December 20, 2004

Mr. Garry L. Randolph
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. Randolph:

By letter dated September 17, 2004 (ULNRC-05056), 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 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 February 14, 2005. Submitting the information by this time will the NRC staff 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

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REQUEST FOR ADDITIONAL INFORMATION

CALLAWAY STEAM GENERATOR REPLACEMENT

RELATED TO CONTAINMENT INTEGRITY CALCULATIONS

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. 50-483

By letter dated September 17, 2004, Union Electric Company (the licensee) requested 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 in the areas of containment, plant systems, and steam generator tube integrity, the NRC staff requests the following additional information.

Containment Integrity Review

1. What version of the GOTHIC computer code was used for the containment integrity calculations for Callaway?

2. Discuss and describe the containment model used for the containment integrity calculations for Callaway, including the following (subroutines in the GOTHIC 7.0 manual may be referenced, if appropriate):

    a. Noding (if more than one node, provide noding diagram).
    b. Initial conditions (temperature, pressure, relative humidity).
    c. Assumed containment volume.
    d. Heat sinks (a description, not a detailed list).
    e. Modeling of the containment spray (including drop size assumption) and fan coolers.
    f. Modeling of heat transfer from the water on the containment floor to the containment atmosphere, for the design-basis loss-of-coolant (LOCA) analyses.
    g. Modeling of the break flow including behavior of drops (e.g., settling, coagulation, impact, etc.) in the flow, if modeled.
    h. Heat transfer correlations for heat transfer to heat sinks and drops.
    i. Significant conservative assumptions.

3. Discuss and verify that the calculations were done consistent with the NRC safety evaluation on GOTHIC 7.0, dated September 29, 2003, which was issued on the Kewaunee plant docket.

4. Describe the quality assurance program used for the application of the GOTHIC computer code to Callaway.

5. Provide curves of containment peak pressure and temperature for the design-basis LOCA and main steam line break accident.

Plant Systems Review

6. Item 15 located on page 5 of Attachment 1 to the application, starting with the second sentence, the following is stated: "A revised Loss of Load/Turbine Trip analysis covering operation with inoperable MSSVs [main steam safety valves] was performed by Westinghouse for the RSG project. From the results of that analysis it was determined that operation with 3 OPERABLE MSSVs per steam generator could not be supported above 45% RTP [rated thermal power]."

   In its review of the application, the NRC staff could not find the basis to justify the operation of the reactor at 45 percent RTP with only 3 operable MSSVs per steam generator. Provide the justification for operation at this power level with only three operable MSSVs per steam generator and include the basis for such justification.

7. In the seventh bullet on page 12 of Attachment 1 to the application, it is stated that the full power normal operating T-avg range is from 570.7EF to 588.4EF. In the two paragraphs following this bullet, the following is also stated: "The analysis of the steam dump valve capacity resulted in a restriction on the proposed T-avg range. The installed steam dump valve capacity is adequate at the RSG conditions, provided that the full-load T-avg is no lower than 573EF. The T-avg range of 570.7EF to 588.4EF is a change to the current Callaway analysis basis and required additional analytical work to demonstrate the acceptability of the plant."

   In its review of the application, the NRC staff could not find the basis to justify the operation of Callaway with T-avg below 573EF. Provide justification for operation with T-avg below 573EF and include the basis for such justification.

Steam Generator Tube Integrity Review

The following four questions apply to the TS Bases in Attachment 4 to the application.

8. The first paragraph in TS Bases 3.4.13, "Applicable Safety Analyses," states that the safety analyses for events resulting in a steam discharge to the atmosphere assume that primary to secondary leakage through all steam generators is 1.0 gpm. However, the third paragraph states that the safety analysis for the steam line break assumes 1.0 gpm leakage in the affected steam generator as an initial condition. Is there a discrepancy between these sentences? If so, what revision to these sentences is necessary to resolve this discrepancy? If not a discrepancy, please provide a short explanation of why this is so.

9. In TS Bases 3.4.13, "Surveillance Requirements SR 3.4.13.1," the words "containment atmosphere radioactivity" are to be revised to "containment atmosphere particulate radioactivity." This change is not included as part of the generic TSTF-449, Revision 2 package. What is the justification for adding the word "particulate" without also adding the words "and gaseous"?

10. In TS Bases 3.4.13, "Surveillance Requirements SR 3.4.13.2," the last sentence of the third paragraph states that (for Modes 3 and 4) if steam generator water samples are less than the minimum detectable activity for each principle gamma emitter, primary to secondary leakage may be assumed to be less than 150 gpd through any one steam generator. Is this sentence supported by an analysis and, if not, discuss the justification for the sentence. The staff notes that Draft Revision 3 to the Electric Power Research Institute (EPRI) PWR [pressurized water reactor] Primary to Secondary Leakage Guidelines, Section 3.3.3, states that measurement of primary to secondary leakage during Mode 3 shall be performed using tritium sampling and analysis. Is tritium sampling and analysis employed in Mode 3 at Callaway, and if not, why not?

11. In TS Bases 3.4.17, "Applicable Safety Analyses," the second sentence of the first paragraph states that the steam generator tube rupture analysis assumes a primary to secondary leakage rate of 1.0 gpm to the unaffected steam generators, in excess of the operational leakage rate limits in Limiting Condition for Operation (LCO) 3.4.13. Please clarify whether the 1.0 gpm refers to the total leakage rate for all of the unaffected steam generators or to the leakage rate for each unaffected steam generator. Also, please clarify whether the 1.0 gpm is in addition to leakage up to the LCO limits or whether the 1.0 gpm includes leakage up to the LCO limits. If the latter, this might be made more clear by replacing the words "in excess of" with "exceeding."

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