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

February 24, 2016

Mr. Oscar A. Limpias
Vice President-Nuclear and CNO
Nebraska Public Power District
72676 648A Avenue
Brownville, NE 68321

SUBJECT: COOPER NUCLEAR STATION - REQUEST FOR INSERVICE INSPECTION PROGRAM ALTERNATIVE RP5-01 FOR IMPLEMENTATION OF CODE CASE N-795 (CAC NO. MF6335)

Dear Mr. Limpias:

By letter dated June 9, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15167A066), as supplemented by letter dated November 9, 2015 (ADAMS Accession No. ML15321A012), Nebraska Public Power District (NPPD, the licensee) submitted the request for alternative RP5-01, "Implementation of Code Case N-795," to the U.S. Nuclear Regulatory Commission (NRC) for review and authorization.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, paragraph 50.55a(z)(2), the licensee proposed to use provisions similar to those of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) Case N-795, "Alternative Requirements for BWR [Boiling Water Reactor] Class 1 System Leakage Test Pressure Following Repair/Replacement Activities, Section XI, Division 1," to perform the leakage testing and associated visual examination for leakage following repair/replacement activities at Cooper Nuclear Station (CNS). ASME Code Case N-795 has not been approved for use by the NRC staff in Regulatory Guide 1.147, Revision 17. The licensee requested to use the 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 and safety.

As set forth in the enclosed safety evaluation, the NRC staff concludes that proposed alternative RP5-01 provides reasonable assurance of structural integrity and leak tightness, and that complying with the ASME Code requirement would result in a 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 use of the proposed alternative at CNS during the fifth 10-year ISI interval that will commence on March 1, 2016, and is scheduled to end on February 28, 2026, until such time as ASME Code Case N-795 is published in a future revision of Regulatory Guide 1.147, which is incorporated by reference in 10 CFR 50.55a.

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.

If you have any questions, please contact Thomas Wengert at 301-415-4037 or via e-mail at Thomas.Wengert@nrc.gov.

Sincerely,

Meena K. Khanna, Chief
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-298

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR ALTERNATIVE RP5-01

IMPLEMENTATION OF CODE CASE N-795 FOR

FIFTH 10-YEAR INSERVICE INSPECTION PROGRAM INTERVAL

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

By letter dated June 9, 2015, as supplemented by letter dated November 9, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML15167A066 and ML15321A012, respectively), Nebraska Public Power District (NPPD, the licensee) submitted request for alternative RP5-01, "Implementation of Code Case N-795," to the U.S. Nuclear Regulatory Commission (NRC) for review and authorization.

Specifically, the licensee proposes to use provisions similar to those of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) Case N-795, "Alternative Requirements for BWR [Boiling Water Reactor] Class 1 System Leakage Test Pressure Following Repair/Replacement Activities, Section XI, Division 1," to perform the leakage testing and associated visual examination for leakage (VT-2) following repair/replacement activities at Cooper Nuclear Station (CNS) during the fifth 10-year inservice inspection (ISI) interval. ASME Code Case N-795 has not been approved for use in Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability ASME Section XI, Division 1," Revision 17 (ADAMS Accession No. ML13339A689). The licensee requested to use the 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 and safety.

2.0 REGULATORY EVALUATION

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, paragraph 50.55a(g)(4), "Inservice inspection standards requirements for operating plants," 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

Enclosure

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 inspection interval and subsequent 10-year inspection 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 inspection interval, subject to the limitations and modifications listed therein.

The regulations in 10 CFR 50.55a(z) state, in part, that alternatives to the requirements of 10 CFR 50.55a(g) may be used, when authorized by the 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.

Based on analysis of the regulatory requirements, the NRC staff concludes that the regulatory authority exists to authorize the licensee's proposed alternative to the ASME Code requirement 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 and safety. Accordingly, the staff has reviewed and evaluated the licensee's request pursuant to 10 CFR 50.55a(z)(2).

3.0 TECHNICAL EVALUATION

3.1 The Licensee's Request for Alternative

The licensee is requesting relief from the pressure requirement of ASME Code, Section XI, required system leakage testing of ASME Code Section III, Class 1 mechanical joints made in the installation of pressure retaining items and the Class 1 pressure retaining boundary on which repair/replacement activities have been performed by welding.

ASME Code Requirements

The Code of Record for the CNS fifth 10-year ISI interval that will commence on March 1, 2016, and is scheduled to end on February 29, 2026, is ASME Code, Section XI, 2007 Edition through the 2008 Addenda.

For mechanical joints resulting from repair/replacement activities[1], ASME Code, 1998 Edition, Section XI, paragraph IWA-4540(c) requires mechanical joints made in the installation of pressure retaining items be pressure tested during a system leakage test in accordance with IWA-5211(a). IWB-5221(a) requires that the system leak test be conducted during operation at nominal operating pressure, or when pressurized to nominal operating pressure and temperature.

[1] 10 CFR 50.55a(b)(2)(xxvi), *Pressure Testing Class 1, 2 and 3 Mechanical Joints,* requires licensees using the ASME Code, Section XI, 2001 Edition and later editions and addenda to use the 1998 Edition of the ASME Code, Section XI, paragraph IWA-4540(c), for pressure testing Class 1, 2, and 3 mechanical joints.

For pressure retaining boundaries on which repair/replacement activities have been performed by welding, ASME Code, Section XI, paragraph IWA-4540 requires a hydrostatic or system leakage test in accordance with IWA-5000 prior to, or as part of, returning to service. IWA-5200 requires that a VT-2 examination be performed to detect leakage while the system is in operation, during a system operability test, or while the system is at test conditions using an external pressurization source at temperature and pressure defined in IWB-5000. IWB-5221(a) requires that the system leakage test to be conducted at a pressure not less than the pressure corresponding to 100 percent rated reactor power.

Licensee's Proposed Alternative

The licensee proposes to perform the system leakage test and associated VT-2 examination following repair/replacement activities in maintenance or forced outages in accordance with the provisions of ASME Code Case N-795, but using longer holding times than those specified in the code case. The system leakage test will be performed during the normal operational start-up sequence at a minimum of 900 pounds per square inch gauge (psig), approximately 90 percent of the pressure corresponding to 100 percent rated reactor power [1005 psig] with VT-2 examination following after a 1-hour holding time for uninsulated components and after an 8-hour holding time for insulated components.

Licensee's Basis for Requesting Relief

During normal startup with normal power ascension, nominal operating pressure of 1005 psig is reached at a reactor power level of approximately 85 percent. If access to containment were permitted at this power level, personnel would be exposed to excessive radiation levels, including significant exposure to neutron radiation fields, which is contrary to station as low as reasonably achievable (ALARA) practices.

The licensee stated that, during a maintenance or forced outage, there is a large decay heat load from the reactor core that is difficult to control once shutdown cooling (SDC) has been removed from service. During a short term mid-cycle shutdown, the projected heatup rate could be in the order of 0.4 degrees Fahrenheit per minute once SDC is removed from service. Under those conditions, the time available to pressurize up to test conditions, perform the VT-2 exam and return to SDC would be greatly reduced, and the hurried time frames may create a more error-likely environment. In addition, there is some inherent risk that mechanical, control or operational problems could occur while the SDC is isolated, which could delay return to SDC. Testing at these abnormal plant conditions/alignments results in additional risks and delays while providing little added benefit beyond tests, which could be performed at slightly reduced pressures under normal plant conditions.

As stated, in part, in the licensee's letter dated June 9, 2015:

> Indication of leakage identified through the VT-2 examinations during a test at a pressure correlating to either the 100% rated reactor power level or at ~90% of that value will not be significantly different between the two tests. Higher pressure under the otherwise same conditions will produce a higher flow rate but the difference is not significant. Code Case N-795 proposes increased hold times, as compared to a test performed at normal operating pressure, to allow for more leakage from the pressure boundary if a through-wall or mechanical joint

leakage condition exists. NPPD proposes to implement longer hold times than those specified by the code case. NPPD believes these longer hold times are justified to allow for additional leakage to accumulate at the area of interest so as to be more evident during the VT-2 examination, should a through-wall or mechanical joint leakage condition exist. This alternate test pressure, when combined with longer hold times, is still adequate to provide evidence of leakage, should a leak exist.

3.2 NRC Staff Evaluation

Performance of a system leakage test of pressure retaining boundaries, including mechanical joints, on which repair/replacement activities have been performed, is an integral part of ASME Code, Section XI requirements. The system leakage test for Examination Category B-P components normally occurs at the end of a refueling outage when the core decay heat has had time to decrease, some spent fuel has been removed, and some new fuel has been added, resulting in a relatively low decay heat load. The low decay heat load, compared to that for the high decay heat load found at the start of an outage, results in low heatup rates. When a system leakage test immediately follows a maintenance or forced outage, there is a large decay heat load from the reactor core that is difficult to control once SDC has been removed from service. Isolating SDC under high decay heat loads requires abnormal plant conditions/alignments and is accompanied by inherent risk, and the hurried time frames that result from the high heatup rates may create a more error-likely environment. In addition, there is inherent risk that mechanical, control, or operational problems could occur while the SDC is isolated.

ASME developed Code Case N-795 to provide an alternative test pressure for some Class 1 pressure tests following repair/replacement activities at BWR plants. The alternative was developed because some BWR licensees believe that the Class 1 pressure tests performed at pressures corresponding to 100 percent reactor power require abnormal plant conditions and alignments that increase risk. Code Case N-795 specifies that the leakage test shall be performed at a test pressure of at least 87 percent of that required by IWB-5221(a). Code Case N-795 requires that, before the VT-2 examination commences, a minimum 15-minute holding time for noninsulated components and a 6-hour holding time for insulated components shall be maintained.

In its response to the NRC staff's request for additional information (RAI), by letter dated November 9, 2015, the licensee detailed three methods that would permit VT-2 inspection while at a pressure corresponding to 100 percent normal operating pressure. The first of these methods would require the reactor pressure vessel (RPV) to be filled with coolant and the steam lines flooded to provide a water-solid condition. Use of this method would result in extensive valve manipulations, system lineups, and procedural controls in order to heat up and pressurize the primary system to establish the necessary test pressure without the withdrawal of control rods. The staff concludes that performance of a system leakage test at the conditions present immediately following a maintenance or forced outage using this method, would present multiple operational challenges, unusual difficulty, present a risk to plant operation, and therefore, would present a hardship.

The second method described in the licensee's RAI response would perform the VT-2 examination during normal startup procedures. Nominal operating pressure of 990 psig can be attained with normal startup and normal power ascension, at a reactor power level of approximately 100 percent. If access to containment were permitted at this power level, personnel would be exposed to excessive radiation levels, including significant exposure to neutron radiation fields, which is contrary to station ALARA practices. Establishing the 990 psig test conditions at a more moderate power level and in a manner to address the radiation concerns would require significant changes to the steam pressure control system. The NRC staff concludes that exposure of workers to high radiation fields would present a hardship.

The third method that could possibly be used would maintain the RPV at its normal level and use decay heat to produce sufficient steam pressure to conduct the test at nominal operating pressure. The licensee states that, while the decay heat load is too high for the water-solid method discussed above, there may not be sufficient decay heat available to perform the test at 1005 psig within a reasonable time period, if at all. The NRC staff concludes that use of this alternate method would also present a hardship.

The licensee proposes to use the provisions of ASME Code Case N-795, with additional conditions, for performance of a system leakage test of pressure retaining boundaries, including mechanical joints, on which repair/replacement activities have been performed. These conditions include:

a. Attainment of at least 90 percent of the operating pressure prior to the start of the holding time.
b. Holding time of 1 hour for uninsulated components prior to the VT-2 visual examination.
c. Holding time of 8 hours for insulated components prior to the VT-2 visual examination.

The system leakage test would comprise a VT-2 visual examination after the required test condition holding time. The NRC staff notes that the licensee has defined the nominal operating pressure to be a minimum of 1005 psig for components within the reactor coolant pressure boundary at CNS. Therefore, the system leakage test pressure must be at least ~900 psig before the holding time is started.

The leak tightness of components involved in the repair/replacement activities must be ensured. Leakage through an orifice will be related to the differential pressure at the point of leakage, or across the connection, and is expected to scale with the square root of the pressure. Therefore, the leakage rate at the required 90 percent test pressure would be approximately 95 percent the leakage rate at 100 percent power. A 10 percent reduction in the test pressure is not expected to result in the arrest of a leak that would occur at nominal operating pressure. In the unlikely event that leakage would occur subsequent to the VT-2 visual examination at pressures associated with 100 percent rated reactor power, leakage would be detected by the drywell monitoring systems that are required by technical specifications. The NRC staff therefore concludes that the VT-2 examination, after the specified holding time at 90 percent of system normal operating pressure, will adequately assure leak tightness of the components in the reactor coolant pressure boundary.

Based on the above evaluation, the NRC staff concludes that performing a VT-2 visual examination during a system leakage test at normal operating pressure following a maintenance or forced outage would present a hardship. The staff further concludes that performing the VT-2 examination at pressures equal to, or greater than, 900 psig, with holding times of 1 hour for non-insulated components and 8 hours for insulated components, will provide reasonable assurance of leak tightness. It is the NRC's position that structural integrity is ensured through compliance with ASME Code requirements for design, fabrication, and nondestructive examination.

4.0 CONCLUSION

As set forth above, the NRC staff concludes that proposed alternative RP5-01, "Implementation of Code Case N-795," provides reasonable assurance of structural integrity and leak tightness, and that complying with the ASME Code requirement would result in a 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 use of the proposed alternative at CNS during the fifth 10-year ISI interval that will commence on March 1, 2016, and is scheduled to end on February 28, 2026, until such time as ASME Code Case N-795 is published in a future revision of Regulatory Guide 1.147, which is incorporated by reference in 10 CFR 50.55a.

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

Principal Contributor: K. Hoffman

Date: February 24, 2016

If you have any questions, please contact Thomas Wengert at 301-415-4037 or via e-mail at Thomas.Wengert@nrc.gov.

Sincerely,

/RA/

Meena K. Khanna, Chief
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-298

Enclosure:
Safety Evaluation

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