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

August 16, 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 7 (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 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 7

RAI 3.1.1.027-1

Background:

License renewal application (LRA) item 3.1.1.027 and Section 3.1.2.2.13 address Union Electric Company d/b/a Ameren Missouri's (the applicant's) aging management review (AMR) results for cracking due to stress corrosion cracking (SCC) and fatigue of the guide tube support pins of the control rod guide tube (CRGT) assemblies. LRA Section 3.1.2.2.13 states that this item is not applicable because this item is only applicable to nickel-alloy guide tube support pins. LRA Section 3.1.2.2.13 also states that Callaway guide tube support pins are made of stainless steel.

LRA Section B2.1.6, "[Pressurized Water Reactors] PWR Vessel Internals," indicates that based on industry operating experience, the applicant replaced the Alloy X-750 guide tube support pins (split pins) with strained hardened (cold worked) 316 stainless steel pins during refueling outage (RFO) 13 (spring 2004) to reduce the susceptibility for SCC in the split pins. The LRA also states that there were no cracked Alloy X-750 pins discovered during the replacement process.

In addition, LRA Table 3.1.2-1 indicates that cracking of the stainless steel CRGT support pins is managed by the Water Chemistry program and PWR Vessel Internals program (Existing Program Components) under LRA item 3.1.1.053. Furthermore, LRA Table 3.1.2-1 indicates that cracking of the stainless steel CRGT support pins is also managed by the American Society of Mechanical Engineers (ASME) Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program under LRA item 3.1.1.032.

In comparison, NUREG-1800, Revision 2, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," (SRP-LR) Section 3.1.2.2.13 states that cracking due to SCC and fatigue could occur in nickel alloy CRGT assemblies, guide tube support pins exposed to reactor coolant, and neutron flux. SRP-LR Section 3.1.2.2.13 also indicates that NUREG-1801, "Generic Aging Lessons Learned Report," (GALL Report) item IV.B2.RP-355 recommends further evaluation of a plant-specific program to ensure this aging effect is adequately managed.

After the issuance of SRP-LR, Revision 2, and GALL Report, Revision 2, the U.S. Nuclear Regulatory Commission (NRC or the staff) issued Revision 1 of the safety evaluation of Electric Power Research Institute (EPRI) MRP-227, Revision 0, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines," (as described in ADAMS Accession No. ML11308A770) on December 16, 2011. Section 3.2.5.3, "Evaluation of the Adequacy of Plant-Specific Existing Programs," of Revision 1 of the safety evaluation states:

> Westinghouse guide tube support pins are made from either 316 stainless steel or Alloy X750. There have been issues with cracking of the original Alloy X750 pins and many licensees have replaced them with type 316 stainless steel

> materials. Applicants/licensees shall evaluate the adequacy of their plant-specific existing program and ensure that the aging degradation is adequately managed during the extended period of operation for both Alloy X750 and type 316 stainless steel guide tube support pins (split pins). Therefore, it is recommended that the evaluation consider the need to replace the Alloy X750 support pins (split pins), if applicable, or inspect the replacement type 316 stainless steel support pins (split pins) to ensure that cracking has been mitigated and that aging degradation is adequately monitored during the extended period of operation.

Revision 1 of the safety evaluation of MRP-227 also states that this issue is Applicant/Licensee Action Item (A/LAI) 3.

Issue:

The LRA does not clearly address how the applicant evaluated the need for inspecting these replacement stainless steel support pins to ensure that aging degradation is adequately monitored during the period of extended operation, consistent with A/LAI 3.

Request:

a) Clarify whether the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program in conjunction with the PWR Vessel Internals program is the existing program that is used to manage cracking of these stainless steel CRGT support pins.

b) Describe how A/LAI 3 of Revision 1 of the staff's safety evaluation regarding MRP-227 was completed.

   As part of the response, provide the technical basis for why the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program is adequate to ensure that aging degradation is adequately monitored during the period of extended operation. Clarify whether the VT-3 examination under Examination Category B-N-3 includes examination of the support pins to manage cracking.

   In addition, confirm whether the applicant's actions in response to A/LAI 3 are consistent with the existing NRC-mandated and vendor/supplier-recommended inspection monitoring bases for the applicant's stainless steel CRGT support pins. Otherwise, provide justification for the inconsistency of the applicant's actions with the inspection monitoring bases.

RAI 3.1.1.050-1

Background:

LRA Section B2, "Aging Management Programs," states that GALL Report AMP XI.M12, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Program (CASS)," is not credited.

LRA Table 3.1-1, item 3.1.1.050 states that portions of the Callaway reactor coolant loops are constructed of CASS and that the straight piping pieces are centrifugally cast and the fittings are statically cast. LRA item 3.1.1.050 also states that since the molybdenum and ferrite values for these fittings and piping pieces are below the industry accepted thermal aging significance threshold, thermal aging of the CASS reactor coolant piping is not a concern.

Callaway Final Safety Analysis Report (FSAR) Table 5.2-2 indicates that the reactor coolant pipe is made of centrifugal-cast SA-351, Grade CF8A and the reactor coolant fittings and branch nozzles are made of SA-351, Grade CF8A (cast stainless steel) and SA-182, (Code Case 1423-2) Grade 316N (non-cast stainless steel). The material information regarding CASS reactor coolant system components in the LRA and FSAR is summarized as follows.

- Reactor coolant pipe: centrifugal-cast low-molybdenum CASS (SA-351, Grade CF8A)
- Reactor coolant fittings: static-cast low-molybdenum CASS (SA-351, Grade CF8A)
- Reactor coolant branch nozzles: low-molybdenum CASS (SA-351, Grade CF8A)

In comparison, GALL Report AMP XI.M12 states that for low-molybdenum content steels (SA-351 Grades CF3, CF3A, CF8, CF8A or other steels with molybdenum not exceeding 0.5 weight percent), only static-cast steels with ferrite greater than 20 percent are potentially susceptible to thermal aging embrittlement. GALL Report AMP XI.M12 also indicates that in the susceptibility screening method, ferrite content is calculated by using the Hull's equivalent factor (described in NUREG/CR-4513, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems," Revision 1) or a staff-approved method for calculating delta ferrite in CASS materials.

Issue:

The staff needs additional information to confirm that the applicant's methodology for screening of CASS for susceptibility to thermal aging embrittlement is consistent with the GALL Report. The LRA does not clearly indicate whether the reactor coolant branch nozzles are made of static-cast material or centrifugal-cast material. In addition, the LRA does not provide the ferrite contents of the CASS materials discussed above. Furthermore, the LRA does not clearly indicate that the applicant's screening method is consistent with NUREG/CR-4513, Revision 1, as referenced in GALL Report AMP XI.M12.

Request:

a) Clarify whether the reactor coolant branch nozzles are made of static-cast CF8A material or centrifugal-cast CF8A material.

b) Provide the bounding-case chemical composition of the reactor coolant fittings and branch nozzles that estimates the highest ferrite content of these CASS components.

c) Provide the calculated ferrite content in order to confirm that the bounding case analysis indicates no susceptibility of these CASS components to thermal aging embrittlement.

As part of the response, clarify whether the applicant's screening method is consistent with the guidance of NUREG/CR-4513, Revision 1, for ferrite content calculations using the Hull's equivalent factor as referenced in the GALL Report.

RAI 3.1.1.063-1

Background:

SRP Table 3.1-1, item 3.1.1.063 addresses steel or stainless steel closure bolting exposed to air with reactor coolant leakage. The GALL Report recommends GALL Report AMP XI.M18, "Bolting Integrity," to manage loss of material due to general (steel only), pitting, and crevice corrosion or wear for this component group. The LRA states that this item is not applicable because the item applies only to boiling water reactor (BWR) plants.

Issue:

The staff lacks sufficient information to evaluate the applicant's claim because although the SRP-LR states that item 3.1.1.063 is applicable to BWRs, the applicant has in-scope steel or stainless steel closure bolting exposed to borated water leakage in LRA Tables 3.1.2-2, 3.2.2-1, 3.2.2-5, 3.3.2-2, 3.3.2-10, 3.3.2-24, and 3.3.2-28. The staff noted that the applicant is managing these items for loss of preload and cracking, but not for loss of material. The staff also noted that the applicant has in-scope steel and stainless steel closure bolting exposed to plant indoor air and atmosphere/weather environments which are being managed for loss of material.

Request:

State the basis for why loss of material is not applicable to in-scope steel or stainless steel closure bolting exposed to air with reactor coolant leakage, or provide an AMP to manage this aging effect.

RAI 3.2.1.063-1

Background:

LRA Table 3.2.1, item 3.2.1.063 and LRA Table 3.3.1, item 3.3.1.120 address stainless steel piping, piping components, and piping elements exposed to air-indoor uncontrolled, air with borated water leakage, concrete, air-dry, or gas, and state that there are no aging effects requiring management and no AMP is proposed. The GALL Report states that there are no aging effects requiring management and no AMP is proposed for these component groups. However, the GALL Report recommends that stainless steel components exposed to treated borated water or condensation be managed for loss of material

LRA Tables 3.2.2-6 and 3.3.2-2 state that stainless steel expansion joints, piping, and tanks exposed internally to borated water leakage have no aging effects requiring management and no AMP is proposed.

Issue:

The GALL Report environment of air with borated water leakage is usually referred to as an external environment. It is unclear to the staff how the components exposed internally to borated water leakage are configured such that exposure to borated water leakage can occur, but the leakage does not accumulate such that the environment becomes borated water or condensation.

Request:

Explain the configuration of the components exposed internally to borated water leakage and how accumulation of the borated water leakage is prevented. If accumulation of borated water leakage can occur, explain why the components have no aging effects requiring management.

RAI 3.5.1.016-1

Background:

SRP-LR Table 3.5-1, item 16, recommends using the XI.S2, "ASME Section XI, Subsection IWL," or XI.S6, "Structures Monitoring" programs to manage increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack for all types of BWR/PWR concrete components of "concrete (accessible areas): basemat, concrete: containment; wall; basemat."

Issue:

LRA Table 3.5-1, item 3.5.1.016, is listed as "not applicable" with a discussion stating that "Callaway is a PWR plant with a concrete containment. This NUREG-1801 line is applicable only for steel containments or BWRs."

It is not clear how the applicant concluded in the LRA that item 3.5.1.016 is applicable for steel containments or BRWs only. GALL item II.A2.CP-72 recommends using the XI.S2, "ASME Section XI, Subsection IWL," or XI.S6, "Structures Monitoring" programs to manage increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack in concrete structures for PWRs, with no statement disclosing PWR containments.

Request:

Provide the technical justification as to why item 3.5.1.016 was considered as "not applicable" in the LRA, and the concrete components of "concrete (accessible areas): basemat, concrete: containment; wall; basemat" does not require aging management at Callaway.

RAI 3.5.1.095-1

Background:

GALL Report AMP XI.S3, "ASME Section XI, Subsection IWF" covers the inspection criteria for ASME Class 1, 2, and 3 component supports for license renewal and recommends visual inspection of a sample of supports.

In LRA Table 3.5.2-12 there are stainless steel ASME Class 1, 2, and 3 supports and mechanical equipment supports exposed to air-indoor uncontrolled or borated water leakage which cites item 3.5.1.095 and state that there are no aging effects requiring management and no AMP is proposed.

Issue:

It is unclear to the staff why no AMP is proposed to manage the ASME Class 1, 2, and 3 supports and mechanical equipment supports, given that they appear to be within the scope of the ASME Section XI, Subsection IWF program.

Request:

Provide justification for why the supports are not being managed using the ASME Section XI, Subsection IWF program; or provide an appropriate program to manage the aging effects.

RAI 4.7.3-1

Background:

LRA Section 4.7.3 states that the evaluation considered a plant life of 40 years, which includes 20 years under the current license plus 20 years for plant life extension. The amount of corrosion was calculated assuming a corrosion rate of 0.001 in/yr for normal operating conditions; an outage corrosion rate of 0.015 in/yr with the average outage duration of less than 8 weeks every 18 months; and a 2-week startup period after each outage with a corrosion rate of 0.010 in/yr. In addition, the LRA also discusses assuming an outage corrosion rate for the entire 40 years, which would yield 0.6 in. of corrosion.

LRA Section 4.7.3 indicates that reactor pressure vessel (RPV) low-alloy steel has been left exposed to the reactor coolant and that the vessel minimum wall thickness evaluation demonstrated that the wall thickness, 5.38 in, minus the maximum degraded area depth, 0.28 in., meets the criterion of NB-3324.2, 4.329 in.

Issue:

LRA Section 4.7.3 did not specify the thickness of the cladding and whether the wall thickness of 5.38 in. includes this cladding thickness. The staff noted that corrosion rates are dependent on the temperature, oxygen content and concentration of the boric acid of the environment and

the applicant has not provided the technical bases regarding the use of the specified corrosion rates basis for its plant-specific conditions.

Since only the maximum degraded area depth of 0.28 in. was discussed in LRA Section 4.7.3 with respect to the criterion of NB-3324.2; the staff could not determine whether the degraded depth was a calculated/measured value at the time the cladding was discovered missing or if it was the maximum degraded area depth at the end of the period of extended operation. It is also not clear whether the applicant's corrosion analysis considered degradation associated with a corrosion rate when meeting the criterion of NB-3324.2. The staff requires this information to verify the adequacy of the disposition of the time-limited aging analysis (TLAA) in accordance with 10 CFR 54.21(c)(1)(i).

Request:

a) Identify the thickness of the cladding and clarify whether the wall thickness of 5.38 in. includes this cladding thickness.

b) Justify any assumptions, on the parameters including but not limited to temperature, oxygen content and concentration of the boric acid, which were used in selecting the corrosion rates discussed in LRA Section 4.7.3. Justify that the corrosion rate of 0.015 in/yr is bounding for the conditions at the plant.

c) Confirm that the maximum degraded area depth of 0.28 in. was a calculated/measured value at the time the cladding was discovered to be missing.

   - If it was not, justify that the corrosion analysis only needed to consider 40 years of operation (i.e., 2004-2044), rather than 60 years of operation (1984-2044). In addition, explain how the maximum degraded area depth of 0.28 in. was determined.
   - If it was, clarify whether the corrosion analysis considered degradation due to a corrosion rate through the period of extended operation. Specifically, explain why LRA Section 4.7.3 appears to indicate that only the "maximum degraded area depth" was used to determine if the wall thickness met the criterion of NB-3324.2.

d) Provide revised LRA Sections 4.7.3 and A3.6.3, as necessary.

RAI 4.7.9-1

Background:

LRA Section 4.7.9 addresses the applicant's TLAA for replacement steam generator tube wear for the period of extended operation. LRA Section B2.1.9, "Steam Generators," indicates that the previous steam generators were replaced in fall 2005 (RFO 14). LRA Section 4.7.9 states that this analysis assumed a cumulative operating service of 45 years. LRA Section 4.7.9 also compared the calculated maximum wear of 0.010 in. to the maximum allowable wear of 40 percent of the tube wall thickness, 0.0156 in.

Issue:

Neither the FSAR nor LRA Section A3.6.9 (the FSAR supplement for the TLAA as amended by letter dated May 3, 2012) include the calculated maximum wear of the steam generator tubes (i.e., 0.010 in. for 45 years), which is the major result of the TLAA.

Request:

Justify why the FSAR or FSAR supplement (LRA Section A3.6.9) does not include the calculated maximum wear of steam generator tubes (i.e., 0.010 in. for 45 years). Alternatively, revise the FSAR supplement to include the calculated maximum wear.

RAI 4.7.9-2

Background:

LRA Section 4.7.9 addresses the applicant's TLAA for replacement steam generator tube wear for the period of extended operation. LRA Section B2.1.9, "Steam Generators," indicates that the previous steam generators were replaced in fall 2005 (RFO 14). LRA Section 4.7.9 states that this analysis assumed a cumulative operating service of 45 years. LRA Section 4.7.9 also compared the calculated maximum wear of 0.010 in. to the maximum allowable wear of 40 percent of the tube wall thickness, 0.0156 in.

In comparison, the applicant's letter dated May 17, 2012, encloses "Callaway Energy Center Steam Generator Tube Inspection Report." Section 4.0 of the inspection report addresses the results of the applicant's in-service inspection of the steam generators, which was conducted during RFO 18 (corresponding to 4.1 effective full power years).

The following summarizes the inspection results and related information for the tube wear observed in the four replacement steam generators as addressed in Table 2 of the steam generator inspection report dated May 17, 2012:

- Callaway's plugging limit for the steam generator tubes was set at 28 percent through-wall, conservatively, for the next three cycles (Cycles 19, 20, and 21).
- All of the in-service tubes were inspected (total 22,143 tubes, except for one tube plugged during manufacturing).
- A total of 258 steam generator tubes indicated anti-vibration bar wear or tube support plate wear.
- Based on the inspection, a total of 29 tubes were plugged (among them, 29 tubes were due to anti-vibration bar wear and one tube was plugged due to tube support plate wear).

Issue:

Some of the actual tube wear rates exceeded the calculated maximum wear rate. Therefore, the staff needs additional information that can justify the validity of the applicant's TLAA.

Request:

a) Provide justification for why the applicant's TLAA is valid even though some of the steam generator tubes indicated wear rates greater than the calculated maximum wear rate (0.010 in. for 45 years).

   As part of the response, provide the number of steam generator tubes that indicated wear rates greater than the calculated maximum wear rate.

b) If necessary, identify any impact of the operating experience on the applicant's TLAA and revise the TLAA accordingly, for adequate aging management.

As part of the response, discuss how the applicant will manage steam generator tube wear in case the actual tube wear indications are greater than the calculated maximum tube wear.

RAI 4.7.9-3

Background:

Recent industry operating experience indicates that steam generator tube-to-tube wear could occur due to tube-to-tube interactions, as addressed in NRC's letter dated March 27, 2012, (ADAMS Accession No. ML12087A323).

LRA Section 4.7.9 addresses the applicant's TLAA for replacement steam generator tube wear. LRA Section 4.7.9 indicates the applicant's TLAA analyzes steam generator tube wear due to the impact/sliding motion of the tubes against their supports.

Issue:

LRA Section 4.7.9 indicates that the applicant's TLAA does not include tube-to-tube wear. The staff needs justification for why the applicant's TLAA does not include tube-to-tube wear.

Request:

Provide the applicant's technical basis for why the applicant's TLAA does not include tube-to-tube wear. As part of the response, confirm whether the plant-specific operating experience supports the applicant's technical basis.

RAI 4.7.9-4

Background:

The applicant's letter dated May 17, 2012, encloses "Callaway Energy Center Steam Generator Tube Inspection Report." Section 4.2, "Secondary Side Inspections" of the inspection report, in part, addresses sludge lancing of the steam generators, which was conducted during RFO 18 (corresponding to 4.1 effective full power years). Section 4.3.2 of the inspection report further indicates that sludge lancing was performed on all four steam generators and a combined total of 11 pounds of sludge was collected.

NRC Information Notice 2007-37, "Buildup of Deposits in Steam Generators," indicates that corrosion product accumulation in the tube support plate (TSP) holes can increase the vibration of the tubes. The staff also notes that the increase in tube vibration has adverse effect on steam generator tube wear.

In addition, the following reference indicates that the clogging of the broached holes of the TSPs may cause perturbation of the steam generator internal hydrodynamics, increase of local flow velocities, and flow-induced fretting (IAEA-TECDOC-1668, "Assessment and Management of Aging of Major Nuclear Power Plant Components Important to Safety: Steam Generators," 2011 Update, International Atomic Energy Agency, November 2011, pages 59, 72 and 73).

In comparison, LRA Section 4.7.9 does not clearly indicate whether the applicant's TLAA for steam generator tube wear considers the potential adverse effect of clogging of TSP holes on tube wear.

Issue:

The staff needs to clarify whether the applicant's TLAA considers the potential adverse effect of clogging of TSP holes or why the TLAA does not need to consider it.

Request:

Clarify whether the applicant's TLAA considers the potential adverse effect of clogging of TSP holes on steam generator tube wear. If the TLAA does not consider the adverse effect of clogging of TSP holes, provide justification for why the TLAA does not consider the adverse effect.

As part of the response, confirm whether the plant-specific operating experience supports the applicant's justification.

RAI 4.7.9-5

Background:

LRA Section 4.7.9 addresses the TLAA for replacement steam generator tube wear for the period of extended operation.

During the audit, the staff noted that Callaway Action Request (CAR) 200500411 describes the failure of a flow meter component due to flow accelerated corrosion (FAC) as addressed in RAI B2.1.7-6 (dated July 18, 2012). The staff also noted that the flow tube separated from its venturi throat, migrated down the pipe, and blocked the minimum recirculation flow line.

EPRI Report, TR-112118, "Nuclear Feedwater Flow Measurement Application Guide," July 1999, indicates that venturi flow meters used to calculate feedwater flow rates in nuclear power plants are susceptible to aging-related degradation that can cause flow rate calculation errors. In addition, EPRI TR-112118 and the following references indicate that plants might enter overpower conditions due to non-conservative flow correction factors with respect to assessment of reactor power level (i.e., due to underestimation of the feedwater flow rate).

- Licensee Event Report 317-2005-003, Revision 1, "Overpower Condition Resulting from Non-conservative Flow Correction Factors," December 14, 2005 (ADAMS Accession No. ML053540215)

- NSAL-03-12, "CROSSFLOW Ultrasonic Flow Measurement System Flow Signal Interference Issues," Westinghouse Electric Company, December 5, 2003 (ADAMS Accession No. ML033421289)

- NRC Regulatory Issue Summary (RIS) 2007-24, "NRC Staff Position on Use of the Westinghouse CROSSFLOW Ultrasonic Flow Meter for Power Uprate or Power Recovery," September 27, 2007 (ADAMS Accession No. ML063450261)

Issue:

Although the failed flow meter addressed in the CAR is not a feedwater venturi flow meter, the plant-specific and industry operating experience indicates that the feedwater venturi flow meter may be subject to similar aging degradation (e.g., FAC or corrosion product deposits). Therefore, the staff requires clarification as to whether the applicant's TLAA adequately considers potential effects of flow rate calculation errors or flow correction factor errors on steam generator tube wear.

Request:

Provide additional information to clarify that the applicant's TLAA adequately considers potential effects of flow rate calculation errors or flow correction factor errors that can accelerate steam generator wear through overpower conditions (i.e., due to underestimation of the feedwater flow rate).

RAI 3.2.1.19-2

Background:

SRP-LR Table 3.2.1, item 19 addresses stainless steel heat exchanger tubes exposed to treated water that are being managed for reduction of heat transfer due to fouling. In its LRA, the applicant states that this item is not applicable because there are no in-scope stainless steel heat exchangers exposed to treated water in the containment spray system.

The staff notes that although the GALL Report only cites an item from the containment spray system for consideration by PWRs, the material, environment, aging effect combination described by this item also applies to other engineered safety feature systems, such as the residual heat removal (RHR) system. The staff also notes that LR-ISG-2011-01, "Aging Management of Stainless Steel Structures and Components in Treated Borated Water," discusses the inappropriate credit previously given to boron as a corrosion inhibitor, and states that aging effects such as reduction of heat transfer may not be adequately managed using the existing guidance. In addition, LR-ISG-2011-01 identifies the additional aging management review items for reduction of heat transfer due to fouling for stainless steel heat exchanger tubes exposed to treated borated water.

Issue:

Although LRA Section 2.1.5.2 states that LR-ISG-2011-01 is applicable to Callaway, it is not clear to the staff why there are no AMR items listed for the stainless steel tubes exposed to treated borated water that are being managed for reduction of heat transfer in LRA Table 3.2.2-6, RHR system for the RHR heat exchangers and the RHR pump seal-water coolers.

Request:

Provide information to justify that SRP-LR Table 3.2.1, item 19 is not applicable, and that no AMR items are needed to manage reduction of heat transfer for stainless steel heat exchanger tubes exposed to treated borated water in any of the engineering safety feature systems. Otherwise, provide additional AMR items, consistent with those listed in LR-ISG-2011-01, to ensure that these components are adequately managed for reduction of heat transfer.

RAI 3.3.1.17-1

Background:

SRP-LR Table 3.3.1, item 17 and item 27 address stainless steel heat exchanger tubes exposed to treated water that are being managed for reduction of heat transfer due to fouling. In its LRA, the applicant states that these items are not applicable because they are only applicable to BWRs.

The staff notes that LR-ISG-2011-01, "Aging Management of Stainless Steel Structures and Components in Treated Borated Water," discusses the inappropriate credit previously given to boron as a corrosion inhibitor, and states that aging effects such as reduction of heat transfer may not be adequately managed using the existing guidance. In addition, LR-ISG-2011-01 identifies the additional AMR items for reduction of heat transfer due to fouling for stainless steel heat exchanger tubes exposed to treated borated water.

Issue:

Although LRA Section 2.1.5.2 states that LR-ISG-2011-01 is applicable to Callaway, it is not clear to the staff why there are no AMR items listed for the stainless steel heat exchanger tubes exposed to treated borated water that are being managed for reduction of heat transfer in LRA Table 3.3.2-2, "Auxiliary Systems – Summary of Aging Management Evaluation – Fuel Pool Cooling and Cleanup System," and Table 3.3.2-10, "Auxiliary Systems – Summary of Aging Management Evaluation – Chemical and Volume Control System."

Request:

Provide information to justify that SRP-LR Table 3.3.1, item 17 is not applicable, and that no AMR items are needed to manage reduction of heat transfer for stainless steel heat exchanger tubes exposed to treated borated water in any of the engineering safety features systems. Otherwise, provide additional AMR items, consistent with those listed in LR-ISG-2011-01, to ensure that these components are adequately managed for reduction of heat transfer.

August 16, 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 7 (TAC NO. ME7708)

Dear Mr. Heflin:

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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,
/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.: ML12216A338 *concurrence via e-mail

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NAME	SCuadrado	YEdmonds	DMorey	SCuadrado
DATE	8/13/12	8/19/12	8/15/12	8/16/12

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Letter to A. Heflin from S. Cuadrado De Jesus dated August 16, 2012

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE CALLAWAY PLANT UNIT 1 LICENSE RENEWAL APPLICATION, SET 7 (TAC NO. ME7708)

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