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

October 12, 2012

Mr. Adam C. Heflin
Senior Vice President and Chief
Nuclear Officer
Union Electric Company
P.O. Box 620
Fulton, MO 65251

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE CALLAWAY PLANT UNIT 1 LICENSE RENEWAL APPLICATION, SET 15 (TAC NO. ME7708)

Dear Mr. Heflin:

By letter dated December 15, 2011, Union Electric Company d/b/a Ameren Missouri (the applicant) submitted an application pursuant to Title 10 of the *Code of Federal Regulations* Part 54 (10 CFR Part 54) for renewal of Operating License No. NPF-30 for the Callaway Plant Unit 1 (Callaway). The staff of the U.S. Nuclear Regulatory Commission (NRC or the staff) is reviewing this application in accordance with the guidance in NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants." During its review, the staff has identified areas where additional information is needed to complete the review. The staff's requests for additional information are included in the enclosure. Further requests for additional information may be issued in the future.

Items in the enclosure were discussed with Sarah G. Kovaleski, of your staff, and a mutually agreeable date for the response is within 30 days from the date of this letter. If you have any questions, please contact me by telephone at 301-415-2946 or by e-mail at Samuel.CuadradoDeJesus@nrc.gov.

Sincerely,

Samuel Cuadrado de Jesús, Project Manager
Projects Branch 1
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket No. 50-483

Enclosure:
As stated

cc w/encl: Listserv

CALLAWAY PLANT UNIT 1
LICENSE RENEWAL APPLICATION
REQUEST FOR ADDITIONAL INFORMATION, SET 15

RAI 2.3.4.2-1a

By letter dated July 2, 2012, the applicant stated in its response to RAI 2.3.4.2-1 that the safety-related scoping boundary for the main steam piping concludes at the wall between the auxiliary building and turbine building due to equivalent anchors characterized as "no break zones." Additionally, in its response to RAI 2.1-2 by letter dated August 9, 2012, the applicant stated that the attached main steam piping, which extends from the "no break zone" and inside the turbine building, was not included within the scope of license renewal. The exclusion of this attached piping appears to contradict the scoping methodology in license renewal application (LRA) Section 2.1.2.2, which states:

> Nonsafety-related SSCs that are directly connected to safety-related SSCs were included within the scope of license renewal to ensure structural integrity of the safety-related SSC up to the first seismic anchor or equivalent anchor past the safety/nonsafety interface.

The staff requests that the applicant justify excluding the attached main steam piping, which extends from the "no break zone" area and into the turbine building, from the scope of license renewal.

RAI B2.1.7-4a

Background:

Based on Callaway Action Requests (CARs) contained in Callaway's operating experience review document, RAI B2.1.7-4 requested Callaway to discuss whether the Flow-Accelerated Corrosion (FAC) program manages aging mechanisms other than FAC. The response dated August 21, 2012, stated that the FAC program does not manage aging mechanisms other than FAC and that none of the CARs cited in the RAI identified wall thinning due to mechanisms other than FAC in components within the scope of license renewal. Based on the response, the staff is now of the understanding that none of those CARs were associated with components within the scope of license renewal, since the CARs did identify wall thinning due to mechanisms other than FAC. The staff notes that Callaway's operating experience review section for FAC program contains all of the CARs cited in the RAI.

In addition, although NUREG-1801 "Generic Aging Lessons Learned (GALL) Report," Revision 2, aging management program (AMP) XI.M17 does not specifically incorporate applicants' responses to IE Bulletin 87-01, "Thinning of Pipe Walls in Nuclear Power Plants," the GALL AMP cites the bulletin in the "operating experience" program element and in its reference section. Callaway's response to IE Bulletin 87-01, dated September 10, 1987, states that the scope of the Erosion/Corrosion program includes plain carbon steel piping, which is "inspected for erosion due to flow assisted corrosion (FAC) and/or cavitation," and stainless steel piping, which is "inspected for erosion due to cavitation." The response included the results of several inspections for essential service water (ESW) piping in Table 1, "Erosion/Corrosion Inspection Results." The staff acknowledges that industry terminology has changed since 1987.

Issue:

Although the specific components associated with the cited CARs may not be within the scope of license renewal, Callaway's inclusion of these CARs in the AMP's operating experience review indicates that the program includes the associated mechanisms. Title 10 of the *Code of Federal Regulations* Part 54.37 (10 CFR 54.37), "Additional records and record keeping requirements," § 54.37 (a) states that information documenting compliance with this part shall be retained by licensee in an auditable and retrievable form. STARS procedure PAMCOBP-PI-2, "Aging Management Review," Section 3.2 states that operating experience reviews are performed to demonstrate the effects of aging are being adequately managed by existing programs. For each operating experience document, the procedure asks whether it addresses a license renewal aging effect, and a license renewal AMP. Callaway Plant License Renewal Aging Management CAR Operating Experience Report for AMP XI.M17, "Flow-Accelerated Corrosion (FAC)" B2.1.7, includes CAR 2006-08992, and states that the FAC program manages "loss of material due to erosion" for components in a raw water environment. The report also states that CAR 2010-04190, which documented erosion in a valve and in adjacent piping is addressed by the FAC program. Since the RAI response states that the FAC program only addresses FAC and does not include erosion mechanisms, the information contained in the operating experience review section for this program appears to be incorrect.

In addition, stainless steel components and raw water systems are typically not susceptible to FAC. However, Callaway's response to IE Bulletin 87-01 specifically states that the scope of its Erosion/Corrosion program includes erosion due to cavitation for both carbon steel and stainless steel components. In addition, Callaway's bulletin response contains a table with program inspection results that includes measurements from the ESW system, which is a raw water system. Since IE Bulletin 87-01did not discuss cavitation, only addressed carbon steel piping, and limited the systems to those with oxygen content less than 50 parts per billion, Callaway's inclusion of the additional scope indicates that plant-specific operating experience warranted consideration of these aspects. Consequently, it is unclear to the staff how the current licensing basis (CLB) associated with Callaway's Erosion/Corrosion program correlates with the FAC program and where the activities associated with wall thinning due to mechanisms other than FAC are being managed.

Request.

a) Discuss whether information in the FAC program's operating experience review incorrectly ascribed aging mechanisms to the FAC program, and if appropriate verify that the information will be corrected.

b) Discuss the CLB associated with Callaway's Erosion/Corrosion program as documented in response dated September 10, 1987, to IE Bulletin 87-01, and how it correlates to the FAC program. If the associated activities discussed in the bulletin response are not currently part of the FAC program, provide the details regarding how these activities are accomplished.

RAI B2.1.7-5a

Background:

AMP XI.M17, "acceptance criteria" states that corrective actions should be considered if the minimum allowed wall thickness will be reached before the next scheduled outage. Callaway's implementing procedure for this AMP, EDP-ZZ-01115, "Flow-Accelerated Corrosion of Piping

and Components Predictive Performance Manual," defines the design minimum wall thickness (T_{DMW}) as the calculated minimum wall thickness required as determined from the primary stress equations of the applicable construction code and predicates actions based on that value. RAI B2.1.7-5 requested information regarding the use of certified test material report (CMTR) data to reduce the minimum wall thicknesses for American Society of Mechanical Engineers (ASME) Code Class 2 and Class 3 and American National Standards Institute (ANSI) B31.11 application as given in Design Guide ME013, "Pipewall Thickness."

The RAI response stated that the bases for determining the allowable stress limits are defined in ASME Section III, Appendix III, Article 3000, and would be applicable to situations where acceptance limits must be established for materials that are not listed in the stress tables. Based on this concept, the applicant stated that CMTR data can be applied when the documented material strength is greater than the minimum required strength for that particular standard, and that use of CMTR data does not result in any reduction of conservatism. The response also stated that engineering evaluations performed on components for reduced thickness or unanticipated loads are beyond the scope of ASME Section III, and that such evaluations should be based on engineering judgment.

Issue:

Engineering evaluations that determine operability of components due to degraded conditions, such as reduced thickness or unanticipated loads, should be consistent with the NRC Inspection Manual, Part 9900, "Operability Determinations & Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to Quality or Safety." As noted in the above guidance, a nonconforming condition occurs when a component does not conform to all aspects of its CLB, including applicable codes. The ASME Code, Section III, for Class 2 and Class 3 and ANSI B31.1 Piping Code consistently state that the allowable stress values to be used for the design of piping systems are given in the tables of either, ASME Section II, Part D "Maximum Allowable Stress Values," or ANSI Appendix A, "Allowable Stress Tables." Although the ASME Code may provide guidance for establishing acceptance limits for materials **not listed** in the allowable stress tables, the Code does not address the use of CMTR data for materials that **are listed** in the allowable stress tables. It is not clear to the staff whether the use of CMTR data to increase the allowable stress values is in accordance with Callaway's CLB.

In addition, the staff noted that Design Guide ME013 also stated the allowable stresses may be increased by 10 percent above the ASME Code specified allowable stress. It is unclear to the staff whether this increase in allowable stress values above those used in the original code of construction are in accordance with Callaway's CLB.

Request:

Provide the documentation, either NRC-approved code cases or CLB information that establishes the use of certified test material report data to increase the allowable stresses above the values given for materials listed in the allowable stress tables for the applicable code of construction. Provide similar documentation regarding the use of the 10 percent increase given in Design Guide ME013. Alternatively, provide the limitations on the applicability for the use of this approach for evaluations of degraded conditions during the period of extended operation.

RAI B2.1.5-4a

Background:

By its letter dated August 21, 2012, the applicant responded to RAI B2.1.5-4 which, in part, addressed the inspection method and frequency for the cladding degradation indications in the reactor vessel bottom head region.

The applicant stated that the indications are inspected opportunistically when the core barrel is pulled during a refueling outage (RFO) such as for ASME category B-N-3 examinations, and that the prior RFO 13 and RFO 15 evaluations of the indications determined that there is no growth expected. In its response, the applicant also indicated that the cladding thickness is 0.22 inch, and 0.28 inch defect depth is conservative for the two indications. The applicant further indicated that the degraded cladding area dimensions of the first and second indications are 1.5 inch x 0.625 inch x 0.28 inch (deep) and 0.53 inch x 0.3 inch x 0.10 inch (deep), respectively. The applicant further indicated that it will assume that the low-alloy steel base metal is reduced by 0.14 inch in addition to the cladding thickness.

Issue:

LRA Section A1.5 or B2.1.5 does not identify the inspections of the degradation indications (including the exposed steel vessel portion) in the reactor vessel bottom head region as a program enhancement to NUREG-1801 "Generic Aging Lessons Learned (GALL) Report," Revision 2, AMP XI.M11B.

It is not clear what inspection methods will be used by the applicant's program. The staff also needs justification for why the opportunistic inspections without a specific inspection frequency are sufficient to manage loss of material of the reactor vessel bottom head that has the degradation indications.

In addition, the staff needs clarification for why the size (1.5 inch x 0.625 inch) of the first indication, which is described in the applicant's response, is different from the size stated in LRA Section 4.7.3 (1.5 inch x 0.75 inch). The staff also needs clarification for why the conservative total depth for the indications is 0.28 inch rather than 0.36 inch if the base metal reduction is assumed to be 0.14 inch beyond the cladding (0.22 inch thick).

Request:

a) Justify why the inspections of the degradation indications in the reactor vessel bottom head are not identified as a program enhancement to GALL Report AMP XI.M11B.

b) Identify the inspection methods that will be used to manage loss of material due to boric acid corrosion of the degradation indications. As part of the response, clarify whether ultrasonic examination will be performed to measure the reactor vessel thickness at the indication locations.

c) Justify why the opportunistic inspections without a specific inspection frequency are sufficient to manage loss of material due to boric acid corrosion of the degradation indications in the reactor vessel bottom head.

As part of the response, provide additional information to confirm that the proposed inspection method and frequency are sufficient to manage the aging effect of the reactor vessel bottom head (internal surfaces).

d) Provide clarification for the following items: (1) clarification for why the size (1.5 inch x 0.625 inch) of the first indication, which is described in the applicant's response, is different from the size stated in LRA Section 4.7.3 (1.5 inch x 0.75 inch), and (2) clarification for why the conservative total depth for the indications is 0.28 inch rather than 0.36 inch if the base metal reduction is assumed to be 0.14 inch beyond the cladding (0.22 inch thick).

   If the dimensions of the degradation indications have increased since the initial detections of the indications, describe the changes in the size and depth.

   If available, provide schematic illustrations of the degradation indications, including the top view and side view (that is, depth profile characteristics), in order to provide the baseline information for degradation morphology and to support the applicant's claim that the degradation indications were not initiated by aging effects.

e) Ensure that the LRA, including the FSAR supplement, is consistent with the response.

October 12, 2012

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Sincerely,
/RA/
Samuel Cuadrado de Jesús, Project Manager
Projects Branch 1
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket No. 50-483
Enclosure:
As stated
cc w/encl: Listserv
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ADAMS Accession No.: ML12276A373 *concurrence via e-mail

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NAME	YEdmonds	SCuadrado	DMorey	SCuadrado
DATE	10/5/2012	10/9/2012	10/11/2012	10/12/2012

OFFICIAL RECORD COPY

Letter to A. Heflin from S. Cuadrado DeJesus dated, October 12, 2012

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