From: Joel Wiebe

Sent: Friday, May 6, 2022 8:58 AM

To: Loomis, Thomas R:

Cc: V Sreenivas; Jason Paige

Subject: Final RAIs 9.1.2021 Constellation Relief Request

REQUEST FOR ADDITIONAL INFORMATION

REQUEST FOR ALTERNATIVE

REGARDING AMERICAN SOCIETY OFMECHANICAL ENGINEERS,

BOILER AND PRESSURE VESSEL CODE, SECTION XI

EXAMINATION REQUIREMENTS FOR STEAM GENERATOR NOZZLE-TO-SHELL WELDS

AND NOZZLE INSIDE RADIUS SECTIONS

DOCKET NOS. 50-456, 50-457, 50-454, 50-455, 50-317, 50-318, AND 50-244

BRAIDWOOD STATION UNITS 1 AND 2

BYRON STATION UNITS 1 AND 2

CALVERT CLIFFS NUCLEAR POWER PLANT UNITS 1 AND 2

R.E. GINNA NUCLEAR POWER PLANT

EXELON GENERATION

By letter dated September 1, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21244A328), Exelon Generation (the licensee) requested U.S. Nuclear Regulatory Commission (NRC) approval of an alternative, RS 21-093, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(1) to the requirements of American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) at Braidwood Station Units 1 and 2 (Braidwood 1 and 2), Bryon Station Units 1 and 2 (Byron 1 and 2), Calvert Cliffs Nuclear Plant Units 1 and 2 (Calvert Cliffs 1 and 2), and R.E. Ginna Nuclear Power Plant (Ginna). The proposed alternative would allow the licensee to forego ASME Code, Section XI-required examinations of steam generator nozzle-to-shell welds and nozzle inside radius sections through the end of the extended licenses.

Specifically, pursuant to Title 10 of the Code of Federal Regulations (10 CFR), Part 50, Paragraph 50.55a(z)(1), the licensee proposed to extend the ISI frequency for the subject components to the end of the current approved Period of Extended Operation, from the current ASME Code Section, Section XI requirement of 10 years. 10 CFR 50.55a(z)(1) requires the licensee to demonstrate that the proposed alternative provides an acceptable level of quality and safety. The licensee referred to the analyses in nonproprietary Electric Power Research Institute (EPRI) Report No. 3002014590, "Technical Bases for Inspection Requirements for PWR Steam Generator Feedwater and Main Steam Nozzle-to-Shell Welds and Inside Radius Sections", April 2019 (ADAMS Accession No. ML19347B107; hereafter EPRI Report) to support the proposed alternative in the submittal. The licensee also included an applicability evaluation of EPRI report 3002014590 to the Braidwood, Byron, Calvert Cliffs, and Ginna units in the submittal. The NRC staff (the staff) needs to issue requests for additional information (RAIs) to complete its review of the licensee's proposed alternative.

RAI 1

Issue

The applicant referenced probabilistic and deterministic analyses (EPRI Report No. 3002014590 noted above) estimating potential fatigue growth in the subject components. The applicant presented information to demonstrate that the referenced analysis would bound the subject components. This information included high-level results from previous inservice inspections (ISI) of the subject components. The applicant provided limited discussion of performance monitoring, primarily focused on justifying application of analyses to components with low inspection coverages (e.g. that leakage would be detected and plants safely shut-down).

The applicant proposed, based on the above, that the ISI interval for the subject components could be extended to the end-of-license, ranging from 12.4 years to 36 years depending on plant unit. For components for which this period would extend substantially beyond 20 years, there appears to be a lack of necessary performance monitoring to approve the request.

Leveraging probabilistic fracture mechanics to define the basis for risk-informing inspection intervals requires knowledge of both the current and future behavior of the material degradation and the associated uncertainties applicable to the subject components during the requested alternative period. Confidence in the results of these analyses hinges on the assurance that the model used adequately represents, and will continue to represent, the degradation behavior in the subject components. Proper performance monitoring through inspections is needed to ensure that the model continues to predict the behavior and that unknown/unpredicted degradation behavior is discovered and dispositioned in a timely fashion.

The licensee discusses the system leakage test as “providing further assurance” for the proposed alternative. However, the NRC staff notes that the visual examinations performed during system leakage tests may not provide sufficient information to ensure that the PFM model continues to predict the material behavior and that emergent degradation is discovered and dispositioned in a timely fashion. Specifically, visual examinations may not directly detect pertinent integrity conditions (e.g., presence or extent of degradation); may not provide direct detection of aging effects prior to potential loss of structure or intended function; and do not provide sufficient validating data necessary to confirm the modeling of degradation behavior in the subject SG welds.

Request

Describe the performance monitoring that will be implemented with this proposed alternative to ensure that the PFM model adequately represents, and will continue to represent, the degradation behavior in the subject components commensurate with the duration of the requested alternative (i.e., plant-specific end date). Justify that this performance monitoring will meet this objective and address the concerns discussed above. Explain how this performance monitoring will provide, over the extended examination interval, (1) direct evidence of the presence and extent of degradation, (2) validation and confirmation of the continued adequacy of the PFM model; and (3) timely detection of novel or unexpected degradation. Describe any actions that will be taken if issues are identified through this performance monitoring to ensure that the integrity of the component is adequately maintained.

RAI 2

Issue

The sensitivity studies and analysis provided in the EPRI report indicate significant potential for risks higher than the base case for Westinghouse steam generator feedwater nozzle cases FEW-P1N and FEW-P3A. As the potential for uncertainties to increase risk for these cases is relatively higher than for the other cases presented in the EPRI report, the relationship between these cases as modeled and the subject plant components is of significant interest in reaching regulatory conclusions.

Request

Compare and contrast the plant specific parameters equivalent to the FEW-P1N and FEW-P3A case components. Indicate where the plant specific parameters suggest plant specific evaluation would result in reduced probabilities relative to the EPRI report base case and/or substantiation of results being materially improved relative to the reported sensitivity results in the EPRI report.

RAI 3

Issue

The probabilistic fracture mechanics (PFM) analysis in the EPRI report assumes a certain number of fatigue cycles for each analyzed transient. The licensee compared corresponding actual fatigue cycles to those assumed in the analysis in Tables A.2, A.3, A.5, A.6, A.8, and A.10 of the submittal. However, Table A.8 does not provide the cycle count for the Loss of Power transient at Calvert Cliffs 1 and 2. The licensee must demonstrate that the assumptions of the generic PFM in the EPRI report are reasonable for Calvert Cliffs 1 and 2.

Request

Provide justification that the transient types and associated cycle numbers analyzed in the EPRI report are reasonable for Calvert Cliffs 1 and 2, including the Loss of Power transient.

RAI 4

Issue

The PFM analysis in the EPRI report assumes certain examination histories, e.g., PSI followed by 10-year inspections. The licensee provided actual examination histories for the subject plants in Appendix B of the submittal. However, there are apparent gaps in the provided examination histories, as described below.

- Appendix B does not provide information on preservice examination history.
- Appendix B does not provide examination history for the Braidwood 2 main steam nozzle inner radius.
- Appendix B does not provide examination history for the Braidwood 1 and 2 inner radius examinations for the 1st ISI interval.
- Appendix B does not provide examination history for the Byron 2 main steam nozzle inner radius.
- Appendix B does not provide examination history for Calvert Cliffs 1 and 2 1st and 2nd ISI interval examinations.
- Appendix B does not provide examination history for Ginna 1st and 2nd ISI interval examinations.

The licensee must demonstrate that the assumptions of the generic PFM in the EPRI report are reasonable for the subject plants.

Request
Describe how the reported examination histories are compliant with Section XI requirements. Describe how the assumed examination histories in the EPRI report's PFM analysis are appropriate for Braidwood 1 and 2, Byron 1 and 2, Calvert Cliffs 1 and 2, and Ginna, given their respective examination histories.

RAI 5
Issue
The licensee reported an indication in the Byron Unit 2 feedwater nozzle. The licensee did not provide sizing information for the flaw. The EPRI report's PFM analysis assumes a certain flaw distribution. The licensee must demonstrate that the assumptions of the generic PFM in the EPRI report are reasonable for Byron Unit 2.

Request
Provide sizing information for the flaw reported in Table B5 of the licensee's submittal. Describe how the assumed flaw distribution in the PFM analysis compares with the actual flaw size.

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