December 7, 2006

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 - RELIEF REQUEST I3R-04 FOR THE THIRD 10-YEAR INTERVAL INSERVICE INSPECTION (TAC NO. MD1158)

Dear Mr. Naslund:

By letter dated March 28, 2006 (ULNRC-05271), Union Electric Company submitted three Relief Requests (RRs) for its third 10-year inservice inspection (ISI) program interval at the Callaway Plant, Unit 1 (Callaway). The three RRs are I3R-01, I3R-02, and I3R-04. This letter addresses RR I3R-04. The remaining RRs will be addressed in future letters.

Based on the enclosed safety evaluation, the NRC staff concludes that compliance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requirements results in a hardship without a compensating increase in the level of quality and safety, and the licensee's proposed alternative provides reasonable assurance of the structural integrity of the reactor pressure vessel support structures. Based on this determination, pursuant to 10 CFR 50.55a(a)(3)(ii), the Commission authorizes the proposed alternatives in RR I3R-04 for the third 10-year ISI interval at Callaway. All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in this relief request remain applicable, including third-party review by the authorized Nuclear Inservice Inspector.

Sincerely,

/RA/

David Terao, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-483

Enclosure: Safety Evaluation

cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO RELIEF REQUEST I3R-04

FOR THE THIRD 10-YEAR INTERVAL INSERVICE INSPECTION

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. 50-483

1.0 INTRODUCTION

By letter dated March 28, 2006 (Agencywide Documents Access and Management System Accession No. ML061010704), Union Electric Company (the licensee) submitted three Relief Requests (RRs) for its third 10-year inservice inspection (ISI) program interval at the Callaway Plant, Unit 1 (Callaway). The three RRs are I3R-01, I3R-02, and I3R-04. This Safety Evaluation (SE) only addresses RR I3R-04.

2.0 REGULATORY EVALUATION

Inservice inspection of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Class 1, 2, and 3 components is performed in accordance with Section XI of the ASME Code and applicable addenda as required by Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(g), except where specific relief has been granted by the Nuclear Regulatory Commission (NRC) pursuant to 10 CFR 50.55a(g)(6)(i). Paragraph 50.55a(a)(3) of 10 CFR states that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if: (i) the proposed alternatives would provide an acceptable level of quality and safety or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the pre-service examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein.

The code of record for the Callaway third 10-year interval ISI program, which began on December 19, 2005, is the 1998 Edition through the 2000 Addenda of Section XI of the ASME Boiler and Pressure Vessel Code (ASME Code).

3.0 TECHNICAL EVALUATION

ASME Code Components

The code components are the Reactor Pressure Vessel Supports (RPV), Component Numbers 2-RBB01-01, 2-RBB01-02, 2-RBB01-03, and 2-RBB01-04.

Applicable ASME Code Requirement

The ASME Code, Section XI, Table IWF-2500-1, Examination Category F-A, Item Number F1.40 requires 100 percent of Class 1 supports, other than piping supports, be subject to a VT-3 visual examination once every inspection interval.

Licensee's Proposed Alternative Examination

In lieu of implementing the requirements of ASME Code, Section XI, Table IWF-2500-1, Category F-A, Item No. F1.40, the licensee proposes to perform a limited VT-3 visual examination, with the walk plate and insulation installed, on the accessible NF portions of the Reactor Vessel support assemblies. If conditions are discovered during this limited VT-3 examination that do not meet the acceptance standards of IWF-3400, the walk plate or insulation will, if necessary, be removed in order to meet the requirements of IWF-3122.2 or IWF-3122.3, as applicable.

Licensee's Basis for Relief Request (As stated in the licensee's March 28, 2006, submittal)

> Pursuant to 10 CFR 50.55a(g)(5)(iii), relief is requested on the basis that compliance with the specified requirements is impractical. Conformance with the applicable inservice inspection requirements would necessitate a design modification to the reactor pressure vessel supports and associated insulation/walkplate to allow 100% visual examination of the subject supports. In addition, limited accessibility and high radiation levels in the area where these supports are located further reduces the percentage of the supports available for visual examination.
>
> The Callaway reactor vessel is supported by two cold leg nozzles and two hot leg nozzles. There is a support assembly at each of these nozzles that consists of a nozzle weld build-up, shoe plate, air-cooled box, and steel support structure embedded in the primary shield wall. Figures 1 and 2 [Figures 1 and 2 are included in the licensee's letter dated March 28, 2006] depict these support assemblies. As shown in these figures, only the nozzle weld build-up and shoe plate are completely accessible for a visual VT-3 examination. Most of the air-cooled box and the entire steel support structure are located beneath a steel walk plate, and only the top of the air-cooled box is directly accessible. An additional 20 to 30 percent of the air-cooled box and a very small percentage of the steel support structure would be made accessible if the steel walk plate and insulation were removed.

The reactor vessel supports are located in a confined space below the refueling pool permanent seal ring. The area can only be accessed through four seal ring hatches. In addition to difficult access, the radiation level in the area is between 1.5 and 2.0 man-rem per hour.

The large cost[1] of a design modification to the reactor pressure vessel supports and associated insulation/walkplate to allow 100% visual examination of the subject supports is deemed an undue burden. Further, it is estimated that the removal and re-installation of the walk plate and insulation in this confined space, combined with the performance of the visual VT-3 examination, would result in an exposure of approximately 36 man-rem. Removal of the walk plate and insulation, under these conditions, in order to increase the examination coverage of the air-cooled box by approximately 20 to 30 percent and a very small percentage of the steel support structure is considered impractical without a commensurate increase in quality or safety.

NRC Staff Evaluation

The ASME Code of record for Callaway in this inspection interval requires that 100 percent of Class 1 supports, other than piping supports, be subject to a visual, VT-3 examination once every inspection interval. As an alternative to the ASME Code requirements, the license proposes to perform a limited VT-3 visual examination, with the walk plate and insulation installed, on the accessible portions of the RPV support assemblies. In addition, the licensee proposed that if conditions are discovered during this limited VT-3 examination that do not meet the acceptance standards of IWF-3400, the walk plate or insulation will, if necessary, be removed in order to meet the requirements of IWF-3122.2 or IWF-3122.3, as applicable.

As described by the licensee in its March 28, 2006, submittal, limited accessibility and high-radiation levels in the area where the subject supports are located reduces the percentage of the supports available for visual examination. According to Figures 1 and 2 of the licensee's submittal, the RPV is supported by two cold-leg nozzles and two hot-leg nozzles. In addition, there is a support assembly at each of these nozzles that consists of a nozzle weld build-up, shoe plate, air-cooled box, and steel support structure embedded in the primary shield wall. The licensee noted that the nozzle weld build-up and shoe plate are accessible for a visual VT-3 examination. The air-cooled box steel support structure is located beneath a steel walk plate and only the top of the air-cooled box is accessible to perform a VT-3 visual examination. If the steel walk plate and insulation were removed, only an additional 20 to 30 percent of the air-cooled box and a small percentage of the steel support structure would be made accessible for examination.

Furthermore, the subject RPV supports are located in a confined space that is below the refueling pool permanent seal ring. This area is only accessible through four seal-ring hatches and access in this area would cause the licensee's personnel to be exposed to a radiation level between 1.5 and 2.0 man-rem per hour. The licensee estimated that the removal and reinstallation of the walk plate and insulation combined with the performance of the visual VT-3 examination, would result in an exposure of approximately 36 man-rem.

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1. Cost of a modification is not considered a basis for relief either by 10 CFR 50.55a(a)(3)(i), 10 CFR 50.55a(a)(3)(ii), or 10 CFR 50.55a(g)(5)(iii).

In its submittal, requesting relief pursuant to 10 CFR 50.55a(g)(5)(iii), the licensee had determined that the ASME Code requirements are impractical for its facility because a design modification was needed to allow 100 percent visual examination of the subject reactor vessel supports. The licensee showed that the design modification was needed because the radiation levels in the area are high and the access to the supports is difficult.

Based on the radiation exposure and the difficulty in obtaining access to the RPV support area, the NRC staff concludes that compliance with the ASME Code requirements results in a hardship without a compensating increase in the level of quality and safety. The NRC staff further concludes that the alternative provides reasonable assurance of the structural integrity of the RPV support structures, based on the licensee's proposed alternative in RR I3R-04.

4.0 CONCLUSION

For RR I3R-04, the NRC staff concludes that compliance with the ASME Code requirements results in a hardship without a compensating increase in the level of quality and safety, and the licensee's proposed alternative provides reasonable assurance of the structural integrity of the RPV support structures. Therefore, the NRC staff concludes that the licensee's proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the Callaway third 10-year ISI interval. All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in this relief request remain applicable, including third-party review by the authorized Nuclear Inservice Inspector.

Principal Contributor: Thomas K. McLellan

Date: December 7, 2006