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

January 29, 2016

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

SUBJECT: CALLAWAY PLANT UNIT 1 - RELIEF REQUEST I4R-02 FROM THE REQUIREMENTS OF THE ASME CODE, SECTION XI, TABLE IWF-2500-1 FOR 100 PERCENT VISUAL EXAMINATION OF CLASS 1 SUPPORTS (CAC NO MF5613)

Dear Mr. Diya:

By letter dated January 26, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15026A686), as supplemented by letter dated October 6, 2015 (ADAMS Accession No. ML15280A045), Union Electric Company (dba Ameren Missouri), the licensee, submitted a request to the U.S.Nuclear Regulatory Commission (NRC) for relief from American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI requirements at the Callaway Plant, Unit 1 (Callaway).

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(2), the licensee requested relief from the requirements of ASME Code, Section XI, Table IWF-2500-1, Category F-A, Item No. F1.40. This inservice inspection requires the performance of a 100 percent visual VT-3 examination of the reactor pressure vessel Class 1 supports, other than piping supports, once every inspection interval. This relief request is being proposed for use during the fourth inspection interval that began December 19, 2014, and ends on December 18, 2024.

The request has been submitted because compliance with the above examination would result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety. Additionally, the licensee included a proposed alternative to the 100 percent examination above.

The NRC staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation, that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(2). Therefore, the NRC grants relief for the subject examination of the components, as requested in the relief request RR I4R-02 for the fourth 10-year inspection interval at Callaway.

All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in the subject request for relief, remain applicable including third-party review by the Authorized Nuclear Inservice Inspector.

If you have any questions, please contact the Project Manager, John Klos at 301-415-5136 or via e-mail at john.klos@nrc.gov.

Sincerely,

Robert J. Pascarelli, Chief
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-483

Enclosure:
Safety Evaluation

cc w/encl: Distribution via ListServ

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST NO. I4R-02 REGARDING FOURTH 10-YEAR INSERVICE

INSPECTION INTERVAL OF REACTOR PRESSURE VESSEL SUPPORTS

UNION ELECTRIC COMPANY (DBA AMEREN MISSOURI)

CALLAWAY PLANT, UNIT 1

DOCKET NO. 50-483]

1.0 INTRODUCTION

By letter dated January 26, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15026A686), as supplemented by letter dated October 6, 2015 (ADAMS Accession No. ML15280A045), Union Electric Company (Ameren Missouri), the licensee, requested relief from the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI for Callaway Plant, Unit 1 (Callaway), reactor vessel supports.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(2), the licensee requested relief on the basis that compliance with the specific requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

2.0 REGULATORY EVALUATION

The regulations in 10 CFR 50.55a(z)(2) state that alternatives to the requirements of paragraph (g) of 10 CFR 50.55a may be used, when authorized by the U.S. Nuclear Regulatory Commission (NRC), if (1) the proposed alternatives would provide an acceptable level of quality and safety or (2) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Additionally, 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 preservice examination requirements, set forth in the ASME Code, Section XI 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, which was incorporated by reference in 10 CFR 50.55a(a)(1)(ii) 12 months prior to the start of the 120-month interval, subject to the conditions listed in 10 CFR 50.55a(b)(2).

3.0 TECHNICAL EVALUATION

The information provided by the licensee in support of the request for relief from ASME Code requirements has been evaluated, and the bases for disposition are documented below.

3.1 The Licensee's Relief Request Request for Relief I4R-02, from ASME Code, Section XI, Table IWF-2500-1, Examination Category F-A, Item F1.40

ASME Code Components Affected

Reactor vessel supports, Component Numbers 2-RBB01-01, 2-RBB01-02, 2-RBB01-03 and 2-RBB01-04, ASME Code Class 1 component supports.

ASME Code Requirement

ASME Code, Section XI, Table IWF-2500-1, Examination Category F-A, Item F1.40 requires a visual examination (VT-3) of 100 percent of Class 1 supports, other than piping supports.

Licensee's Basis for Relief Request

In its letter dated January 26, 2015, the licensee stated that:

> Pursuant to [10 CFR 50.55a(z)(2)], relief is requested on the basis that compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality or safety. Conformance with the applicable inservice inspection requirements would necessitate a design modification to the Reactor Pressure Vessel (RPV) supports and associated insulation/walk-plate 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 RPV 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 depict these support assemblies. As shown in the 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 RPV supports are located in a confined space below the refueling pool permanent seal ring. The area can only be accessed through seal ring hatches which are partially obstructed by the reactor vessel insulation neutron detector wells. In addition to difficult access, the radiation level in the area is between 1.5 and 2.0 man-rem per hour.
>
> Changing the RPV supports and associated insulation/walk-plate to allow 100% visual examination of the subject supports would be an extensive and costly modification. 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 a hardship without a compensating increase in the level of quality or safety.

Licensee's Proposed Alternative Examination

In its letter dated January 26, 2015, the licensee stated that:

> In lieu of implementing the requirements of Table IWF-2500-1, Category F-A, Item No. F1.40, Callaway proposes to perform a limited VT-3 visual examination, with the walk-plate and insulation installed, on the accessible NF portions of the RPV 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 to in order to meet the requirements of IWF-3122.2 or IWF-3122.3, as applicable.

3.2 NRC Staff Evaluation

The licensee has requested relief from ASME Code requirements pursuant to 10 CFR 50.55a(z)(2). The ASME Code of record for the fourth 10-year interval inservice inspection program, which began on December 19, 2014, and is scheduled to end on December 18, 2024, is the 2007 Edition through 2008 Addenda of Section XI of the ASME Code.

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 reactor vessel 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 in the licensee's submittal, limited accessibility and high-radiation levels in the area where the subject supports are located reduce the percentage of the supports available for visual examination. According to Figures 1 and 2 of the licensee's submittal, the reactor vessel 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 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 reactor vessel 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.

Based on the radiation exposure and the difficulty in obtaining access to the reactor vessel support areas, the NRC staff concludes that compliance with the ASME Code requirements would result 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 reactor vessel support structures, based on the licensee's proposed alternative in relief request I4R-02.

4.0 CONCLUSION

The NRC staff concludes that the examinations performed to the extent practical provide reasonable assurance of structural integrity of the subject components and that compliance with the specified requirements would result in hardship without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed the regulatory requirements set forth in 10 CFR 50.55a(z)(2). Therefore, the NRC staff grants relief for the subject examination of the components, as requested in relief request RR I4R-02 for the fourth 10-year inservice inspection interval for Callaway, which began on December 19, 2014, and is scheduled to end on December 18, 2024.

All other ASME Code, Section XI requirements for which relief was not specifically requested and approved remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: C. Fairbanks

Date: January 29, 2016

If you have any questions, please contact the Project Manager, John Klos at 301-415-5136 or via e-mail at john.klos@nrc.gov.

Sincerely,

/RA/

Robert J. Pascarelli, Chief
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-483

Enclosure:
Safety Evaluation

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