "Loss-Of-Coolant-Accident-Induced Fuel Fragmentation, Relocation and Dispersal with Leak-Before-Break Credit - Alternative Licensing Strategy," (ADAMS Accession No. ML24121A207) for U.S. Nuclear Regulatory Commission (NRC) review and approval. The primary objective of 3002028673 is to present a regulatory approach (called the Alternative Licensing Strategy (ALS)) for addressing loss-of-coolant-accident (LOCA) induced fuel fragmentation, relocation, and dispersal (FFRD). To complete its review, the NRC staff requests additional information (RAI).

Regulatory Basis

The NRC promulgated regulations to permit power reactor licensees and license applicants to implement an alternative regulatory framework with respect to "special treatment," where special treatment refers to those requirements that provide increased assurance beyond normal industrial practices that structures, systems, and components (SSCs) perform their design-basis functions. Title 10 of *Code of Federal Regulations* Part 50, "Domestic licensing of production and utilization facilities," Section 46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," (referred to as 10 CFR 50.46) provides the regulatory requirements for emergency core cooling system (ECCS) performance for light-water reactors. Additional relevant regulatory criteria are found in Appendix A to 10 CFR, "General Design Criteria [GDC] for Nuclear Power Plants," including Criteria GDC-4, "Environmental and dynamic effects design bases," GDC-14, "Reactor coolant pressure boundary," GDC-30, "Quality of reactor coolant pressure boundary," GDC-31, "Fracture prevention of reactor coolant pressure boundary," GDC-32, "Inspection of reactor coolant pressure boundary," and GDC-35, "Emergency core cooling."

Section 50.46 of 10 CFR and related GDCs requirements ensure that a licensed nuclear power plant has a sufficiently capable reactor coolant makeup system to safely handle breaks in pipes in the reactor coolant pressure boundary up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system.

RAI 1

Issue

Section 1.9 of EPRI's ALS Report, 3002028673, states that the regulatory approach focuses on demonstrating that large break LOCAs have an extremely low likelihood of occurrence; therefore, FFRD can be excluded from core cooling analyses of large break LOCAs. The approach does this by leveraging past leak-before-break (LBB) approvals and recent probabilistic fracture mechanics analyses results for the time between leakage and rupture. NRC policy statement, 54 *Federal Register* (FR) 18649, "Policy Statement on Additional Applications of Leak-Before-break technology," states that the Commission determined not to modify the regulation to support the use of LBB beyond the requirements of GDC-4 and its application to the removal of dynamic effects from the design basis.

Request

Since this topical report utilizes the LBB technology beyond that of GDC-4, please discuss how this work conforms with that policy and/or if the exemption process, per 54 FR 18649 and 10 CFR 50.12, will be used by the licensees attempting to apply this report to a licensing action.

RAI 2

Issue

Section 4.1.2 of EPRI's ALS Report, 3002028673, states that LBB technology cannot be applied to pipe systems with active material degradation. NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," SRP Section 3.6.3, "Leak-Before-Break Evaluation Procedures," states that an evaluation must demonstrate that primary water stress corrosion cracking (PWSCC) is not a potential source of pipe rupture. The LBB approvals (mid 1980s to early 1990s) for most pressurized water reactors (PWRs) in the U.S. fleet did not include this analysis since PWSCC in dissimilar metal welds (Alloy 82/182 material) of the main coolant piping was not discovered until the early 2000s.

Request

Since PWSCC is now considered an active degradation mechanism in some piping systems that have been approved for LBB, please further explain how PWSCC and its mitigation of any kind, including inspections, impact the results in this report.

RAI 3

Issue

Section 4.4.7 of EPRI's ALS Report, 3002028673, states that in some cases plant staff misdiagnosed abnormal leakage trends and returned the reactor to operation only to find the abnormal leakage persisted. This trend implies there are possibilities where human factors may impact LOCA frequencies.

Request

1. Please provide additional information on these incidents, including how many similar events have occurred. Please discuss the steps taken to ensure the issues that caused these leakage cases to be mischaracterized do not repeat. Please indicate how these incidents are qualitatively accounted for in the analysis.

2. Please discuss how the analyses would be impacted if these pipe systems, with low leakage rates, were near rupture.

RAI 4

Issue

Section 6.1 states that “Alternative approaches are employed to assess the vulnerability to a component rupture large enough to potentially cause FFRD.” From the sections that follow (Sections 6.2 - 6.7), it appears that this approach is a qualitative assessment of the vulnerability of these components to failure.

Request

Please discuss the rationale or basis that qualitative arguments for non-piping components are acceptable where quantitative arguments are given for piping.

RAI 5

Issue

1. Throughout the document, credit is taken for both piping and non-piping components for the American Society of Mechanical Engineers (ASME) code inservice inspections that typically occur each inspection interval. However, a recent effort by industry and the ASME code, e.g., Code Case N-935, “Alternative Examination Requirements for PWR Steam Generator Welds and Nozzle Inside Radius Sections, Section XI, Division 1,” is focused on the optimization of inspection that reduces the number of inspections that occur on these major components each inspection interval. When applying probabilistic fracture mechanics (PFM) to a passive component, inspections are typically credited for ensuring the analyses (assumptions and results) remains applicable throughout the component life.

    a. For the piping systems applicable for ALS, the dissimilar metal welds are well maintained through the ASME Code Case N-770 process. However, the similar metal welds are typically within a risk-informed inspection (RI-ISI) program, where only about 10 percent of the welds in the same category are inspected (no known degradation). There is no clear evidence that any of the welds applicable to ALS are inspected.

2. In EPRI Report 3002028673, the analyses used as the basis for determining the probability of failure for both piping and non-piping components incorporated the impacts of seismic loading. Section 3.2 of this report, which references MRP-480, “xLPR Estimation of PWR Loss-of-Coolant-Accident Frequencies (MRP-480),” states seismic loading was investigated through sensitivity cases, and Table 6.5 of EPRI

Report 3002028673 states that seismic loading was considered for non-piping components. However, seismic loading can be very plant specific and sometimes difficult to quantify.

Request

1.

    a. Besides the dissimilar metal (DM) welds, please discuss the percentage of piping welds in the piping systems applicable to ALS that are currently receiving inspections each inspection interval, i.e., about how many of the similar metal welds applicable for ALS are inspected each inspection interval. Please explain how the analyses presented in EPRI's ALS Report, 3002028673 (and the companion MRP-480) remain applicable throughout the component's lifetime with the number of applicable similar metal welds inspected each inspection interval.

    b. For the non-piping systems, please expand on how the ongoing efforts to optimize the inspection (to about 25% of the required inspections) of the steam generator, pressurizer and reactor pressure vessel shell welds impact the conclusions of this effort.

2. Please further explain if the analyses conducted for both piping and non-piping components include bounding seismic loading with respect to those plants within the scope of ALS. If the loading is not bounding, please further justify the use of these analyses for plant specific applications.

RAI 6

Issue

Section 2.3.2 "Level of Conservatism" includes a brief discussion of the Technical Specifications (TS) and other criteria that a licensee needs to be considered suitable for the ALS approach. Additional related information is provided in Section 4.4 "RCS Leak Detection and Response". Based on the TS listed, an unidentified leakage of near 1 gpm could theoretically exist for 72 hours before being identified during a surveillance. This would equate to more than 4000 gallons. Additional leak detection programs are mentioned (in Sections 2.3.2, 4.4, and Appendix A) which would greatly reduce the amount of possible leakage, but it is not clear if they are required by a licensee desiring to adopt ALS.

Request

Please clarify the required leak detection mechanisms and the maximum possible unidentified leakage with consideration for additional required mass balance checks, TS, trip setpoints, etc.