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

August 23, 2016

Mr. Adam C. Heflin
President, Chief Executive Officer,
and Chief Nuclear Officer
Wolf Creek Nuclear Operating Corporation
Post Office Box 411
Burlington, KS 66839

SUBJECT: WOLF CREEK GENERATING STATION – REQUEST FOR RELIEF I4R-02 FOR ALTERNATIVE TO ASME CODE INSPECTION REQUIREMENTS FOR CLASS 1 REACTOR PRESSURE VESSEL SUPPORTS (CAC NO. MF7425)

Dear Mr. Heflin:

By application dated February 23, 2016, the Wolf Creek Nuclear Operating Corporation (WCNOC, the licensee) submitted its inservice inspection (ISI) program for the fourth 10-year interval at Wolf Creek Generating Station (WCGS). Included in the licensee's submittal (Attachment II to the application) was request for relief (RR) I4R-02 from certain requirements of the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code), Section XI, requirements for ISI of the Class 1 reactor pressure vessel (RPV) supports.

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) paragraph 50.55a(z)(2), the licensee requested to use a proposed alternative 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.

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation, that imposition of the ASME Code requirements would result in a hardship without a compensating increase in quality and safety, and that 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 I4R-02 is authorized pursuant to 10 CFR 50.55a(z)(2) for the fourth 10-year ISI interval, which began on September 3, 2015, and ends on September 2, 2025.

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

The detailed results of the NRC staff review are provided in the enclosed safety evaluation. If you have any questions concerning this matter, please call Mr. F. Lyon of my staff at (301) 415-2296 or by electronic mail at fred.lyon@nrc.gov.

Sincerely,

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

Docket No. 50-482

Enclosure
Safety Evaluation

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOURTH 10-YEAR INTERVAL INSERVICE INSPECTION

REQUEST FOR RELIEF I4R-02

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

1.0 INTRODUCTION

By application dated February 23, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16061A072), the Wolf Creek Nuclear Operating Corporation (WCNOC, the licensee) submitted its inservice inspection (ISI) program for the fourth 10-year interval at Wolf Creek Generating Station (WCGS). Included in the licensee's submittal (Attachment II to the application) was request for relief (RR) I4R-02 from the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code), Section XI, requirements for ISI of the Class 1 reactor pressure vessel (RPV) supports. The relief was requested for the fourth 10-year interval ISI program, which began on September 3, 2015, and ends on September 2, 2025.

2.0 REGULATORY REQUIREMENTS

Pursuant to Title 10 of the *Code of Federal Regulations* (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, "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 also 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(a) 12 months prior to the start of the 120-month interval, subject to the conditions listed in 10 CFR 50.55a(b).

The Code of record for the fourth 10-year interval ISI program, which began on September 3, 2015, and ends on September 2, 2025, is the 2007 Edition with the 2008 Addenda of Section XI of the ASME Code.

The licensee may request alternatives to the requirements of Section XI of the ASME Code on the basis that compliance with the specified requirements would result in hardship or unusual

Enclosure

difficulty without a compensating increase in the level of quality or safety, pursuant to 10 CFR 50.55a(z)(2). The regulations in 10 CFR 50.55a(z) state, in part, 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 the licensee demonstrates that: (1) the proposed alternative provides 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. The proposed alternative must be submitted and authorized prior to implementation.

Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request the use of an alternative and the NRC to authorize the proposed alternative.

3.0 TECHNICAL EVALUATION

ASME Code Components

ASME Code Class: Class 1

Components: Reactor Vessel Supports

Component IDs: RBB01-01
RBB01-02
RBB01-03
RBB01-04

Applicable ASME Code Requirement

The licensee has proposed an alternative to the ASME Code requirements in Section XI, Table IWF-2500-1, Category F-A, Item Number F1.40, which requires that 100 percent of Class 1 supports, other than piping supports, be subject to a visual, VT-3 examination once every inspection interval.

Basis for Licensee's Proposed Alternative Examination

In its application, the licensee's statements that are the basis for the proposed alternative to the ASME Code requirements are as follows:

> In lieu of implementing the requirements of [ASME Code, Section XI,] Table IWF-2500-1, Category F-A, Item No. F1.40, Wolf Creek Nuclear Operating Corporation (WCNOC) proposes to perform a limited VT-3 visual examination, with the walk plate and insulation installed, on the accessible [Division 1, Subsection] 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.

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 supports and associated insulation/walk plate to allow 100 percent 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 Wolf Creek Generating Station (WCGS) 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 [in Attachment II to the licensee's submittal] 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. The majority 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.

A design change to the reactor pressure vessel supports and associated insulation/walk plate to allow 100 percent visual examination of the subject supports would be an extensive and costly modification. Further, it is estimated that the removal and reinstallation 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

Table IWF-2500-1 of the ASME Code requires that 100 percent of Category F-A, Item Number F1.40 supports be subject to a visual, VT-3 examination once every inspection interval. Table IWF-2500-1 allows that examination may be limited to portions of supports that are accessible for examination without disassembly or removal of support members, but makes no allowance to limit examinations based on high radiation levels. The acceptance criteria for the

VT-3 examination of Category F-A, Item Number F1.40 supports are described in Article IWF-3400 of the ASME Code. Article IWF-3410(a) of the ASME Code includes a list of component support conditions which are unacceptable for continued service. If VT-3 examination detects relevant conditions which are unacceptable for continued service, Articles IWF-3122.2 and IWF-3122.3 describe how some of these conditions may be accepted based on corrective measures, repair/replacement activities, or evaluation/testing.

According to Figures 1 and 2 of Attachment II to the licensee's submittal, the RPV is supported at the two cold leg nozzles and the two hot leg nozzles by Category F-A, Item Number F1.40 supports. The RPV support assembly at each of these nozzles 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 majority 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 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 re-installation 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.

As an alternative to the ASME Code requirements which would necessitate the removal of the steel walk plate and insulation, the licensee 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. The licensee stated that if conditions that do not meet the Article IWF-3400 acceptance standards are discovered during this limited VT-3 examination, the walk plate or insulation will be removed, if necessary, in order to meet the requirements of Articles IWF-3122.2 or IWF-3122.3.

The NRC staff determined that based on the radiation exposure and difficulty in obtaining access to the RPV support area, the ASME Code requirements are a hardship without a compensating increase in quality and safety. The NRC staff further determined that the licensee's proposed alternative provides reasonable assurance of the structural integrity of the RPV support structures.

4.0 CONCLUSION

For RR I4R-02, the NRC staff concludes that imposition of the ASME Code requirements would result in a hardship without a compensating increase in quality and safety, and that 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(z)(2) for the fourth 10-year ISI interval. 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: J. Jenkins, NRR/DE/EVIB

Date: August 23, 2016

The detailed results of the NRC staff review are provided in the enclosed safety evaluation. If you have any questions concerning this matter, please call Mr. F. Lyon of my staff at (301) 415-2296 or by electronic mail at fred.lyon@nrc.gov.

Sincerely,

/RA/

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

Docket No. 50-482

Enclosure
Safety Evaluation

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DATE	8/19/16	8/18/16	8/17/16	8/23/16

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