May 4, 2005

Mr. Charles D. Naslund
Senior Vice President and Chief Nuclear Officer
Union Electric Company
Post Office Box 620
Fulton, MO 65251

SUBJECT: CALLAWAY PLANT, UNIT 1 - REQUEST FOR ADDITIONAL INFORMATION RELATED TO THE STEAM GENERATOR REPLACEMENT LICENSE AMENDMENT REQUEST (TAC NO. MC4437)

Dear Mr. Naslund:

By letter dated September 17, 2004 (ULNRC-05056; available in the Agencywide Documents Access Management System under Accession No. ML042870364), Union Electric Company submitted an application for a license amendment in support of the replacement steam generators to be installed in the fall of 2005 in Refueling Outage 14. The license amendment included changes to the Callaway, Unit 1 Technical Specifications.

Enclosed is a request for additional information (RAI), which is needed by the Nuclear Regulatory Commission (NRC) staff to complete its review of the steam generator replacement in the areas of containment, plant systems, and steam generator tube integrity. The RAI has been discussed with your staff and they have agreed to submit the information in the RAI by June 10, 2005. Submitting the information by this date will allow the NRC staff time to complete its review by the date requested in the application.

Sincerely,

/RA/
Jack Donohew, Senior Project Manager, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-483

Enclosure: Request for Additional Information

cc w/encl: See next page

cc:

Professional Nuclear Consulting, Inc.
19041 Raines Drive
Derwood, MD 20855

John O'Neill, Esq.
Shaw, Pittman, Potts & Trowbridge
2300 N. Street, N.W.
Washington, D.C. 20037

Mr. Mark A. Reidmeyer, Regional
Regulatory Affairs Supervisor
Regulatory Affairs
AmerenUE
P.O. Box 620
Fulton, MO 65251

U.S. Nuclear Regulatory Commission
Resident Inspector Office
8201 NRC Road
Steedman, MO 65077-1302

Mr. Les H. Kanuckel
Manager, Quality Assurance
AmerenUE
P.O. Box 620
Fulton, MO 65251

Missouri Public Service Commission
Governor Office Building
200 Madison Street
Jefferson City, MO 65102-0360

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-4005

Mr. Ronald A. Kucera
Deputy Director for Public Policy
Department of Natural Resources
P.O. Box 176
Jefferson City, Missouri 65102

Mr. Rick A. Muench
President and Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
P.O. Box 411
Burlington, KA 66839

Mr. Dan I. Bolef, President
Kay Drey, Representative
Board of Directors Coalition for the
Environment
6267 Delmar Boulevard
University City, MO 63130

Mr. Lee Fritz, Presiding Commissioner
Callaway County Court House
10 East Fifth Street
Fulton, MO 65151

Mr. David E. Shafer
Superintendent, Licensing
Regulatory Affairs
AmerenUE
P.O. Box 66149, MC 470
St. Louis, MO 63166-6149

Mr. Keith D. Young
Manager, Regulatory Affairs
AmerenUE
P.O. Box 620
Fulton, MO 65251

Mr. Scott Clardy, Director
Section for Environmental Public Health
P.O. Box 570
Jefferson City, MO 65102-0570

Certrec Corporation
4200 South Hulen, Suite 630
Fort Worth, TX 76109

May 4, 2005

Mr. Charles D. Naslund
Senior Vice President and Chief Nuclear Officer
Union Electric Company
Post Office Box 620
Fulton, MO 65251

SUBJECT: CALLAWAY PLANT, UNIT 1 - REQUEST FOR ADDITIONAL INFORMATION RELATED TO THE STEAM GENERATOR REPLACEMENT LICENSE AMENDMENT REQUEST (TAC NO. MC4437)

Dear Mr. Naslund:

By letter dated September 17, 2004 (ULNRC-05056; available in the Agencywide Documents Access Management System under Accession No. ML042870364), Union Electric Company submitted an application for a license amendment in support of the replacement steam generators to be installed in the fall of 2005 in Refueling Outage 14. The license amendment included changes to the Callaway, Unit 1 Technical Specifications.

Enclosed is a request for additional information (RAI), which is needed by the Nuclear Regulatory Commission (NRC) staff to complete its review of the steam generator replacement in the areas of containment, plant systems, and steam generator tube integrity. The RAI has been discussed with your staff and they have agreed to submit the information in the RAI by June 10, 2005. Submitting the information by this date will allow the NRC staff time to complete its review by the date requested in the application.

Sincerely,
/RA/
Jack Donohew, Senior Project Manager, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-483

Enclosure: Request for Additional Information

cc w/encl: See next page

DISTRIBUTION:
PUBLIC
PDIV-2 Reading
RidsNrrDlpmPdiv(HBerkow)
RidsNrrDlpmPdiv2(RGramm)
RidsNrrPMJDonohew
RidsNrrLALFeizollahi
RidsOgcRp
RidsACRSACNWMailCenter
RidsRegion4MailCenter(DGraves)
RLobel
EMarinos
LLois
KDesai
HLi

ACCESSION NO.: ML051150102 **NRR- 088**

OFFICE	PDIV-2/PM	PDIV-2/LA	EEIB/SC	SRXB/SC	PDIV-2/SC
NAME	JDonohew:sp	LFeizollahi	EMarinos	GHsii for LLois	RGramm
DATE	05/02/05	5/2/05	04/28/05	04/28/05	5/4/05

DOCUMENT NAME: E:\Filenet\ML051150102.wpd
OFFICIAL RECORD COPY

REQUEST FOR ADDITIONAL INFORMATION

RELATED TO THE CALLAWAY STEAM GENERATOR REPLACEMENT

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. 50-483

By letter dated September 17, 2004, Union Electric Company (the licensee) requested Nuclear Regulatory Commission (NRC) approval for changes to the Technical Specifications (TSs) for the Callaway Plant, Unit 1 (Callaway) to support the installation of the replacement steam generators (RSGs) in the fall of 2005 in Refueling Outage 14. Based on its review of the licensee's application dated September 17, 2004, in the areas of instrumentation and controls, and reactor systems, the NRC staff requests the following additional information:

Instrumentation and Controls Review:

1. Provide the setpoint calculation documentation for the following protection functions which have allowable values (AV) being revised in this license amendment request:

    (1) Steam Generator Water Level Low-Low (TS Table 3.3.1-1 Functions 14a. and 14b., and TS Table 3.3.2-1 Function 5e.(1), 5e.(2), 6d.(1) and 6d.(2))

    (2) Steam Line Pressure Low (TS Table 3.3.2-1 Functions 1e. and 4e.(1))

    (3) Steam Generator Water Level High-High (TS Table 3.3.2-1 Function 5c.)

2. The TSs define Limiting Safety System Settings (LSSS) as an AV. During reviews of proposed license amendments that contain changes to LSSS setpoints, the NRC staff identified concerns regarding the method used by some licensees to determine the AVs. AVs are identified in the TSs as LSSS to provide acceptance criteria for determination of instrument channel operability during periodic surveillance testing. The NRC staff's concern relates to one of the three methods for determining the AV as described in the Instrument Society of America (ISA) recommended practice ISA-RP67.04-1994, Part II, "Methodologies for Determination of Setpoints for Nuclear Safety-Related Instrumentation."

    The NRC staff has determined that to ensure a plant will operate in accordance with the assumptions upon which the plant safety analyses have been based, additional information is required regardless of the methodology used to establish LSSS values in technical specifications. Details about the NRC staff's concerns are available on the NRC's public website under the Agencywide Documents Access Management System Accession Numbers ML041690604, ML041810346, and ML050670025.

In order for the NRC staff to assess the acceptability of the license amendment request related to this issue, the NRC staff requests the following additional information:

a. Describe the setpoint methodology used to establish AVs associated with LSSS setpoints.

b. In discussing the methodology used, address the following questions regarding the use of the methodology:

   (1) Discuss how the methodology and controls you have in place ensure that the analytical limit (AL) associated with an LSSS will not be exceeded (the AL is a surrogate that ensures the safety limits will not be exceeded). Include in your discussion information on the controls you employ to ensure the trip setpoint established after completing periodic surveillances satisfies your methodology. If the controls are located in a document other than the TS, discuss how those controls satisfy the requirements of 10 CFR 50.36.

   (2) Discuss how the TS surveillances ensure the operability of the instrument channel. This should include a discussion on how the surveillance test results relate to the technical specification AV and describe how these are used to determine the operability of the instrument channel. If the requirements for determining operability of the LSSS instrument being tested are in a document other than the TS (e.g., plant test procedure), discuss how this meets the requirements of 10 CFR 50.36.

c. In discussing the methodology, the following explicit regulatory commitments and proposed TS changes are requested by the NRC staff to complete its review of the methodology:

   (1) To adopt the final Technical Specification Task Force (TSTF) TS changes adopted by NRC for plant TSs to come into conformance with the requirements of 10 CFR 50.36 for LSSS.

   (2) To assess the operability of tested instrumentation based on the previous as-left instrument setting and accounting for the uncertainties associated with the test or calibration.

   (3) To revise the TSs for the LSSS being changed by the license amendment request to incorporate a footnote in the TSs that states: "The as-left instrument setting shall be returned to a setting within the tolerance band of the trip setpoint established to protect the safety limit."

3. In Attachment 1 to the application, Page 7 of 51, Section 3.2, "TTD Elimination," states that upon NRC approval of the amendment request, the 7300 Process Protection System will be modified to eliminate the trip time delay (TTD) circuitry. Provide detailed justification and related protection system logic changes including any markup drawings for the proposed changes. Is there any precedent for these changes?

4. Discuss and identify any safety-related instrumentation change for this steam generator replacement.

5. Describe the locations of the new steam generator level instrument tapes and their impact to the steam generator level setpoint calculation described in item 1 above.

6. Identify any environmental data change (include reference leg data) after the steam generator replacement and discuss the impact on the steam generator level setpoint calculation described in item 1 above.

<u>Reactor Systems Branch Review</u>:

General:

7. State (a) is the RETRAN-2 nodalization used in the Callaway RSG analysis in agreement with the corresponding nodalization in WCAP-14882PA and (b) does the analysis for the RSG using the ANC and the VIPRE codes satisfy the staff limitations on these codes. Explain if there are deviations.

8. Section 4.2.2 discusses the steam generator (SG) blowdown system. The minimum full-load steam pressure could be as low as 867 psia (i.e., 41 psi lower than the current full load pressure of 908 psia). This decrease in blowdown system inlet pressure will impact the required maximum lift of the blowdown flow control valves. Justify that the valves are adequate for the range of the NSSS design parameters approved for the RSG. Provide the document MP 00-1013 which discusses the blowdown control valves.

9. Table 2-1 lists the thermal design parameters for RSG conditions. Provide a tabulation of the expected operating and safety analyses values assumed in the safety analyses to demonstrate that the safety analyses assumptions are adequate or conservative.

10. Discuss the effects of the RSG conditions on the station blackout coping analysis for Callaway.

Low Temperature Overpressure (LTOP) Protection:

11. Section 4.3.5 discusses low-temperature overpressure protection. It is stated that the design-basis Mass Input (MI) transient remains applicable for the RSG program and Heat Input (HI) transient re-analysis was performed. The revised LTOP setpoints were developed using the results of HI transient and the current MI results. Provide the results and assumptions (temperature difference between SG and RCS prior to start a RCP) used in the reanalysis of HI transient.

    Describe the basis for the wide-range pressure uncertainty for LTOP which has increased from 85 to 93 psig for the RSG program and its effects on LTOP design setpoints.

    The MI transient is limiting for RCS temperature less than about 180 degree F. The current LTOP arming temperature of 275 degree F remains applicable for the RSG program. Describe which transient is more limiting between180 degree F and 275 degree F and its effects on LTOP design setpoints.

12. The SG replacement package does not discuss the effect of the SG replacement on overpressure protection analyses (Standard Review Plan 5.2.2, Section II, A). It is apparent that the RSGs would have some effect on overpressure scenarios. Identify the present Callaway overpressure protection basis (is this WCAP-7769) and discuss the impact, if any, of the SG replacement on this basis.

Loss of Coolant Accident (LOCA) Analyses:

13. The SG replacement package does not discuss the effect of the SG replacement on the Callaway LOCA analyses. Page 22 of 51 of Attachment 1 indicates that LOCA analyses are not affected by the new SGs. However, the discussion on pages 19 and 20 of 51 of Attachment 1 indicates that the calculated peak containment pressure for a LOCA event is greater than 1 psi lower with the new SGs than previously analyzed. This would seem to increase the calculated LBLOCA peak cladding temperature. Discuss the effect of the SG replacement on LBLOCA events.

14. The Callaway submittal (WCAP-16265, Section 6.2) provides a summary of LOCA analysis parameters. This submittal also refers to information specific to the large break LOCA (LBLOCA) analyses performed to define the licensing basis for the Callaway LBLOCA. The staff requests the following information to address the programmatic requirements of 10 CFR 50.46(c). To show that the referenced generically approved LOCA analysis methodologies continue to apply specifically to Callaway, provide a statement that Ameren UE and its vendor have ongoing processes which assure that the ranges and values of the input parameters for the Callaway LOCA analysis conservatively bound the ranges and values of the as-operated plant parameters. Furthermore, if the Callaway plant-specific analyses are based on the model and/or analyses of any other plant, then justify that the model or analyses apply to Callaway (e.g., if the other design has a different vessel internals design, the model wouldn't apply to Callaway).

15. The Callaway LOCA submittals did not address slot breaks at the top and side of the cold leg discharge pipe. Justify why these breaks are not considered for the Callaway LOCA submittal.

16. The Callaway submittal states that the Westinghouse LOCBART extension method was used to address downcomer boiling for Callaway. This method has not been approved at this time for a generic application. To show that this method is applicable to Callaway:

    a. Provide comparative results (peak clad temperature (PCT) versus time graph) for the most limiting Callaway LOCA scenario using the LOCBART extension method and WCOBRA/TRAC.

    b. Provide the LBLOCA analysis results tables and graphs to at least 1600 seconds to show that stable and sustained quench is established.

    c. Indicate on the graph the period of downcomer boiling for each calculation.

17. Tables 6.2.1-5 and 6.2.2-5 of WCAP-16265, Section 6.2 (Attachment 1), provide LBLOCA and small break LOCA (SBLOCA) analyses results (PCT, maximum local oxidation, and total hydrogen generation) for the Callaway RSG. For maximum local oxidation, include consideration of both pre-existing and post-LOCA oxidation, and cladding outside and post-rupture inside oxidation. Also, include the limiting local oxidation results for a fuel resident from previous cycles.

18. Provide a table listing the key assumptions and input parameter values for all accident analyses in this license amendment request, before and after the proposed license amendment. Include the core kinetics assumptions used for the steam line break analyses.

19. Use the table of Section 3.1 in Attachment I in the application, which compares the key design parameters of the Framatome Model 73/19T RSGs to the design parameters of the old SGs to discuss how differences in these design parameters would affect analyses and evaluations of the Callaway accidents and transients.

20. Based upon operating history at Callaway and other pressurized water reactors (PWRs), explain the benefits of removing the TTD system, and retaining the environmental allowance monitor (EAM) system, in terms of the predicted effect, if any, upon the frequency of unnecessary reactor trips at Callaway.

<u>Section 6.3.3</u> <u>Steam System Piping Failure (analysis)</u>:

21. Section 6.3.3.1 states, "The effective throat diameter of the flow restrictor nozzles of 6.0315 inches is considerably smaller than the diameter of the main steam pipe. These restrictors are located in the outlet nozzles of the steam generators and serve to limit the maximum steam flow for any break at any location." This is equivalent to a throat area of about 0.2 ft^2, or a much less than half the typical 1.4 ft^2 throat area of current Westinghouse steam generators with integral flow restrictors. Section 6.3.3.2 states, "Since the steam generators are provided with integral flow restrictors with a 1.39 ft^2 throat area, any rupture with a break greater than 1.39 ft^2, regardless of the location, would have the same effect on the nuclear steam supply system (NSSS) as the 1.39 ft^2 break." Please reconcile these statements.

    Note that flow restrictor nozzles with an effective throat diameter of 16.0315 inches (if the 6.0315 inch value is a typo) would have an equivalent throat area of 1.40 ft^2. Please identify the maximum steam line break size that is analyzed for Callaway, and relate it to the steam pipe and flow nozzle dimensions.

22. For Section 6.3.3.2:

    (a) Explain how the coefficient assumption was revised for the RSG analysis to improve the core physics prediction of the point kinetics core model.

    (b) Explain how all reactivity physics parameters are weighted toward the core sector exposed to the greatest cooldown from the faulted loop.

23. Table 6.3.3-1 indicates that, for both hot zero power (HZP) cases, the low steam line pressure setpoint is reached 2 seconds after the steam line rupture. Adding 2 seconds for signal processing and 15 seconds for main steam isolation valve (MSIV) stroke time results in an MSIV closure time of 19 seconds; but the Table lists MSIV closure at 18 seconds. Explain how the 18 seconds value was determined.

24. Provide transient plots depicting steam generator mass and auxiliary feedwater flow, for the faulted and intact steam generators, for both HZP steam line break cases.

<u>Section 6.3.5</u> <u>Loss of Non-Emergency AC Power to the Station Auxiliaries/Loss of Normal Feedwater Flow (analysis)</u>:

25. Section 6.3.5.2 states, "The dual-analysis approach has been previously used by Westinghouse in one other LONF [loss of normal feedwater] analysis of a Westinghouse-designed plant. That previous analysis has been accepted by the NRC and the NRC will again review the results of this dual-analysis approach as part of the RSG license amendment." Reference this LONF analysis and the NRC acceptance documentation.

<u>Section 6.3.10</u> <u>Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power</u>:

26. Discuss, or cite discussions of the effect the SG replacement would have upon the core limits, protection lines, and Overtemperature ΔT trip setpoint calculations used in the Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power event analyses.

27. What is the minimum possible reactivity insertion rate for Callaway? Provide analysis results for this minimum possible reactivity insertion rate, assuming various initial power levels and minimum reactivity feedback, to show that the pressurizer would not fill before the reactor is tripped.

28. Section 6.3.10.4 states, "The [reactor coolant system] RCS pressure safety analysis limit of 2,748.5 psia is confirmed to be met via a generic evaluation." Cite the generic evaluation and verify that the evaluation (1) does not credit operation of spray or power operated control valves (PORVs), and (2) applies to Callaway with the RSGs.

<u>Section 6.3.14</u> <u>Inadvertent Actuation of ECCS at Power (analysis)</u>:

29. How was the current licensing basis for the Inadvertent Actuation of Emergency Core Cooling System (ECCS) at Power event established (i.e., by 10 CFR 50.59 evaluation or by staff review and approval). If by 10 CFR 50.59 evaluation, provide a copy of the evaluation. If by staff review and approval, then please cite the license amendment.

30. The pressurizer PORVs are predicted to open when the pressurizer is water-solid. Are they expected to reseat properly?

    a. If yes, then (1) state how they, and their associated discharge piping, have been qualified for water relief during an Inadvertent Actuation of ECCS at Power event, (2) verify that the automatic control circuitry of these valves meets Class 1E requirements, and (3) indicate the PORV opening setpoint, setpoint tolerance, and surveillance requirements for operation under water relief conditions.

b. If no, then explain how the analysis results demonstrate that the event will not develop into a more serious event, e.g., an SBLOCA.

31. Provide the ECCS flow delivery rate, as a function of RCS pressure, that was assumed for the inadvertent actuation of ECCS at power event analysis. Compare this flow delivery rate to that assumed in the current licensing basis analyses.

Section 6.3.15 Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory (evaluation):

32. Section 15.5.2 of the Callaway Final Safety Analysis Report (FSAR) concludes that the Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory will not fill the pressurizer before the operator can terminate the transient, at about 15 minutes. Provide information to demonstrate that the operator can terminate the transient, as the result of following established procedures, before the calculated pressurizer fill time.

Section 6.3.17 Anticipated Transients Without SCRAM (analysis):

33. The Anticipated Transients Without SCRAM (ATWS) analyses of (1) WCAP-8330, "Westinghouse Anticipated Transients Without Trip Analysis," dated August 1974, and (2) Westinghouse letter NS-TMA-2182, "Anticipated Transients Without Scram for Westinghouse Plants," dated December 1979, are based upon the design parameters of Westinghouse steam generator series 44 and 51, and models D and F, not the Framatome Model 73/19T design. Explain how the referenced Westinghouse ATWS analyses and methods apply to Callaway, as equipped with the RSGs.

34. Section 6.3.0.6 indicates that, "LOFTRAN remains the system transient code for the analyses of . . . the anticipated transients without scram (ATWS, FSAR 15.8)." LOFTRAN has a single-node steam generator shell side model. Describe how LOFTRAN models the primary-to-secondary heat transfer rate as the shell-side water level drops and the steam generator tubes are exposed, for the Framatome Model 73/19T steam generator design.

35. Provide transient plots and sequence of events tables denoting the time and value of peak RCS pressure for analyses of the Callaway Loss of Feedwater and Loss of Load ATWS events, assuming the design characteristics of the Framatome Mode l73/19T RSGs.

36. List and explain the analysis assumptions (e.g., moderator temperature coefficient and initial steam generator mass) used in the ATWS analysis.

37. Verify that the maximum differential pressure, across the tubesheet and tubes of the

Framatome Model 73/19T RSGs, matches or exceeds the value listed in Appendix C of WCAP-8330.

Section 6.4 Steam Generator Tube Rupture (analysis):

38. Show that all assumed operator action times are verified in Callaway simulator exercises.

39. What is the tube rupture size (in sq. ft or sq. in) that is analyzed? Are all tubes in the RSGs of the same diameter?

40. Confirm that there is not an intermediate value of initial average temperature (T_{avg}) that will produce more severe results for the SG tube rupture event.

41. Why are pressurizer heaters not assumed to be operating prior to a reactor trip?

42. Section 6.4.1.2 states that, "Feedwater isolation is completed 4.3 seconds after reactor trip/SI." What is the basis for the 4.3 second value and why is this time not 17 seconds, like all the other applicable events listed in Table 6.3-6? Why is the SG Tube Rupture event not listed in Table 6.3-6?

43. One of the assumptions in Section 6.4.2.2 is, "Additional active failure: The ruptured steam generator's safety valve fails partially open (5-percent effective area) after water relief." If the steam generator's safety valve is not qualified for water relief, then the valve would be assumed to stick open, as a consequential failure, not as an additional active failure. What are the analysis results of this case, assuming a steam generator safety valve that sticks open following water relief?

44. Does the steam generator replacement affect the system dynamics and the timing for boric acid precipitation following large break LOCAs. Explain.

45. In the SBLOCA analyses, while the PCT for the worst break is low, review of the analysis results suggests that a break size between 3 and 4 inches would be more limiting. Since accumulator injection terminates the clad temperature rise for the 4-inch break, a slightly smaller break wherein the RCS pressure just remains of the accumulator actuation pressure could produce a higher PCT. Even though the two-phase level would be higher than that for the 4-inch break, uncovery would persist for a potentially much longer period of time causing the PCT and clad oxidation percentage to increase beyond that for the 4-inch break. Based on these considerations, explain the rationale for assuring the 4-inch break is the limiting break when accumulator injection terminates the clad heat-up.

46. Table 6.2.2-4 identifies loop seal clearing times for each break. Identify the loop seals that cleared and was there any residual water predicted to remain in any of the loop seal piping horizontal sections. Explain.

Containment Integrity Review:

47. In the response to question 2.i of the licensee's submittal dated February 11, 2005, it is stated that "the values for containment volume, heat sink areas, and . . . included conservatisms." Provide examples of these conservatisms.