From: Lamb, John
Sent: Thursday, October 22, 2020 1:21 PM
To: Lowery, Ken G.
Subject: RAIs for Vogtle Relief Request - EPRI Report (L-2020-LLR-0109)

Importance: High

Ken,

By letter dated December 11, 2019 (ADAMS Accession No. ML19347B105), as supplemented by letter dated September 9, 2020 (ADAMS Accession Nos. ML20253A311), Southern Nuclear Operating Company (SNC or the licensee) submitted to the U.S. Nuclear Regulatory Commission (NRC) a proposed alternative to the inservice inspection (ISI) requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) for the steam generator (SG) main steam outlet nozzle-to-vessel welds (NVWs) and SG feedwater NVWs and nozzle inside radius (NIR) sections of the Vogtle Electric Generating Plant (Vogtle), Units 1 and 2.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Paragraph 50.55a(z)(1), the licensee proposed to increase the ISI interval for the subject components to 30 years, 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 included in its submittal non-proprietary Electric Power Research Institute (EPRI) Report No. 3002014590, "Technical Bases for Inspection Requirements for PWR [Pressurized Water Reactor] Steam Generator Feedwater and Main Steam Nozzle-to-Shell Welds and Nozzle Inside Radius Sections," April 2019 (ADAMS Accession No. ML19347B107) as the technical basis for the proposed alternative. The licensee also included an applicability evaluation of the EPRI technical basis report to Vogtle, Units 1 and 2. The licensee's proposed alternative relies heavily on the probabilistic fracture mechanics (PFM) analyses in the EPRI report. The NRC staff needs to issue requests for additional information (RAIs) to complete its review of the licensee's proposed alternative. See below for the RAIs.

The licensee's proposed alternative relies on analysis provided in the EPRI Report. The NRC is reviewing the EPRI Report to determine the sufficiency of the analysis to support the licensee's request for authorization of its proposed plant-specific alternative for Vogtle, Units 1 and 2. The NRC is not reviewing the EPRI Report for generic use.

On October 16, 2020, the NRC staff provided draft RAI questions to SNC to make sure that the RAIs are understandable, the regulatory basis is clear, to ensure there is no proprietary information, and to determine if the information was previously docketed. On October 16, 2020, SNC requested a clarifying call with the NRC staff. On October 22, 2020, a clarifying call between the NRC staff and the SNC staff was held. SNC stated that SNC would provide the RAI response within 30 days of the date of this email.

If you have any questions, you can contact me at 301-415-3100.

Sincerely,

Thanks.

John

REQUEST FOR ADDITIONAL INFORMATION

Regulatory Basis

The NRC has established requirements in 10 CFR Part 50 to protect the structural integrity of structures and components in nuclear power plants. Among these requirements are the ISI requirements of Section XI of the ASME Code incorporated by reference in 10 CFR Part 50.55a to ensure that adequate structural integrity of SG components (specifically the SG main steam NVW and SG feedwater nozzle NVW and NIR of Vogtle, Units 1 and 2) are maintained through the service life of the reactor. Therefore, the regulatory basis for the following RAI has to do with demonstrating that the proposed alternative ISI requirements that rely on PFM would ensure adequate structural integrity of the SG main steam NVW and SG feedwater nozzle NVW and NIR of Vogtle, Units 1 and 2, and thereby providing an acceptable level of quality and safety per 10 CFR 50.55a(z)(1) for these components.

RAI 1

Issue

Section 5.2 of the EPRI report states that it did not consider test conditions beyond a system leakage test in the analyses and that, since any pressure tests will be performed at operating pressure, no separate test conditions need to be included in the analyses because the test conditions are captured in the other transients included in the analyses. Even though the test conditions are not included in the analyses, the NRC staff determined that the appropriate temperature conditions for an upper shelf fracture toughness (KIC) value of 200 ksi√in assumed in the EPRI report must exist during the secondary side system leakage and secondary side hydrostatic tests. The NRC staff noted in Sections 3.9.N.1.1.1.15 and 3.9.N.1.1.5.2 of the Vogtle, Units 1 and 2, Updated Final Safety Analysis Report (UFSAR, ADAMS Accession No. ML19296C722) the minimum temperature of 120°F specified for the secondary side system leakage and secondary side hydrostatic tests. The NRC staff noted this minimum temperature is too low for an upper shelf KIC value of 200 ksi√in assumed in the EPRI report. The NRC staff further noted that the value of the parameter "T – RTNDT" used in calculating the ASME Code KIC value must be at least 105°F for the material to be on the upper shelf and a KIC value of 200 ksi√in to be appropriate. Section 8.2.2.7 of the EPRI report states that it assumed an RTNDT value of 60°F for the subjection SG components. Therefore, the temperature "T" in T – RTNDT must be at least 105°F + 60°F = 165°F in order for a KIC value of 200 ksi√in to be appropriate.

Request

Confirm that, when the secondary side system leakage and secondary side hydrostatic tests at Vogtle, Units 1 and 2, are performed at the maximum pressures specified for the tests, the temperature is least 165°F.

RAI 2

Issue

Note 6 in Table 5-5 of the EPRI report states that the Loss of Power transient affects only the feedwater nozzle, and therefore, was applied only to the feedwater nozzle analysis. Table A2 in Enclosure 1 to the September 9, 2020 supplement compares the three cycles of the Loss of Power transient of Vogtle, Units 1 and 2, to the cycles analyzed in the EPRI report. The NRC staff conducted an audit of the PFM software that was used for the PFM analyses in the EPRI report. The NRC staff expects to issue the audit report shortly (ADAMS Accession No. ML20258A002). During the audit, the NRC staff noted that two of the output files for the limiting feedwater nozzle case contain all the transients listed in Table 5-5 of the EPRI report except for the Loss of Power transient (Items 2.e.i and 2.e.ii of the audit report). Table 5-5 indicates that during the Loss of Power transient the pressure is 1,120 psig, and the through-wall stress distribution plots in Figures 7-32 through 7-35 of the EPRI report for the feedwater nozzle clearly show thermal transient stress distributions for the Loss of Power transient at 619 seconds. Pressure and thermal stresses due to the Loss of Power transient could lead to an applied stress intensity factor that exceeds KIC. Also, Table 5-4 of the EPRI report suggests that Loss of Power could have large pressure and temperature fluctuations; large pressure and temperature fluctuations could have a large impact on fatigue crack growth. The description of the Loss of Power transient on page 5-10 of the EPRI report does not provide sufficient details on the transient.

The NRC staff also noted in Items 2.e.i and 2.e.ii of the audit report what appears to be a low temperature overpressure (LTOP) event in two of the input files for the limiting feedwater nozzle case. The NRC staff noted that the EPRI report did not contain information on LTOP; additionally, the NRC staff noted that the design bases in Section 5.2.2.1 of the Vogtle, Units 1 and 2 UFSAR (ADAMS Accession No. ML19296C741) states that the overpressure protection for the steam system is provided by the SG safety valves, which suggests there might be LTOP events that could affect the subject SG components of Vogtle, Units 1 and 2. Therefore, the NRC staff requests the following regarding the Loss of Power transient and LTOP.

Request

a) Explain if the Loss of Power transient was included in the feedwater nozzle stress analyses used as input to the PFM analyses. If it was included, explain why it is not in the output files for the feedwater nozzle case. If not included, explain why in terms of the pressure specified for the Loss of Power transient in Table 5-5 of the EPRI report, the thermal transient stress distributions in Figures 7-32 through 7-35 of the EPRI report, the total of which can lead to a potential exceedance of KIC, and possible pressure/temperature fluctuations during this transient that warrant exclusion from fatigue crack growth calculations.

b) Clarify if LTOP was included in the PFM analyses in the EPRI report and explain whether LTOP is an event that could affect the subject SG components of Vogtle, Units 1 and 2.

RAI 3

Issue

Section 8.2.2.3 of the EPRI report states that the probability of detection (POD) curve used in the analyses is from the PFM analyses performed in proprietary report BWRVIP-108, "BWR [Boiling Water Reactor] Vessel and Internals Project; Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii, October 2002." The NRC staff noted that the NVWs and NIR sections analyzed in BWRVIP-108 were associated with the reactor pressure vessel and that the POD curve was, therefore, developed based on the ultrasonic testing (UT) requirements in ASME Code, Section XI, Appendix VIII. The NVWs and NIR sections in the EPRI report are associated with the SG vessel for which the UT requirements of ASME Code, Section V apply. The NRC staff noted that in practice the POD curve based on the UT requirements of ASME Code, Section V, could be lower than the POD curve based on the UT requirements of ASME Code, Section XI, Appendix VIII. Since ASME Code, Section V is used for the UT examination of SG vessel components, applying the ASME Code, Section XI, Appendix VIII-based POD curve to the PFM analyses of SG vessel components could be nonconservative.

Request

Explain how the ASME Code, Section XI, Appendix VIII-based POD curve is sufficient for the PFM analyses of the subject SG vessel components in the EPRI report.

RAI 4

Issue

Section 8.2.2.4.1 of the EPRI report states that transient stresses are normally distributed. The NRC staff noted that this is consistent with Table 8-3 of the EPRI report, which indicates a normal distribution for transient stresses. However, in the list of inputs for the PFM base cases in Table 8-7, EPRI stated that the uncertainties on transients are "None," which implies that there is no statistical distribution on transient stresses. During the audit of the software used in the PFM analyses in the EPRI report, the NRC staff reviewed one of the input files, which seemed to indicate that the transient stresses have a statistical distribution.

Request

Clarify whether transient stresses (pressure and thermal) were random or not. If random, provide and justify the mean and standard deviation values.

Hearing Identifier: NRR_DRMA
Email Number: 856

Mail Envelope Properties (DM6PR09MB5671345286DAA03E1F3E8E13FA1D0)

Subject: RAIs for Vogtle Relief Request - EPRI Report (L-2020-LLR-0109)
Sent Date: 10/22/2020 1:20:53 PM
Received Date: 10/22/2020 1:20:53 PM
From: Lamb, John

Created By: John.Lamb@nrc.gov

Recipients:
"Lowery, Ken G." <KGLOWERY@southernco.com>
Tracking Status: None

Post Office: DM6PR09MB5671.namprd09.prod.outlook.com

Files	**Size**	**Date & Time**
MESSAGE	11078	10/22/2020 1:20:53 PM

Options
Priority: High
Return Notification: No
Reply Requested: No
Sensitivity: Normal
Expiration Date: