June 16, 2006

Mr. Rick A. Muench
President and Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
Post Office Box 411
Burlington, KS 66839

SUBJECT: WOLF CREEK GENERATING STATION - RELATED TO RELIEF REQUEST NO. I3R-04 FOR THE THIRD 10-YEAR INTERVAL INSERVICE INSPECTION (TAC NO. MD0299)

Dear Mr. Muench:

By letter dated March 2, 2006 (ET 06-0010), the Wolf Creek Nuclear Operating Corporation (the licensee) submitted its inservice inspection (ISI) program for the third 10-year interval at Wolf Creek Generating Station (WCGS). Included in the submittal were the following three relief requests (RRs): I3R-01, I3R-02, and I3R-04. This letter only addresses RR I3R-04.

In the enclosed safety evaluation (SE), the Nuclear Regulatory Commission (NRC) staff has evaluated the information provided by the licensee for the proposed third 10-Year ISI interval RR I3R-04 for WCGS. Based on the SE, the NRC staff concludes that compliance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requirements would result in a hardship without a compensating increase in quality and safety, and the licensee's proposed alternative provides reasonable assurance of the structural integrity of the reactor pressure vessel support structures. Therefore, the NRC staff concludes that the licensee's proposed alternative is authorized pursuant to paragraph 50.55a(a)(3)(ii) of Title 10 of the *Code of Federal Regulations* for the 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.

Sincerely,

/RA/

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

Docket No. 50-482

Enclosure: Safety Evaluation

cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO THE REQUEST FOR RELIEF NO. I3R-04

FOR THE THIRD 10-YEAR INTERVAL INSERVICE INSPECTION

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

1.0 INTRODUCTION

By application dated March 2, 2006 (Agencywide Documents Access and Management System Accession No. ML060720142), the Wolf Creek Nuclear Operating Corporation (the licensee) submitted its inservice inspection (ISI) program for the third 10-year interval at Wolf Creek Generating Station (WCGS). Included in the licensee's submittal were the following three relief requests (RRs): I3R-01, I3R-02, and I3R-04. This safety evaluation addresses only RR I3R-04.

2.0 REGULATORY REQUIREMENTS

ISI of Class 1, 2, and 3 components, is performed in accordance with Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (i.e., the ASME Code), and applicable addenda as required by paragraph 50.55a(g) of Part 50 of Title 10 of the *Code of Federal Regulation* (10 CFR), except where specific relief from the ASME Code has been granted by the 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. In addition, paragraphs 50.55a(g)(5)(iii) and 50.55a(g)(6)(i) state that a licensee, upon the determination that an ASME Code requirement is impractical for its facility, can notify the NRC and the NRC may grant such relief from the requirement and may impose such an alternative as it determines is authorized by law and will not endanger life or property or the common defense and security, and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

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) twelve months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein.

The ASME Code of record for the current WCGS third 10-year interval ISI program, which began on September 3, 2005, and ends on September 2, 2015, is the 1998 Edition through the 2000 Addenda of Section XI of the ASME Code.

3.0 TECHNICAL EVALUATION FOR RR NO. I3R-04

ASME Code Components:

The applicable ASME Code components for the relief request are the following: the reactor pressure vessel (RPV) supports, which have component numbers RBB01-01, RBB01-02, RBB01-03, and RBB01-04.

Applicable ASME Code Requirement:

The licensee has requested relief from 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 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(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 Wolf Creek 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 3 to the licensee's application] 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. 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.

The large cost 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 WCGS 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 stated that if conditions that do not meet the 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 IWF-3122.2 or IWF-3122.3.

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 March 2, 2006, 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 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.

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 significant 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.

In its application, the licensee requested relief pursuant to 10 CFR 50.55a(g)(5)(iii), in that 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. However, the licensee showed that the design modification was needed because of the radiation levels in the area and the access to the supports is difficult, and the licensee provided an alternative to the ASME Code requirements. Based on this, the NRC staff concludes that the alternative may be authorized pursuant to 10 CFR 50.55a(3)(ii) in that there is a significant hardship in meeting the ASME Code requirements without a compensating increase in quality and safety and the licensee's proposed alternative provides reasonable assurance of the structural integrity of the RPV support structures.

4.0 CONCLUSION

For RR I3R-04, 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 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 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: Tom McLellan

Date:

cc:

Jay Silberg, Esq.
Pillsbury Winthrop Shaw Pittman LLP
2300 N Street, NW
Washington, D.C. 20037

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

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P.O. Box 311
Burlington, KS 66839

Chief Engineer, Utilities Division
Kansas Corporation Commission
1500 SW Arrowhead Road
Topeka, KS 66604-4027

Office of the Governor
State of Kansas
Topeka, KS 66612

Attorney General
120 S.W. 10th Avenue, 2nd Floor
Topeka, KS 66612-1597

County Clerk
Coffey County Courthouse
110 South 6th Street
Burlington, KS 66839

Chief, Radiation and
Asbestos Control Section

Chief, Radiation and
Asbestos Control Section
Kansas Department of Health
and Environment
Bureau of Air and Radiation
1000 SW Jackson, Suite 310
Topeka, KS 66612-1366

Vice President Operations/Plant Manager
Wolf Creek Nuclear Operating Corporation
P.O. Box 411
Burlington, KS 66839

Supervisor Licensing
Wolf Creek Nuclear Operating Corporation
P.O. Box 411
Burlington, KS 66839

U.S. Nuclear Regulatory Commission
Resident Inspectors Office/Callaway Plant
8201 NRC Road
Steedman, MO 65077-1032