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

September 11, 2024

Fadi Diya
Senior Vice President and
Chief Nuclear Officer
Ameren Missouri
Callaway Energy Center
8315 County Road 459
Steedman, MO 65077

SUBJECT: CALLAWAY PLANT, UNIT NO. 1 – REGULATORY AUDIT QUESTIONS FOR LICENSE RENEWAL COMMITMENTS 34 AND 35 (EPID L-2024-LRO-0009)

Dear Fadi Diya:

By letter dated December 20, 2023 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML23354A244), Union Electric Company, doing business as Ameren Missouri (the licensee), submitted to the U.S. Nuclear Regulatory Commission (NRC) a response to license renewal (LR) Commitment Nos. 34 and 35 for the Callaway Plant, Unit No. 1 (Callaway). In March 2015, the NRC published NUREG-2172, "Safety Evaluation Report Related to the License Renewal of Callaway Plant, Unit 1," which documents the technical review of the Callaway LR application (ML15068A342).

On May 7, 2024 (ML24122A150), NRC staff issued an audit plan, which provided the pertinent details regarding the audit and a list of documents needed on the electronic portal for NRC staff's review of the resolution for LR Commitments 34 and 35. The NRC staff has developed a list of audit questions for further discussions with the licensee representative in the virtual audit phase. The audit will be conducted to increase the staff's understanding of the submittal and identify information that will need to be docketed to support staff's regulatory findings. The NRC project manager will arrange the audit meetings in coordination with your licensing manager upon issuance of this letter.

The audit questions are enclosed. The NRC staff has determined that the audit questions contain proprietary information pursuant to Title 10 of the *Code of Federal Regulations* Section 2.390, "Public inspections, exemptions, request for withholding." Proprietary information, which is provided as enclosure 1, is indicated by **bold** text enclosed with **[[double brackets]]**. Accordingly, the NRC staff has prepared a redacted publicly available non-proprietary version of the audit questions, which is provided as enclosure 2.

Enclosure 1 to this letter contains proprietary information. When separated from Enclosure 1, this document is DECONTROLLED.

If you have any questions, please contact me at 301-415-8371 or via email at Mahesh.Chawla@nrc.gov.

Sincerely,

/RA/

Mahesh L. Chawla, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-483

Enclosures:
1. Proprietary Audit Questions
2. Nonproprietary Audit Questions

cc: Listserv w/o Enclosure 1

ENCLOSURE 2

(NON-PROPRIETARY)

AUDIT QUESTIONS

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT NO. 1.

DOCKET NO. 50-483

Proprietary information pursuant to Title 10 of the *Code of Federal Regulations* Section 2.390 has been redacted from this document.

Redacted information is identified by blank space enclosed within [[double brackets]].

REGULATORY AUDIT QUESTIONS FOR

LICENSE RENEWAL COMMITMENTS 34 AND 35

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. 50-483

CALLAWAY LICENSE RENEWAL COMMITMENTS RELATED TO THE STEAM GENERATOR DIVIDER PLATE AND TUBE-TO-TUBESHEET WELDS

AUDIT QUESTIONS (SET 1)

BACKGROUND

By letter dated December 20, 2023 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML23354A244), Union Electric Company doing business as Ameren Missouri (the licensee), submitted to the U.S. Nuclear Regulatory Commission (NRC) a response to license renewal (LR) Commitment Nos. 34 and 35 for the Callaway Plant, Unit No. 1 (Callaway). In March 2015, the NRC published NUREG-2172 “Safety Evaluation Report Related to the License Renewal of Callaway Plant, Unit 1,” Volume 1, which documents the technical review of the Callaway LR application (ML15068A342).

In accordance with the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) part 54, “Requirements for Renewal of Operating Licenses for Nuclear Power Plants,” and NUREG-2172, appendix A, “License Renewal Commitments,” the NRC staff will perform an audit to gain a better understanding of: (1) how the Callaway steam generator (SG) reactor coolant system (RCS) pressure boundary is adequately maintained in the presence of divider plate weld cracking, and (2) how the susceptibility of the Callaway SG tube-to-tubesheet welds to primary water stress corrosion cracking (PWSCC).

REGULATORY AUDIT BASES, SCOPE AND METHODOLOGY

LR requirements are specified in 10 CFR part 54. In March 2015, the NRC published NUREG-2172, a final safety evaluation report to the LR of Callaway.

In NUREG-2172, appendix A, “Callaway Plant Unit 1 License Renewal Commitments,” Callaway Commitment No. 34 provided the licensee three options to fulfill this commitment. The licensee has chosen Option 2: Analysis, which requires Callaway to:

> Perform an analytical evaluation of the steam generator divider plate welds in order to establish a technical basis which concludes that the steam generator RCS pressure boundary is adequately maintained with the presence of steam generator divider plate weld cracking. The analytical evaluation will be submitted to the NRC for review and approval.

Callaway Commitment No. 35 provided the licensee three options to fulfill this commitment. The licensee has chosen Option 2: Analysis, which requires Callaway to: Perform an analytical evaluation of the SG tube-to-tubesheet welds either determining that the welds are not susceptible to PWSCC or redefining the reactor coolant pressure boundary of the tubes, where the steam generator tube-to-tubesheet welds are not required to perform a reactor coolant pressure boundary function. The redefinition of the reactor coolant pressure boundary will be submitted as part of a license amendment request requiring approval from the NRC. The evaluation for determination that the welds are not susceptible to PWSCC and do not require inspection will be submitted to the NRC for review.

The licensee performed an analytical evaluation specific to the Callaway SGs because it could not be determined if the SGs were bounded by the industry analyses, Electric Power Research Institute (EPRI) Steam Generator Management Program Report 3002002850, “Investigation of Crack Initiation and Propagation in the Steam Generator Channel Head Assembly,” that the NRC staff accepted for the divider plate PWSCC evaluation in LR Interim Staff Guidance (ISG) LR-ISG-2016-01, “Changes to Aging Management Guidance for Various Steam Generator Components” (ML16237A383).

The NRC staff is unable to determine from the information submitted whether the licensee’s analysis establishes an adequate technical basis for (1) concluding that the SG reactor coolant boundary would be maintained in the presence of divider plate cracking, and (2) evaluating the susceptibility of the SG tube-to-tubesheet welds to PWSCC. The audit team will view documentation and calculations the licensee referenced as technical support, focusing on the key assumptions, methodology, and inputs.

<u>AUDIT QUESTIONS</u>

General and Editorial

1. Clarification of applicable American Society of Mechanical Engineers (ASME) Code versions.

Please clarify the ASME Code versions applicable to the analysis for the Callaway SGs. Section 6.2.2.3 of Enclosure 1 of the submittal states that fatigue crack growth is only implemented for ASME Code years 1992 and 1995 with 1996 Addenda in AREVACGC, and it provides a comparison to ASME Code year 2019. However, in document 32-9360111-000,
[[

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2. Visual inspection of tube-to-tubesheet welds.

Please clarify the approach to aging management for the tube-to-tubesheet welds in the Callaway SGs. Page 3 of the submittal cover letter states that the tube-to-tubesheet welds are not susceptible to primary water stress corrosion cracking and do not require inspection based on the chromium content of at least 22 percent. Section 4.0 of Enclosure 2 states that a <u>plant-specific aging management program (AMP)</u> is not needed because the chromium content of the tube-to-tubesheet welds is greater than 22 percent. However, the guidance in LR-ISG-2016-01 states that for SGs with Alloy 690 tubing and Alloy 600 type tubesheet cladding, a plant-specific AMP is unnecessary for a combination of 22 percent chromium in the tube-to-tubesheet welds <u>and</u> compressive stress in the cladding. In the ISG, higher chromium is considered to reduce susceptibility but not eliminate it, and general visual inspection of tube-to-tubesheet welds is a part of the Steam Generators AMP applicable to all pressurized-water reactors.

Materials and Materials Properties

3. Clarification of Component Materials.

Table 5-2 of Enclosure 1 in the submittal describes the component materials as “Typical Material.” **[[**

]] Please explain the meaning of “typical material” in this context and describe any effect on the material property values used in the stress and crack growth analyses.

4. Tubesheet Cladding Material

Please clarify a discrepancy in the identification of the tubesheet cladding materials in Document 51-9268036-000. Section 3.2 (on page 11) **[[**

]] This is consistent with **[[**

]] to the question of whether the tubesheet is clad on the primary side with **[[** **]]**

5. Justification of RT_{NDT} value

Section 3.2.5 of Enclosure 1 in the submittal identifies an RT_{NDT} of 10°F for Callaway based on Reference 9 (“Callaway Unit 1 Replacement Steam Generators – Section 13: Non Ductile Failure Risk”). Please discuss whether this value is justified by testing.

Physical Dimensions

6. Clarification of physical dimensions used in the analyses.

The NRC staff observed apparent discrepancies in some of the physical dimensions identified in the documents submitted or posted in the portal. The table below shows, for example, three different values for channel head wall thickness depending on the source document. Please discuss:

a. The effect of any discrepancies on the conclusions of the stress and crack growth analyses, and

b. Any corrections needed to the documents submitted to resolve the commitments (i.e., package ML23354A244, 12/20/2023).

Dimension	Submittal Enclosure 1 Table 5-1	Document 329360111000 Table 5-1	Document 519268036000 Appendix B
Divider plate t (inches)	2.08	[[ ]]	[[ ]]
Channel Head wall thickness at the triple point (inches)	Not identified. (o.d.-i.d.)/2= 4.66 (with cladding?)	[[ ]]	[[ ]]
Channel head cylinder portion o.d. (inches)	135.65	[[ ]]	[[ ]]
Channel head cylinder portion i.d. (inches)	126.33 (with cladding?)	[[ ]]	[[ ]]
Channel head cladding thickness (inches)	0.2	[[ ]]	[[ ]]
Tubesheet o.d. (inches)	135.65	[[ ]]	[[ ]]
Tubesheet i.d. primary side, base metal (inches)	126.33	[[ ]]	[[ ]]
Tubesheet t (inches)	21.34	[[ ]]	[[ ]]
TS Cladding t (inches)	0.315	[[ ]]	[[ ]]

Loads

7. Clarification of heat transfer coefficient (HTC) units and values.

The “Thermal Analysis” discussion in Section 6.1.2 of Enclosure 1 and **[[ ]]** include HTC values. In both documents, the HTC units are BTU/hr [British Thermal Unit/hour]-s^2-°F [degree Fahrenheit], which appears to be an error because the denominator has two units of time and no unit of surface area. The industry analysis in EPRI report 3002002850, for example, uses HTC units of BTU/hr-ft^2-°F and BTU/sec-in^2-°F. If the units are incorrect, please provide the correct units and values, and describe any effects on the stress and crack growth analyses. The NRC staff notes that Section 6.1.2 of Enclosure 1 states that more detail about the HTCs is in Reference 3 (document 32-9360111-000), however, **[[ ]]**

8. Support lug loads

The last paragraph of Section 5.3 (“Loads”) in Enclosure 1 describes axial force and bending moment loads from the support ring and support lugs. This is discussed in more detail in **[[ ]]** Please clarify how these loads are represented in the crack growth analysis, such as Table 3-1 (“Relevant Sources of Stress for Fatigue Flaw Growth Analysis”). For example, are these loads part of the “Axial Stress due to Pipe Loads (Deadweight, Thermal Expansion)?”

Stress Analysis

9. Clarification of the residual stress used in the flaw growth analysis.

The "Justified Assumptions and Model Simplifications" discussed in Section 4.2 of Enclosure 1 of the submittal and **[[ ]]** state that residual stresses are not explicitly calculated as part of this project, a residual stress value of +2 ksi [kilopound per square inch] is assumed to exist through the thickness of the cladding, and no residual stress is assigned to the low alloy steel material. For stress sources in the flaw growth analysis, Table 3-1 of Enclosure 1 lists "Residual Axial Stress at Shutdown Condition" and "Residual Hoop Stress at Shutdown Condition" for circumferential and axial flaws, respectively. However, the industry analysis in EPRI report 3002002850 assumes a residual stress profile of up to +8 ksi through the channel head thickness based on not knowing the location of the channel head seam welds relative to the triple point. Please address the following about how residual stress was used in the crack growth analysis:

a. Is the residual stress listed in Table 3-1 of Enclosure1 the +2 ksi cladding residual stress described in the model assumptions and simplifications? If not, please clarify the difference.

b. Please clarify the basis for a residual stress value of +2 ksi. Based on Section 4.3.2 of EPRI report 3002002850, the staff understands this residual stress value to be based on the 1993 EPRI Report TR-100251, "White Paper on Reactor Vessel Integrity Requirements for Level A and B Conditions." **[[ ]]**

c. Discuss the basis for not considering residual stress in the channel head base material.

10. Tubesheet cladding compressive stress

Section 6.1.4 of Enclosure 1 indicates that for all nodes on the cladding surface of the tubesheet primary side perforated area the steady state stress is compressive. Section 6.3 of Enclosure 1 describes the criteria in LR-ISG-2016-01 for needing a plant-specific AMP unless the tube-to-tubesheet welds contain approximately 22 percent chromium and the tubesheet cladding is in compression. Section 4.0 of Enclosure 2 states that a plant-specific AMP is not necessary for Callaway because the tube-to-tubesheet welds contain greater than 22 percent chromium. Please clarify whether you are concluding from your analyses of chromium content and compressive stress in the tubesheet cladding that the criteria in LR-ISG-2016-01 are met for the combination of chromium content and compressive stress.

Crack Growth Analysis

11. Selection of paths for crack growth analysis

Section 6.3.1 of Enclosure 1 explains that the stresses used in the crack growth calculations were extracted from the finite element analyses along several predefined paths. Please verify that the paths chosen include the limiting (highest) stress path for use in the crack growth analyses.

12. Verification of the crack growth tool

During our audit of the validation of the AREVACSC code, the direct comparison with the **[[** **]]** was unclear. Please provide, for a fatigue crack growth case, a table comparing **[[** **]]** directly to the AREVACGC calculations and step us through the example during an audit meeting.

13. Clarification of the design life considered in the analysis

Please clarify how the design life of the plant was determined for the stress and crack growth analyses with respect to the renewed license, which is scheduled to expire in 2044. For example, is this assumption based on the SGs being installed in 2005 and a having a design life of **[[** **]]** Section 4.2 of Enclosure 1 states that the cycle counts from the original Design Reports are used in the crack growth analysis and are assumed to cover any proposed period of extended operation. Section 5.3 of Enclosure 1 references the design specification for the Callaway replacement steam generators for the design life of the replacement steam generators. This reference is identified as not retrievable from Framatome Records Management.

SUBJECT: CALLAWAY PLANT, UNIT NO. 1 – REGULATORY AUDIT QUESTIONS FOR LICENSE RENEWAL COMMITMENTS 34 AND 35 (EPID L-2024-LRO-0009) DATED SEPTEMBER 11, 2024

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ADAMS Accession Nos.:
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Regulatory Audit Plan – ML24247A303 (Proprietary)
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NAME	MChawla	PBlechman (Rohrer for)	SBloom
DATE	9/1/2024	9/4/2024	9/4/2024
OFFICE	NRR/DORL/LPL4/BC	NRR/DORL/LPL4/PM	
NAME	JRankin	MChawla	
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