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

February 12, 2015

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 NO. I3R-10 FOR THE THIRD 10-YEAR INSERVICE INSPECTION PROGRAM INTERVAL (TAC NO. MF4305)

Dear Mr. Heflin:

By letter dated June 26, 2014, as supplemented by letter dated September 30, 2014, Wolf Creek Nuclear Operating Corporation (WCNOC, the licensee) proposed an alternative to the inservice inspection (ISI) interval requirements of the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code), Section XI, ISI Program, for the Wolf Creek Generating Station (WCGS).

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) paragraph 50.55a(a)(3)(ii), relief request I3R-10 proposed an alternative to the pressure test requirements of ASME Code, Section XI, paragraph IWB-5220 for the Class 1 piping and components connected to the reactor coolant system (RCS) that are isolated from direct RCS pressure during normal operation by their location, either upstream of a check valve, between two check valves or between two closed valves that must remain closed during the unit's operation in Modes 1, 2, or 3. The licensee proposed an alternative pressure boundary for the ASME Code system leakage test on the basis that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The paragraph headings in 10 CFR 50.55a were changed by *Federal Register* notice dated November 5, 2014 (79 FR 65776), which became effective on December 5, 2014 (e.g., 10 CFR 50.55a(a)(3)(i) is now 50.55a(z)(1), and 50.55a(a)(3)(ii) is now 50.55a(z)(2)). See the cross-reference tables, which are cited in the notice, in the Agencywide Documents Access and Management System (ADAMS) at Accession No. ML14015A191 and ADAMS package Accession No. ML14211A050.

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation, that the proposed alternative provides reasonable assurance of structural integrity and leak tightness of the subject piping portions, and complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(2). Therefore, the NRC staff authorizes the use of I3R-10 at WCGS for remainder of the third 10-year ISI interval, which will end on September 2, 2015.

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.

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,

E. R. Oesterle

Eric R. Oesterle, Acting Chief
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-482

Enclosure
Safety Evaluation

cc w/encl: Distribution via Listserv

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

THIRD 10-YEAR INSERVICE INSPECTION PROGRAM INTERVAL

REQUEST FOR RELIEF NO. I3R-10

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

1.0 INTRODUCTION

By letter dated June 26, 2014, as supplemented by letter dated September 30, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML14182A087 and ML14280A483, respectively), Wolf Creek Nuclear Operating Corporation (WCNOC, the licensee) proposed an alternative to the inservice inspection (ISI) interval requirements of the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code), Section XI, ISI Program, for the Wolf Creek Generating Station (WCGS).

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) paragraph 50.55a(a)(3)(ii), relief request I3R-10 proposed an alternative to the pressure test requirements of ASME Code, Section XI, paragraph IWB-5220 for the Class 1 piping and components connected to the reactor coolant system (RCS) that are isolated from direct RCS pressure during normal operation by their location, either upstream of a check valve, between two check valves or between two closed valves that must remain closed during the unit's operation in Modes 1, 2, or 3. The licensee proposed an alternative pressure boundary for the ASME Code system leakage test on the basis that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. The relief request is for the remainder of the third 10-year ISI interval, which ends on September 2, 2015.

The paragraph headings in 10 CFR 50.55a were changed by *Federal Register* notice dated November 5, 2014 (79 FR 65776), which became effective on December 5, 2014 (e.g., 10 CFR 50.55a(a)(3)(i) is now 50.55a(z)(1), and 50.55a(a)(3)(ii) is now 50.55a(z)(2)). See the cross-reference tables, which are cited in the notice, at ADAMS Accession No. ML14015A191 and ADAMS package Accession No. ML14211A050.

Enclosure

2.0 REGULATORY EVALUATION

Pursuant to 10 CFR 50.55a(g)(4), the ASME Code Class 1, 2, and 3 components (including supports) must 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.

Pursuant to 10 CFR 50.55a(z), alternatives to the requirements of paragraph (g) of 10 CFR 50.55a may be used when authorized by the Director, Office of Nuclear Reactor Regulation. A proposed alternative must be submitted and authorized prior to implementation. The licensee must demonstrate (1) the proposed alternative would provide an acceptable level of quality and safety; or (2) compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

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

3.0 TECHNICAL EVALUATION

3.1 Component Affected

The components affected are ASME Code Class 1 piping and valves. In accordance with IWB-2500 (Table IWB-2500-1), they are classified as Examination Category B-P, Items B15.50 and B15.70.

The licensee identified this piping in the tables in the Attachment to RR I3R-10, as Portion 1 piping in the auxiliary pressurizer spray, Portion 2 piping in the RCS to the residual heat removal (RHR) pump suction, Portion 3 piping in the emergency core cooling system (ECCS) high head safety injection, Portion 4 piping in the ECCS intermediate and low head safety injection, Portion 5 piping in the hot-leg safety injection, and Portion 6 piping in the RCS vents and drains. These piping segments are connected to the RCS but isolated from direct RCS pressure (i.e., 2235 pounds per square inch gauge (psig)) during normal operation by their location (upstream of a check valve, between two check valves or between two closed valves that must remain closed during operation in Mode 1, 2, or 3).

The material of construction of this piping is Type 304 austenitic stainless steel.

3.2 Applicable Code Edition and Addenda

The Code of record for the third 10-year ISI interval is the 1998 Edition through 2000 Addenda to the ASME Code.

3.3 Duration of Relief Request

The licensee submitted this relief request for the third 10-year ISI interval, which will end on September 2, 2015.

3.4 ASME Code Requirement

The ASME Code, Section XI, IWB-2500, Table IWB-2500-1, Examination Category B-P, requires the system leakage test be conducted according to IWB-5220 and the associated VT-2 visual examination according to IWA-5240 prior to plant startup following each refueling outage. In accordance with IWB-5221(a), the system leakage test shall be conducted at a pressure not less than the pressure corresponding to 100 percent rated reactor power. In accordance with IWB-5222(a), the pressure retaining boundary during system leakage test shall correspond to the reactor coolant boundary, with all valves in the position required for normal reactor operation startup. The visual examination shall, however, extend to and include the second closed valve at the boundary extremity. In accordance with IWB-5222(b), the pressure retaining boundary during system leakage test conducted at or near the end of each inspection interval shall extend to all Class 1 pressure retaining components within the system boundary.

3.5 Proposed Alternative, Basis for Use, and Reason for Relief

The licensee proposed an alternative to IWB-5222(b). For all piping (i.e., Portions 1 through 6), the proposed alternative is to use a reduced test pressure equal to 300 psig or greater. The licensee stated that the actual pressures used in the system leakage test of the Portions 1 through 6 piping will exceed 300 psig as the test planning process, personnel safety, and/or the station and system conditions allow.

The licensee stated that the piping under consideration is subjected to borated water, which would leave a boron residue and show up as a visual indication of a leak if leakage should occur in any of the piping. The VT-2 visual examinations that are performed each refueling outage on all Class 1 piping would identify the boron residue as a relevant condition and potential leak location that would require further examination and evaluation.

The licensee stated that the reasons for this relief request are as follows. For Portion 1 piping between the upstream globe valve and the downstream check valve, initiating the auxiliary pressurizer spray flow at the RCS operating pressure to support testing of this piping would cause a thermal design transient "inadvertent auxiliary spray" due to the captive piping fluid volume being at ambient containment temperature (110 degrees Fahrenheit (°F)). This design transient has a limited number of allowed cycles (total of 10 thermal transients for the life of the plant) according to the WCGS Updated Safety Analysis Report (USAR), Table 3.9(N)-13. The transient would also exceed the nominal maximum allowed differential temperature limit specified between spray water and pressurizer steam space of 320 °F.

The licensee stated that Portion 2 piping is part of the RHR suction supply header from the hot leg. Normal operational practice prohibits open alignment of the inboard Class 1 RHR suction isolation valves (BBPV8702A and BBPV8702B) at an RCS pressure above 425 psig. The inboard isolation valves BBPV8702A and BBPV8702B are equipped with a permissive open interlock which requires the RCS pressure to be less than 360 psig. The BBPV8702A and

BBPV8702B interlock is a safety-related function to preclude the potential for an intersystem loss-of-coolant accident (LOCA). Opening the BBPV8702A or BBPV8702B at the full RCS operating pressure to support testing of Portion 2 piping to the ASME Code requirement would result in Technical Specification (TS) Limiting Condition of Operation (LCO) 3.4.14 not being met for defeat of the open permissive interlock and entry into Condition C. The BBPV8702A and BBPV8702B were not designed to assure closure against the resulting differential pressure associated with a line break or leak downstream of BBPV8702A/B.

The licensee stated that Portion 3 piping is associated with the high-pressure coolant safety injection supply to the cold legs. Normal operational practice prohibits establishment of flow through isolation valves EMV8815, BBV001, BBV0022, BBV0040, and BBV0059 with the full RCS pressure except for emergency or transient conditions. Alignment of this flow path with full RCS pressure conditions during Mode 3 would constitute a safety injection, and further result in a cold-leg thermal design transient for the associated piping and valves (design limit is 60 thermal transients for the life of the plant). TS LCO 3.4.14 imposes allowable leakage limits for the RCS pressure isolation valves (PIV), and establishment of the required test pressure could result in the LCO not being met and entry into Condition A, if an inboard PIV were to become unseated.

The licensee stated that Portion 4 piping is associated with the low and intermediate head safety injection supply to the cold leg on each loop. During normal operation, this piping is pressurized to the cold-leg accumulator pressure of 600 psig with a nominal differential pressure across the disc of approximately 1650 pounds per square inch (psi). Establishing the RCS operating pressure between the two check valves requires use of temporary test rig with the resulting personnel and equipment safety hazards. TS LCO 3.4.14 imposes allowable leakage limits for the RCS PIVs, and establishment of the required test pressure could result in the LCO not being met and entry into Condition A, if an inboard PIV were to become unseated.

The licensee stated that Portion 5 piping is associated with the low head and intermediate head safety injection supply to the hot legs. The piping cannot be pressurized to the operating pressure by alignment of either the intermediate or low head safety injection pumps through the normally closed hot-leg isolation valves (EMHV8802A and EMHV8802B) and, therefore, would require use of a temporary test rig with the resulting personnel and equipment safety hazards. Normal operational practice prohibits flow through these flow paths above Mode 4, except for emergency or transient conditions. Alignment of these flow paths above Mode 4 would constitute a manual safety injection, and could further result in a thermal design transient for the hot-leg safety injection nozzle. Alignment of the intermediate head safety injection pump hot-leg flow path would further require entry into the TS 3.5.2, ECCS-Operating during Modes 1, 2, and 3. TS LCO 3.4.14 imposes allowable leakage limits for RCS PIVs, and establishment of the required test pressure could result in the LCO not being met and entry into Condition A, if an inboard PIV were to become unseated.

The licensee stated that Portion 6 piping is associated with the double isolation manual valve pairs for RCS pressure boundary. During normal operation, all of the valves are maintained in a closed position, thus only the upstream side of the inboard isolation valve is assured to be exposed to full RCS pressure. No connections exist between the valve pairs for test connection, therefore testing of the piping between the RCS drain/vent double isolation valves by hydro pump or temporary jumper is not possible. Opening the inboard manual RCS isolation valve at full RCS operating pressure and elevated temperatures to support testing the subject piping in accordance with the ASME Code requirement would temporarily disable the "double valve barrier" between the RCS pressure boundary and the non-Code piping or the containment atmosphere as required by 10 CFR 50.55a(c)(2)(ii). Manually opening and closing these inboard valves at RCS pressure and temperature creates potential personnel safety hazards.

In its letter dated September 30, 2014, the licensee provided the following additional information in response to an NRC staff request for additional information dated August 21, 2014 (ADAMS Accession No. ML14230A757). The licensee stated that Portions 1 through 5 piping are insulated. In Portion 6 piping, the drain lines are insulated but the vent lines are not insulated. Portions 1 through 6 piping are accessible for the VT-2 visual examination in accordance with the ASME Code requirements. The licensee stated it will meet the VT-2 visual examination requirements under IWA-5241(b) for the non-insulated parts of Portions 1 through 6 piping, and IWA-5242(b) for the insulated parts.

The licensee stated that welds in Portions 1 through 6 piping are all butt welds. No socket welds are present within the boundaries identified for RR I3R-10. All butt welds identified within Portions 1 through 6 piping are included in the risk-informed ISI program, except four welds of the 1-inch line segments for the reactor vessel head vent in Portion 6 piping. Of the risk-informed welds identified, eight welds are included for volumetric examination in the third 10-year ISI interval. Five of these welds have been previously inspected and three of these welds are scheduled for inspection in the spring of 2015 refueling outage. There has not been any Class 1 pressure boundary leakage identified in Portions 1 through 6 piping during the third 10-year ISI interval.

The licensee stated that in the unlikely event of a through-wall leak during normal operation in Portions 1 through 6 piping, the leak would result in unidentified RCS leakage. WCGS's RCS leakage detection instrumentation has been designed to aid operators in differentiating between possible sources of detected leakage within the containment and identifying the physical location of the leak. The RCS leakage detection instrumentation in WCGS consists of the sump level and flow monitoring system, the containment atmosphere particulate radioactivity monitors, the containment cooler condensate monitoring system, containment gaseous radioactivity monitors, the containment humidity monitoring system, and containment temperature and pressure monitoring. TS 3.4.13, "RCS Operational Leakage," specifies leakage limits to limit system operation in the presence of leakage from the RCS components to amounts that do not compromise safety.

3.6 Basis for Hardship

The licensee stated that Portions 1, 3, 4, and 5 piping have check valves as the inboard or downstream RCS Class 1 isolation valve. This would requires the pressure test to be performed with the use of a test pressure rig (hydro pump) and the non-Class test pressure rig connections to attain the required RCS normal operating pressure. Operation of the test equipment would require that personnel be stationed near the open vent or drain valves which would expose them to unnecessary safety hazards in the event of a leak or break in any of the non-class connections. Additionally, pressurization of any of the piping in this manner has the potential to unseat the inboard check valves, creating the potential for introduction of test medium fluid (non-borated water) into the reactor coolant with a resulting inadvertent RCS dilution. TS LCO 3.4.14, "RCS Pressure Isolation Valve Leakage," imposes allowable leakage limits for RCS PIVs, and establishment of the required test pressure could result in the LCO not being met and entry into Condition A, if an inboard PIV were to become unseated.

The licensee stated that for Portion 2 piping, use of the test pressure rig and the non-class test pressure rig connections to test this piping at the full RCS operating pressure would require application of a compatible pressurized medium. This would result in exposing personnel stationed near open vent or drain valves to unnecessary safety hazards in the event of a leak or break in the non-Class connections.

The licensee stated that for Portion 6 piping, the inboard manual isolation valves would need to be opened to check for leakage in the piping between the inboard and outboard valves. This requires personnel to be in close proximity to high pressure and temperature during this operation. The personnel safety hazards are lessened at the proposed lower test pressure and temperature.

3.7 NRC Staff Evaluation

The NRC staff has evaluated RR I3R-10 pursuant to 10 CFR 50.55a(z)(2). The NRC staff focuses on whether compliance with the specified requirements of 10 CFR 50.55a(g), or portions thereof, would result in hardship or unusual difficulty, without a compensating increase in the level of quality and safety.

Hardship

The NRC staff found that requiring the licensee to comply with IWB-5222(b) and extend the pressure boundary to all Class 1 components within the system boundary when conducting system leakage test at or near the end of ISI interval would result in hardship. The basis for the hardship is as follows. During normal operation, the piping under consideration (Portions 1 through 6) is isolated from the reactor coolant by two valves. These valves were designed to serve as double isolation barrier to the reactor coolant pressure boundary. Opening or bypassing the valves to pressurize the pipe to perform the required system leakage test defeats the double isolation criteria and reduces safety of the plant operation. Alternatively, use of a temporary external hydro pump, non-Code connections, and compatible medium to pressurize the piping to the RCS operating pressure would pose unnecessary safety hazards to personnel operating the equipment and performing the test, and to those working nearby, in case of a break in any temporary non-code connections. During pressurization by use of a hydro pump, there also exists a potential to unseat the inboard check valves and inadvertently dilute or contaminate the RCS with test medium fluid. Exceeding allowable leakage limits if the inboard valve was to become unseated causes the TS to not be met.

The NRC staff notes that for Portion 1 piping, the licensee could initiate the auxiliary pressurizer spray flow at the RCS operating pressure to facilitate ASME Code required system leakage testing. However, this would cause a thermal design transient "inadvertent auxiliary spray." This transient is limited to a total of ten occurrences for the life of the plant. In addition, this transient would exceed the nominal maximum allowed differential temperature limit between spray water and pressurizer steam space. For Portion 2 piping, the licensee could open the RHR inboard suction isolation valve at the full RCS operating pressure to facilitate ASME Code required system leakage testing. This would result in TS LCO 3.4.14 not being met for defeat of the open permissive interlock. For Portion 3 piping, the licensee could establish flow through pressure isolation valves with the full RCS pressure. This would constitute a safety injection, and would result in a cold-leg thermal design transient. This transient is limited to a total of 60 occurrences for the life of the plant. This would also result in TS LCO 3.4.14 not being met. For Portion 5 piping, the licensee could align the intermediate or low head safety injection pumps through hot-leg injection valves which would constitute a manual safety injection. This would result in thermal design transient for the hot-leg safety injection nozzle, and would require entry into the TS 3.5.2. Therefore, the NRC staff determines that: concerns from defeating the double isolation requirements in 10 CFR 50.55a(c)(2)(ii), creating conditions that result in TS not being met, creating unnecessary thermal design transients, and subjecting personnel to unnecessary safety hazards constitute a hardship.

Test Pressure

In evaluating the licensee's proposed alternative, the NRC staff assessed whether it appeared that the licensee used the highest achievable test pressure to conduct system leakage testing and the manner in which the licensee adequately preformed the testing and the associated VT-2 visual examinations of the piping for leakage. The NRC staff found that the licensee will use the highest possible pressure that is obtainable to conduct system leakage test of Portions 1 through 6 piping. This testing will be accomplished without major modifications to existing configuration of Portions 1 through 6 piping, without causing thermal design transients, and without exceeding TS allowable requirements which also includes TS allowable leakage limits for pressure isolation valves. Specifically, the licensee's proposed system leakage test will subject Portions 1 through 6 piping to pressure of at least 300 psig. Higher pressures will be used as the testing process and the safety of plant systems and personnel allow. As part of this system leakage testing, the licensee will perform the associated VT-2 visual examination of the piping under consideration in accordance with the IWA-5240 requirements to identify any leak or boron residue. Therefore, the NRC staff determines that the licensee's proposed system leakage test using the proposed test pressure accompanied with the IWA-5240 required VT-2 visual examination is adequate because the licensee will perform the test with use of the highest obtainable pressure.

Safety Significance of Alternative Test Pressure

In addition to the analysis described above, the NRC staff evaluated the safety significance of performance of the system leakage test at an alternative reduced pressure. The NRC staff notes that piping contained in Portions 1 through 6 is made of austenitic stainless steel. Potential degradation mechanism of this piping can include fatigue and stress corrosion cracking (SCC). However, fatigue crack is known to have relatively slow growth and field experience has shown that SCC under the conditions associated with the piping under consideration is not expected. It is expected that any significant degradation of the piping under consideration would be detected by the system leakage test performed under proposed maximum possible test pressure.

The NRC staff notes that Portions 1 through 6 piping contains only butt welds. The volumetric examinations performed in accordance with requirements of the WCGS risk-informed ISI program during the third 10-year ISI interval revealed no Class 1 pressure boundary leakage in Portions 1 through 6 piping.

The NRC staff notes that if in an unlikely event, Portion 1-6 piping developed a through wall flaw and a leak, the WCGS existing reactor coolant leakage detection systems will be able to identify the leakage during normal operation, and the licensee will take appropriate corrective actions in accordance with the plant technical specifications. Therefore, the NRC staff determines that based on the alternative system leakage testing that subjects this piping to the maximum possible pressure and the performance of the ASME Code required VT-2 visual examinations, it is reasonable to conclude that if significant service-induced degradation occurs, evidence of that degradation will be detected either by the proposed examinations or the RCS leakage detection systems.

Therefore, the NRC staff finds that the proposed system leakage testing using the proposed test pressure is adequate to provide a reasonable assurance of structural integrity and leak tightness of Portions 1 through 6 piping. Complying with the requirement specified in IWB-5222(b) would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

4.0 CONCLUSION

As set forth above, the NRC staff determines that the proposed alternative provides reasonable assurance of structural integrity and leak tightness of the subject piping portions, and complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(2). Therefore, the NRC staff authorizes the use of I3R-10 at WCGS for remainder of the third 10-year ISI interval, which will end on September 2, 2015.

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

Principal Contributor: A. Rezai, NRR/DE/EPNB

Date: February 12, 2015

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.

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/

Eric R. Oesterle, Acting 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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