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

July 31, 2018

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

SUBJECT: COOPER NUCLEAR STATION – REQUESTS FOR RELIEF ASSOCIATED WITH THE FIFTH 10-YEAR INSERVICE INSPECTION INTERVAL PROGRAM (CAC NOS. MG0175 THROUGH MG0179; EPIDS L-2017-LLR-0062 THROUGH L-2017-LLR-0066)

Dear Mr. Dent:

By letter dated August 17, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17241A048), as supplemented by letters dated March 8, 2018, March 14, 2018, April 26, 2018, and May 16, 2018 (ADAMS Accession Nos. ML18078A264, ML18082A563, ML18131A159, and ML18143B464, respectively), Nebraska Public Power District (the licensee) submitted Relief Requests RI5-01, RI5-02, Revision 1, RI5-03, RR5-02, and RR5-03, to the U.S. Nuclear Regulatory Commission (NRC). The licensee proposed alternatives to or requested relief from certain inservice inspection (ISI) test requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, "Rules for Inservice Inspection (ISI) of Nuclear Power Plant Components," at Cooper Nuclear Station (CNS), for the fifth 10-year ISI interval program, which commenced on April 1, 2016, and will end on February 28, 2026.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(z)(1), the licensee requested to use the proposed alternatives in Relief Requests RI5-01; RI5-02, Revision 1; and RI5-03, on the basis that the alternatives provide an acceptable level of quality and safety. Pursuant to 10 CFR 50.55a(z)(2), the licensee requested to use the proposed alternatives in RR5-02 and RR5-03 on the basis that the proposed alternatives will provide reasonable assurance of quality and safety of the subject components and compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The NRC staff has reviewed the subject requests and concludes as set forth in the enclosed safety evaluations, that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1) for requests RI5-01, RI5-02, Revision 1, and RI5-03, and in 10 CFR 50.55a(z)(2) for requests RR5-02 and RR5-03, and is in compliance with the ASME Code requirements. Therefore, the NRC staff authorizes alternative requests RI5-01, RI5-02, Revision 1, RI5-03, RR5-02, and RR5-03 at CNS for the fifth 10-year ISI interval program.

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

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

Sincerely,

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

Docket No. 50-298

Enclosures:
1. Safety Evaluation – Relief Request RI5-01
2. Safety Evaluation – Relief Request RI5-02
3. Safety Evaluation – Relief Request RI5-03
4. Safety Evaluation – Relief Request RR5-02
5. Safety Evaluation – Relief Request RR5-03

cc: Listserv

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

ALTERNATIVE REQUEST NO. RI5-01

FOR THE FIFTH 10-YEAR INSERVICE INSPECTION INTERVAL

FOR THE PERIOD OF EXTENDED OPERATION

REGARDING REACTOR PRESSURE VESSEL

CIRCUMFERENTIAL WELD EXAMINATIONS

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

By letter dated August 17, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17241A048), Nebraska Public Power District (the licensee) submitted a request to the U.S. Nuclear Regulatory Commission (NRC) for the use of alternatives to certain requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, "Rules for Inservice Inspection (ISI) of Nuclear Power Plant Components," for the reactor pressure vessel (RPV) circumferential shell weld examinations at Cooper Nuclear Station (CNS). The licensee's proposed alternative is identified as request for alternative RI5-01. The licensee's request for the use of this alternative was submitted, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(z)(1), on the basis that the alternative would provide an acceptable level of quality and safety.

The ASME Code, Section XI alternative proposed in the licensee's submittal dated August 17, 2017, would eliminate the requirement to inspect the RPV circumferential shell welds, except for the areas of intersection with the axial welds, for the duration of the unit's 20-year extended license term, also referred to as the period of extended operation (PEO). The licensee's proposed alternative addressed the specific guidance provided in the NRC staff's final safety evaluation (SE) dated July 28, 1998, for Boiling Water Reactor (BWR) Vessel and Internals Project (BWRVIP) Topical Report BWRVIP-05, "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations" (Legacy Library Accession No. 9808040037). This specific guidance provided staff expectations and acceptance criteria for plant-specific applications for Code alternatives to implement the BWRVIP-05 probabilistic fracture mechanics (PFM) methodology in lieu of the subject RPV circumferential shell weld examinations for the original 40-year license term.

2.0 REGULATORY EVALUATION

The ISI of ASME Code Class 1, 2, and 3 components is to be performed in accordance with Section XI of the ASME Code and applicable editions and addenda as required by 10 CFR 50.55a(g), "Preservice and inservice inspection requirements," except where specific relief has been granted by the NRC pursuant to 10 CFR 50.55a(g)(6)(i), "Impractical ISI requirements: Granting of relief."

Pursuant to 10 CFR 50.55a(z), "Alternatives to codes and standards requirements," alternatives to the requirements of paragraph (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.

Pursuant to 10 CFR 50.55a(g)(4), "Inservice inspection standards requirements for operating plants," 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, 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 ISI 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)(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).

CNS is currently in the fifth 10-year ISI interval, which began on April 1, 2016. The applicable ASME Code of record for the fifth 10-year ISI intervals at CNS is the ASME Code, Section XI, 2007 Edition through 2008 Addenda.

2.1 Requirements Related to Neutron Fluence

The NRC has established requirements in Appendix G, "Fracture Toughness Requirements," to 10 CFR Part 50, in order to protect the integrity of the reactor coolant pressure boundary (RCPB) in nuclear power plants. The regulations in 10 CFR Part 50, Appendix G, require that the pressure-temperature (P-T) limits for an operating light-water nuclear reactor be at least as conservative as those that would be generated if the methods of Appendix G, "Fracture Toughness Criteria for Protection Against Failure," to Section XI of the ASME Code were used to generate the P-T limits. The regulations in 10 CFR Part 50, Appendix G, also require that applicable surveillance data from RPV material surveillance programs be incorporated into the calculations of plant-specific P-T limits, and that the P-T limits for operating reactors be generated using a method that accounts for the effects of neutron irradiation on the material properties of the RPV beltline materials.

Table 1 of 10 CFR Part 50, Appendix G, provides the NRC staff's criteria for meeting the P-T limit requirements of the ASME Code, Section XI, Appendix G, as well as the minimum temperature requirements of the rule during normal and pressure testing operations.
In addition, the NRC staff's regulatory guidance related to P-T limit curves is found in Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," dated May 1988 (ADAMS Accession No. ML003740284), and NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water

Reactor] Edition," Section 5.3.2, "Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock" (ADAMS Accession No. ML070380185).

The P-T limit curve calculations are based, in part, on the reference nil-ductility temperature (RT_{NDT}) for the material, as specified in the ASME Code, Section XI, Appendix G. The regulations in 10 CFR Part 50, Appendix G, require that RT_{NDT} values for materials in the RPV beltline region be adjusted to account for the effects of neutron radiation. The guidance in RG 1.99, Revision 2, contains methodologies for calculating the adjusted RT_{NDT} (ART) due to neutron irradiation. The ART is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin term.

The ΔRT_{NDT} is a product of a chemistry factor (CF) and a fluence factor. The CF is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Revision 2, or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the CF was determined using the tables in RG 1.99, Revision 2, or surveillance data. The margin term is used to account for uncertainties in the values of the initial RT_{NDT}, the copper and nickel contents, the neutron fluence, and the calculational procedures. The guidance in RG 1.99, Revision 2, describes the methodology to be used in calculating the margin term.

Appendix H, "Reactor Vessel Material Surveillance Program Requirements," to 10 CFR Part 50, provides the NRC staff's criteria for the design and implementation of RPV material surveillance programs for operating LWRs.

In March 2001, the NRC staff issued RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" (ADAMS Accession No. ML010890301). Fluence calculations for use in ART and P-T limit curve analyses are acceptable if they are performed with approved methodologies or with methods that are shown to conform to the guidance in RG 1.190.

The guidance in RG 1.190 describes methods and assumptions acceptable to the NRC staff for determining the pressure vessel neutron fluence with respect to the General Design Criteria (GDC) contained in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50. In consideration of the guidance set forth in RG 1.190; GDC 14, "Reactor coolant pressure boundary"; GDC 30, "Quality of reactor coolant pressure boundary"; and GDC 31, "Fracture prevention of reactor coolant pressure boundary," are applicable. GDC 14 requires the design, fabrication, erection, and testing of the RCPB so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. GDC 30 requires, among other things, that components comprising the RCPB be designed, fabricated, erected, and tested to the highest quality standards practical. GDC 31 pertains to the design of the RCPB, which states:

> The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties,

(2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

3.0 TECHNICAL EVALUATION

3.1 ASME Code Requirement to which the Alternatives are Requested

The ASME Code, Section XI, 2007 Edition through 2008 Addenda, Table IWB-2500-1, Examination Category B-A, Item B1.11 requires a volumetric examination of all the RPV circumferential shell welds each ISI interval, to include volumetric examination of "essentially 100 percent" (i.e., greater than 90 percent) of the length of the welds.

3.2 Component(s) for which the Alternatives are Requested

Code Class: 1
Examination Category: B-A
Item Number: B1.11, Circumferential Shell Welds
Weld Nos.: VCB-BB-1, VCB-BA-2, VCB-BB-3, VCB-BB-4
Examination Method: Volumetric

3.3 Licensee's Proposed Alternatives to the ASME Code Section XI

The licensee's application dated August 17, 2017, identified that CNS was operating with NRC-authorized Code alternatives that allowed plant-specific implementation of the BWRVIP-05 PFM methods in lieu of the subject RPV circumferential shell weld examination requirements for the remainder of the 40-year license term. This 40-year Code alternative was authorized for CNS in an NRC letter dated February 6, 2008 (ADAMS Accession No. ML080230288).

The 40-year license term ended on January 18, 2014, for CNS. Therefore, request for alternative RI5-01 was submitted to implement the BWRVIP-05 PFM methods in lieu of the subject RPV circumferential shell weld examination requirements for the duration of the 20-year extended license term.

3.4 Licensee's Basis for the Proposed Alternatives

The licensee submitted request for alternative RI5-01 in accordance with 10 CFR 50.55a(z)(1), on the basis that the proposed alternatives would provide an acceptable level of quality and safety. The licensee's technical basis for determining an acceptable level of quality and safety included plant-specific evaluations for demonstrating that the limiting RPV circumferential shell weld at CNS has conditional failure probabilities that are bounded by (i.e., less than) the NRC staff's acceptance criteria for the weld failure probabilities, considering projected RPV weld neutron embrittlement through the end of the PEO. The NRC staff's specific acceptance criteria for these circumferential shell weld failure probabilities were established in its SE dated July 28, 1998, for the BWRVIP-05 report.

The licensee determined that these RPV circumferential shell weld evaluations demonstrate that implementation of the proposed Code alternative for the duration of the 20-year extended license term would provide an acceptable level of quality and safety at CNS.

3.5 NRC Staff Evaluation

By letter dated February 6, 2008, the NRC staff authorized an alternative to the volumetric examination requirements of the ASME Code, Section XI, for the subject RPV circumferential shell welds at CNS, pursuant to 10 CFR 50.55a(a)(3)(i), which is now 10 CFR 50.55a(z)(1). This NRC-authorized alternative allowed for plant-specific implementation of the BWRVIP-05 RPV PFM analyses, as approved by the NRC staff in its BWRVIP-05 SE, in lieu of the subject ASME Code, Section XI examination requirements, for the duration of the unit's 40-year license term. The subject Code alternative expired when CNS entered the 20-year extended license term on January 18, 2014. Therefore, plant-specific implementation of the BWRVIP-05 PFM methods in lieu of the subject ASME Code, Section XI requirements during the PEO requires the submittal of a new request for a Code alternative, pursuant to 10 CFR 50.55a(z)(1).

The licensee's application dated August 17, 2017, requested alternatives to the subject circumferential weld examination requirements for the PEO at CNS, based on plant-specific implementation of the NRC-approved BWRVIP-05 methods for the limiting circumferential shell weld at CNS, considering projected RPV weld neutron embrittlement through 60 years of facility operation. The proposed 60-year Code alternatives included plant-specific calculations demonstrating that projected neutron embrittlement for the CNS limiting RPV circumferential shell weld is less than that used by the NRC staff for calculating an acceptable circumferential shell weld conditional failure probability[1], as documented in the NRC SE for BWRVIP-05.
To project neutron embrittlement for 60 years, the licensee used calculations based on updated fluence values from its Pressure and Temperature Limits Report (PTLR) dated January 9, 2017 (ADAMS Accession Nos. ML17018A151 and ML17018A152). These updated fluence values were based on 54 effective full power years (EFPY), which is equivalent to 60 years of facility operation at CNS. The NRC staff confirmed that the updated fluence values are the most recent licensing basis and would bound 60 years of facility operation at CNS. The conditional failure probabilities documented in the NRC SE for BWRVIP-05 are based on 64 EFPY, which would bound 60 years of facility operation at CNS.

The NRC staff confirmed that the proposed 60-year Code alternatives continued implementation of certain operator procedures and training needed to limit the frequency of cold overpressure events, per the criteria specified in the staff's SE for BWRVIP-05. The staff had previously endorsed these provisions in Section 4.2.5 of its safety evaluation report for the CNS license renewal application (LRA) (NUREG-1944, dated October 2010 (ADAMS Accession No. ML103070009)), regarding the subject circumferential weld analysis. The operator training and procedures are specifically needed to ensure that the overall plant-specific RPV failure probability per reactor operating year (a product of the weld conditional failure probability and the cold overpressure event frequency) is less than the acceptance criterion specified in the staff's SE for BWRVIP-05.

The specific RPV weld neutron embrittlement parameter used for this evaluation is referred to as the mean RT_{NDT}. The mean RT_{NDT} value for demonstrating an acceptable RPV weld conditional failure probability is calculated based on three inputs:

(1) *The Projected RPV Neutron Fluence*: RPV neutron fluence, as determined based on staff-approved calculation methodologies, is the key time-dependent

[1] The weld conditional failure probability quantifies the probability of weld failure if the RPV was subjected to a cold overpressure event, as addressed in BWRVIP-05.

parameter for all RPV integrity analyses that consider neutron embrittlement of the RPV beltline materials. The projected neutron fluence input to the mean RT_{NDT} value, for demonstrating an acceptable RPV weld conditional failure probability at the end of the licensed operating term, shall include the effects of any power uprates that are implemented during the licensed operating term of the unit.

(2) *The Weld Chemistry Factor (CF)*: The CF is determined based on both copper and nickel content, or the application of credible RPV material surveillance data from a 10 CFR Part 50, Appendix H RPV material surveillance program. If the weld is represented in the plant-specific or industry integrated surveillance program, all credible RPV surveillance data shall be used for the CF calculation, per the requirements of 10 CFR Part 50, Appendix G. CF values shall be periodically recalculated based on new credible RPV surveillance data that becomes available when a surveillance capsule is withdrawn from the RPV and tested in accordance with 10 CFR Part 50, Appendix H surveillance program requirements.

(3) *The Initial (Unirradiated) RT_{NDT}*: The initial RT_{NDT} is determined in accordance with the requirements of 10 CFR Part 50, Appendix G, based on the procured RPV material impact test data or the use of NRC-approved methods in NUREG-0800, Branch Technical Position 5-3, "Fracture Toughness Requirements" (ADAMS Accession No. ML070850035), as applicable to the unit. This is expected to remain fixed throughout the operating life of the plant.

It should be noted that the LRA mean RT_{NDT} calculations used RPV weld neutron fluence and CFs that were valid at the time of the LRA review. Accordingly, the licensee's application dated August 17, 2017, for the subject Code alternatives considered that it was necessary to recalculate the limiting circumferential weld mean RT_{NDT} values using updated neutron fluence values from the updated PTLR dated January 9, 2017. Based on increased neutron fluence values, the limiting circumferential weld mean RT_{NDT} values increased as well, but remained below the bounding circumferential weld mean RT_{NDT} values from BWRVIP-05.

The NRC staff reviewed the CF value and initial RT_{NDT} value for the limiting RPV circumferential shell weld at CNS and determined that they are the same as those used for the updated PTLR dated January 9, 2017. The staff also confirmed that the licensee correctly calculated the limiting circumferential shell weld mean RT_{NDT} value for CNS. Therefore, the staff determined that the licensee's mean RT_{NDT} calculation for the proposed 60-year Code alternative adequately demonstrated that the limiting circumferential shell weld at CNS satisfies the mean RT_{NDT} acceptance criteria established in the staff's SE for BWRVIP-05, for ensuring an acceptable circumferential shell weld conditional failure probability. Accordingly, the staff finds that the licensee's analysis of the CNS limiting circumferential shell weld, as provided in its submittal dated August 17, 2017, is acceptable for satisfying the specific circumferential shell weld PFM acceptance criteria established in the NRC staff's BWRVIP-05 SE for the PEO at CNS.

Analysis of RPV Axial Welds for BWR Plants that have Entered the PEO (BWRVIP-74-A)

The NRC staff's acceptance of U.S. BWRs 40-year Code alternatives for the RPV circumferential shell welds was based, in part, on having an acceptable generic RPV axial weld failure probability for the BWR fleet. Notably, the staff's March 7, 2000, supplemental SE

(ADAMS Accession No. ML003690281) for BWRVIP-05 specifically addressed the BWRVIP's generic analysis of RPV axial weld failure probability for supporting the plant-specific 40-year Code alternatives for elimination of RPV circumferential shell weld exams. In its supplemental SE, the staff stated that based on its review of the BWRVIP's generic axial weld PFM results, the limiting RPV axial weld failure probability for the BWR fleet at the end of the 40-year license term for all BWR units is acceptable, given the assumptions described in the supplemental SE. The supplemental SE for BWRVIP-05 also stated that licensees would need to perform plant-specific evaluations (referred to as time-limited aging analyses, or TLAAs) of axial weld failure probability in LRAs to support demonstration that the PFM basis for elimination of circumferential shell weld exams remains acceptable for PEOs. These plant-specific axial weld evaluations would need to demonstrate acceptability using the NRC staff's specific acceptance criteria for axial weld failure probabilities from the supplemental SE for BWRVIP-05, dated March 7, 2000.

Subsequently, by letter dated October 18, 2001 (ADAMS Accession No. ML012920549), the NRC staff issued its final license renewal safety evaluation report (LR-FSER) for the BWRVIP-74-A report, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines," wherein the staff identified that BWR licensee renewal applicants referencing the BWRVIP-74-A for their RPV neutron embrittlement TLAAs must evaluate both the RPV circumferential shell weld and axial shell weld failure probabilities as TLAAs for their proposed PEOs. The LR-FSER for BWRVIP-74-A indicates that an acceptable plant-specific evaluation of axial weld failure probability may consist of a plant-specific determination of the mean RT_{NDT} of the most limiting RPV axial beltline weld, based on projected neutron embrittlement at the end of the 60-year license term, and demonstrating that it is less than the values specified in Table 1 of the LR-FSER for BWRVIP-74-A. The LR-FSER Table 1 values correspond to the axial weld acceptance criteria cited above from the March 7, 2000, supplemental SE for BWRVIP-05.

Based on the above acceptance criteria, Section 4.2.5 of the CNS LRA includes TLAAs that determined the 60-year projected mean RT_{NDT} values for the limiting RPV circumferential and axial shell welds. As documented in Sections 4.2.5.4 of NUREG-1944, the NRC staff concluded that these analyses are acceptable for demonstrating compliance with the requirement for TLAAs set forth in 10 CFR 54.21(c)(1)(ii). The staff's finding was based on its determination that the 60-year projected mean RT_{NDT} values for the limiting RPV circumferential and axial welds satisfied the BWRVIP-74-A acceptance criteria at the time the staff performed the LRA review.

It should be noted that the LRA mean RT_{NDT} calculations used RPV weld neutron fluence and CFs that were valid at the time of the LRA review. Accordingly, the licensee's application dated August 17, 2017, for the subject Code alternative, considered that it was necessary to recalculate the limiting axial weld mean RT_{NDT} values using updated neutron fluence values from the updated PTLR dated January 9, 2017. Based on increased neutron fluence values, the limiting axial weld mean RT_{NDT} values increased as well, but remained below the bounding axial weld mean RT_{NDT} values from the BWRVIP-05.

The NRC staff independently confirmed that the limiting axial weld mean RT_{NDT} calculation supporting the axial weld TLAA, as documented in Section 4.2.5 of the LRA and approved by the NRC in NUREG-1944, remains bounding for the subject Code alternative, because the projected axial weld mean RT_{NDT} remained below the bounding axial weld mean RT_{NDT} values from the BWRVIP-05. Therefore, the staff determined that the limiting circumferential and axial welds satisfy the PFM acceptance criteria established in the BWRVIP-74-A for the PEO at CNS.

Accordingly, the staff finds that request for alternative RI5-01, to implement the BWRVIP PFM results in lieu of subject RPV circumferential shell weld examination requirements, will provide an acceptable level of quality and safety, and thus should be authorized pursuant to 10 CFR 50.55a(z)(1).

NRC Staff Evaluation Concerning Neutron Fluence

The projected RPV neutron fluence is an input to the determination of the mean RT_{NDT} value needed to demonstrate an acceptable RPV weld conditional failure probability. The RPV neutron fluence, as determined based on NRC staff-approved calculation methodologies, is the key time-dependent parameter for all RPV integrity analyses that consider neutron embrittlement of the RPV beltline materials. The projected neutron fluence input to the mean RT_{NDT} value, for demonstrating an acceptable RPV weld conditional failure probability at the end of the licensed operating term, is expected to be reflective or bounding of the as-operated core design, including major changes like the implementation of power uprates.

On February 22, 2013, the NRC issued Amendment No. 245, authorizing a revision to Technical Specification 3.4.9, "RCS Pressure and Temperature (P/T) Limits," for 32 EFPY (ADAMS Accession No. ML13032A526). In its SE, the NRC staff states: "The primary staff consideration for acceptability is the fact that RAMA [Radiation Analysis Modeling Application] has been found adherent to RG 1.190, and in particular for calculating vessel fluence values for BWR/4 vessel geometries such as [CNS]...the NRC staff concludes that the neutron fluence calculation supporting the proposed 32 EFPY P-T limits is acceptable," which demonstrates that the fluence method used at CNS, adheres to the guidance contained in RG 1.190. The same fluence methods were reviewed by the NRC staff and found acceptable for use in support of issuing Amendment No. 256, which allowed the licensee to implement administrative control of its P/T limits via a PTLR (ADAMS Accession No. ML16158A022). In its present review, the NRC staff verified that the same fluence methods were used to support the 10 CFR 50.55a relief request, meaning that the methods are NRC-approved and adhere to the guidance contained in RG 1.190. The calculations also reflect past and present operational characteristics, and the fluence projection for future cycles is representative of the most recent operating cycles at CNS, based on consistency with the most recent revision of the CNS PTLR submitted to the NRC on January 9, 2017 (ADAMS Accession Nos. ML17018A151 and ML17018A152). Therefore, the NRC staff finds the use of the two fluence values reported in the Attachment to the letter dated August 17, 2017, acceptable for use as inputs to demonstrate an acceptable RPV weld conditional failure probability at the end of the licensed operating term, based on use of fluence values calculated using an NRC-approved fluence methodology, which were used to derive valid fluence projections, as reported using an NRC-approved PTLR methodology.

4.0 CONCLUSION

The NRC staff finds that the information submitted by the licensee demonstrates that the conditional failure probabilities for the CNS limiting RPV circumferential and axial shell welds at the end of the PEO satisfies the NRC staff's acceptance criteria for these evaluations in its SEs for BWRVIP-05 and BWRVIP-74-A. Additionally, the licensee will continue to implement operator training and procedures to limit the frequency of cold overpressure events in accordance with the staff's SE for the BWRVIP-05 report, consistent with the staff's previous approval of these methods for the PEO, as documented in Section 4.2.5 of NUREG-1944 for the license renewal of CNS. The licensee has therefore satisfied the plant-specific conditions required to obtain NRC authorization for this specific Code alternative.

On this basis, the NRC staff concludes that implementation of the BWRVIP-05 and BWRVIP-74-A methods, in lieu of the specific ASME Code, Section XI, Category B-A, Item No. B1.11 requirements for volumetric examination of the subject RPV circumferential shell welds, will provide an acceptable level of quality and safety at CNS for the duration of the unit's 20-year extended license term. The NRC staff has reviewed the subject request and concludes as set forth above, that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Therefore, pursuant to 10 CFR 50.55a(z)(1), CNS request for alternative RI5-01 is authorized for the remaining term of the CNS renewed operating license, which ends on January 18, 2034.

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

Principal Contributors: J. Jenkins, NRR/DMLR/MVIB
A. Patel, NRR/DSS/SNPB

Date: July 31, 2018

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

ALTERNATIVE REQUEST NO. RI5-02, REVISION 1

FOR THE FIFTH 10-YEAR INTERVAL INSERVICE INSPECTION

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

By letter dated June 9, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15167A066), as supplemented by letters dated October 21, 2015 (ADAMS Accession No. ML15301A249) and December 21, 2015 (ADAMS Accession No. ML15364A013), Nebraska Public Power District (the licensee) submitted proposed alternative Request No. RI5-02 for its fifth 10-year interval inservice inspection (ISI) program plan for its reactor vessel internals (RVI) components at Cooper Nuclear Station (CNS). In this safety evaluation (SE), the term "RVI components" includes reactor pressure vessel interior surfaces, attachments, and core support structures. In Request No. RI5-02, the licensee proposed to use Boiling Water Reactor (BWR) Vessel and Internals Project (BWRVIP) guidelines as an alternative to certain requirements of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) for ISI of the reactor pressure vessel interior surfaces, attachments, and core support structures. These proposed alternatives were requested for the fifth 10-year ISI interval, which began on April 1, 2016, and will end on February 28, 2026. By letter dated February 17, 2016 (ADAMS Accession No. ML16034A479), the U.S. Nuclear Regulatory Commission (NRC) staff authorized the proposed alternative in Request No. RI5-02 pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) paragraph 50.55a(z)(1) on the basis that the alternative provides an acceptable level of quality and safety.

By letter dated August 17, 2017 (ADAMS Accession No. ML17241A048), the licensee submitted proposed alternative Request No. RI5-02, Revision 1, for its fifth 10-year interval ISI program plan for its RVI components at CNS. Request No. RI5-02, Revision 1, updates the specified revision of BWRVIP-18 "BWR Core Spray Internals Inspection and Flaw Guidelines," which was one of the guidelines referenced in Request No. RI5-02. Request No. RI5-02, Revision 1, also updated the inspection history to include the fall 2016 refueling outage.

2.0 REGULATORY EVALUATION

The ISI of ASME Code Class 1, 2, and 3 components is to be performed in accordance with Section XI of the ASME Code and applicable edition and addenda as required by 10 CFR 50.55a(g), "Preservice and inservice inspection requirements," except where specific relief has been granted by the NRC pursuant to 10 CFR 50.55a(g)(6)(i), "Impractical ISI requirements: Granting of relief."

Pursuant to 10 CFR 50.55a(z), "Alternatives to codes and standards requirements," alternatives to the requirements of paragraph (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.

Pursuant to 10 CFR 50.55a(g)(4), "Inservice inspection standards requirements for operating plants," 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, 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 ISI 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)(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).

The regulation in 10 CFR 50.55a(g)(4)(iv), "Applicable ISI Code: Use of subsequent Code editions and addenda," states that inservice examination of components and system pressure tests may meet the requirements set forth in subsequent editions and addenda that are incorporated by reference in paragraph 10 CFR 50.55a(a), subject to the limitations and modifications listed in 10 CFR 50.55a(b) and subject to Commission approval. Portions of editions or addenda may be used provided that all related requirements of the respective editions or addenda are met. The applicable ASME Code of Record for the fifth 10-year ISI interval for CNS, is the ASME Code, Section XI, 2007 Edition through 2008 Addenda.

3.0 TECHNICAL EVALUATION

3.1 The Components for Which an Alternative is Requested

ASME Code, Section XI, Class 1, Examination Categories B-N-1 and B-N-2, Code Item Nos. B13.10 (Vessel Interior), B13.20 (Interior Attachments within Beltline Region), B13.30 (Interior Attachments beyond Beltline Region), and B13.40 (Core Support Structure).

3.2 Examination Requirements for Which an Alternative is Requested

ASME Code, Section XI, requires the visual examination (VT) of certain RVI components. These examinations are included in Table IWB-2500-1, Categories B-N-1 and B-N-2, and identified with the following item numbers:

B13.10 - Examine accessible areas of the reactor vessel interior each period using a technique which meets the requirements for a VT-3 examination, as defined in paragraph IWA-2213 of the ASME Code, Section XI.

B13.20 - Examine interior attachment welds within the beltline region each interval using a technique which meets the requirements for a VT-1 examination, as defined in paragraph IWA-2211 of the ASME Code, Section XI.

B13.30 - Examine interior attachment welds beyond the beltline region each interval using a technique which meets the requirements for a VT-3 examination, as defined in paragraph IWA-2213 of the ASME Code, Section XI.

B13.40 - Examine surfaces of the core support structure each interval using a technique which meets the requirements for a VT-3 examination, as defined in paragraph IWA-2213 of the ASME Code, Section XI.

These examinations are performed to assess the structural integrity of the reactor pressure vessel interior surfaces, attachments, and core support structures.

3.3 Licensee's Basis for Requesting an Alternative and Justification for Granting Relief

In proposed alternative Request No. RI5-02, Revision 1, the licensee, in lieu of ASME Code, Section XI requirements, submitted an alternative inspection program per the BWRVIP guidelines for B-N-1 and B-N-2 reactor pressure vessel interior surfaces, attachments, and core support structures at CNS. The licensee stated that implementation of the alternative inspection program will maintain an acceptable level of quality and safety and will avoid duplicate or unnecessary inspections, while conserving radiological dose. The licensee further indicated that the BWRVIP has established reporting protocol for examination results and deviations, and that the NRC has agreed with the BWRVIP approach in principle and has issued SEs for many of these guidelines.

The licensee proposed to examine the CNS RVI components in accordance with the following BWRVIP guidelines:

- BWRVIP-03, Revision 17, "BWR Vessel and Internals Project Reactor Pressure Vessel and Internals Examination Guidelines"
- BWRVIP-18, Revision 2-A, "BWR Vessel and Internals Project, BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines"
- BWRVIP-25, "BWR Vessel and Internals Project, BWR Core Plate Inspection and Flaw Evaluation Guidelines"
- BWRVIP-26-A, "BWR Vessel and Internals Project, BWR Top Guide Inspection and Flaw Evaluation Guidelines"
- BWRVIP-27-A, "BWR Vessel and Internals Project, BWR Standby Liquid Control System/Core Plate ΔP Inspection and Flaw Evaluation Guidelines"
- BWRVIP-38, "BWR Vessel and Internals Project, BWR Shroud Support Inspection and Flaw Evaluation Guidelines"

- BWRVIP-41, Revision 3, "BWR Vessel and Internals Project, BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines"
- BWRVIP-47-A, "BWR Vessel and Internals Project, BWR Lower Plenum Inspection and Flaw Evaluation Guidelines"
- BWRVIP-48-A, "Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines"
- BWRVIP-76, Revision 1-A, "BWR Vessel and Internals Project, BWR Core Shroud Inspection and Flaw Evaluation Guidelines"
- BWRVIP-94NP, Revision 2, "BWR Vessel and Internals Project, Program Implementation Guide"
- BWRVIP-100-A, "BWR Vessel and Internals Project, Updated Assessment of the Fracture Toughness of Irradiated Stainless Steel for BWR Core Shrouds"
- BWRVIP-138, Revision 1-A, "BWR Vessel and Internals Project, Updated Jet Pump Beam Inspection and Flaw Evaluation"

With the exception of BWRVIP-18, Revision 2-A, the above BWRVIP guidelines were also referenced in the evaluation of proposed alternative Request No. RI5-02, which was approved by the NRC staff on February 17, 2016.

In Table 1 of proposed alternative Request No. RI5-02, Revision 1, the licensee provided a comparison of the ASME Code, Section XI, examination requirements for B-N-1 and B-N-2 categories of the reactor pressure vessel interior surfaces, attachments, and core support structures with the above current BWRVIP Inspection and Evaluation Guidelines. In proposed alternative Request No. RI5-02, Revision 1, the licensee also provided additional information regarding the BWRVIP inspection guidelines for the following components of the reactor pressure vessel interior surfaces, attachments, and core support structures and their subcomponents representing each of the ASME Code, Section XI, Item Nos. B13.10, B13.20, B13.30, and B13.40:

- Reactor Vessel Interior (B13.10)
- Interior Attachments within Beltline (B13.20)
- Interior Attachments beyond Beltline (B13.30)
- Core Support Structure (B13.40)

The comparison provided by the licensee in proposed alternative Request No. RI5-02, Revision 1, is identical with the comparison provided by the licensee in proposed alternative Request No. RI5-02, except that the examination of core spray piping and spargers will be in accordance with BWRVIP-18, Revision 2-A in lieu of BWRVIP-18, Revision 1-A.

3.4 NRC Staff Evaluation

The NRC staff reviewed the information provided by the licensee in its submittal dated August 17, 2017, regarding its proposed alternatives to the ASME Code, Section XI, ISI requirements and the technical bases for the licensee's proposed alternatives. The staff found the referenced BWRVIP reports to be acceptable, with any additional conditions associated with the implementation of the subject BWRVIP reports outlined in the corresponding staff SE for that report.

Examination of Reactor Vessel Interior (Item B13.10)

The ASME Code requires a VT-3 examination of the reactor vessel interior, which is above and below the core beltline, and which is made accessible during normal refueling outages. For the first 10-year inspection interval, the ASME Code requires inspection at the first refueling outage and subsequent refueling outages at approximately 3-year intervals. For the second and successive 10-year inspection intervals, the ASME Code requires inspection once each inspection period.

Except for the core spray piping and spargers, the BWRVIP alternatives proposed by the licensee for Item B13.10 components in its submittal dated August 17, 2017, are identical with the alternatives approved by the NRC staff in its SE dated February 17, 2016, of proposed alternative Request No. RI5-02. For the core spray piping and spargers, the licensee proposes examination in accordance with BWRVIP-18, Revision 2-A. By letter dated May 9, 2012 (ADAMS Accession No. ML12139A153), the Electric Power Research Institute (EPRI) submitted BWRVIP-18, Revision 2, "BWR Vessel and Internals Project, BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines" to the NRC staff for review. The changes in this revision included a revised operating experience and susceptibility discussions for core spray internals, a revised inspection program for core spray internals, and additional guidance for evaluation of cracking associated with sparger bracket locations. By letter dated February 22, 2016 (ADAMS Accession No. ML16011A199), the NRC staff found that BWRVIP-18, Revision 2 is acceptable for referencing in licensing applications for nuclear power plants. The NRC staff finds that the licensee's proposal in alternative Request No. RI5-02, Revision 1, to use BWRVIP-18, Revision 2-A, for the examination of core spray piping and spargers is acceptable.

The last examination of core spray piping, which occurred during the fall 2016 outage, did not reveal any changes in the existing indications and identified no new indications.

The NRC staff finds that the licensee's proposed alternative Request No. RI5-02, Revision 1, provides an acceptable level of quality and safety for the Item B13.10 components.

Examination of Interior Attachments within Beltline (Item B13.20)

The ASME Code requires a VT-1 examination of accessible reactor vessel interior attachment welds within the beltline during each inspection interval. The BWRVIP alternatives proposed by the licensee for Item B13.20 components in its submittal dated August 17, 2017, are identical with the alternatives approved by the NRC staff in its SE dated February 17, 2016, of proposed alternative Request No. RI5-02.

The NRC staff finds that the licensee's proposed alternative Request No. RI5-02, Revision 1, provides an acceptable level of quality and safety for the Item B13.20 components.

Examination of Interior Attachments beyond Beltline (Item B13.30)

The ASME Code requires a VT-3 examination of accessible reactor vessel interior attachment welds beyond the beltline during each inspection interval. The BWRVIP alternatives proposed by the licensee for Item B13.30 components in its submittal dated August 17, 2017, are identical with the alternatives approved by the NRC staff in its SE dated February 17, 2016, of proposed alternative Request No. RI5-02.

The NRC staff finds that the licensee proposed alternative Request No. RI5-02, Revision 1, provides an acceptable level of quality and safety for the Item B13.30 components.

Examination of Core Support Structure (Item B13.40)

The ASME Code requires a VT-3 examination of accessible surfaces of the core support structure during each inspection interval.

The BWRVIP alternatives proposed by the licensee for Item B13.40 components in its submittal dated August 17, 2017, are identical with the alternatives approved by the NRC staff in its SE dated February 17, 2016, of proposed alternative Request No. RI5-02.

The NRC staff finds that the licensee's proposed alternative Request No. RI5-02, Revision 1, provides an acceptable level of quality and safety for the Item B13.40 components.

4.0 CONCLUSION

Based on the information provided in the licensee's submittal dated August 17, 2017, the NRC staff concludes that the alternative proposed by the licensee will ensure that the integrity of the RVI components is maintained with an acceptable level of quality and safety.

The NRC staff has reviewed the subject request and concludes as set forth above, that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Therefore, pursuant to 10 CFR 50.55a(z)(1), the licensee's proposed alternative for CNS is authorized for the fifth 10-year ISI interval with the condition that in the event the licensee wishes to take exceptions to, or deviations from, the NRC-approved BWRVIP inspection guidelines authorized as a proposed alternative, the licensee must revise and resubmit its request for authorization to use the proposed alternative under 10 CFR 50.55a.

Any ASME Code, Section XI, RVI components that are not included in this request for alternative will continue to be inspected in accordance with the ASME Code, Section XI requirements. The Inspection and Evaluation guidelines addressed in the relevant BWRVIP reports should be implemented for the non-ASME Code, Section XI, RVI components at CNS.

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

Principal Contributor: J. Jenkins, NRR/DMLR/MVIB

Date: July 31, 2018

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

ALTERNATIVE REQUEST NO. RI5-03 FOR THE

FIFTH AND SIXTH 10-YEAR INTERVAL INSERVICE INSPECTIONS

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

By letter dated August 17, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17241A048), as supplemented by letters dated March 14, 2018, and April 26, 2018 (ADAMS Accession Nos. ML18082A563 and ML18131A159 respectively), Nebraska Public Power District (the licensee), requested relief from the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section XI, Table IWB-2500-1 and instead proposes to use the inspection requirements documented in ASME Code Case N-702, "Alternative Requirements for Boiling Water Reactor (BWR) Inner Nozzle Radius and Nozzle-to-Shell Weld, Section XI, Division 1." For the VT-1 visual examinations allowed by ASME Code Case N-702, the licensee proposes to use ASME Code Case N-648-1, "Alternative Requirements for Inner Radius Examination of Class 1 Reactor Vessel Nozzles, Section XI, Division 1," with associated required conditions specified in Regulatory Guide (RG) 1.147, Revision 17, "Inservice Inspection Code Case Acceptability, Section XI, Division 1," dated August 2014.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) paragraph 50.55a(z)(1), the licensee requested to use the proposed alternative on the basis that the alternative provides an acceptable level of quality and safety.

2.0 REGULATORY EVALUATION

Adherence to Section XI of the ASME Code is mandated by 10 CFR 50.55a(g)(4), "Inservice inspection standards requirements for operating plants," which states, in part, that ASME Code Class 1, 2, and 3 components will meet the requirements, except the design and access provisions and the pre-service examination requirements, set forth in Section XI of the ASME Code.

Enclosure 3

The regulation at 10 CFR 50.55a(z), "Alternatives to codes and standards requirements," states, in part that,

> Alternatives to the requirements of paragraphs (b) through (h) of [10 CFR 50.55a] or portions thereof 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 applicant or licensee must demonstrate that:
>
> (1) *Acceptable level of quality and safety.* The proposed alternative would provide an acceptable level of quality and safety; or
>
> (2) *Hardship without a compensating increase in quality and safety.* 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 U.S. Nuclear Regulatory Commission (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

3.1 Background

For all reactor pressure vessel (RPV) nozzle-to-vessel shell welds and nozzle inner radii, ASME Code, Section XI, requires 100 percent inspection during each 10-year inservice inspection (ISI) interval. However, ASME Code Case N-702 provides an alternative, which reduces the inspection of RPV nozzle-to-vessel shell welds and nozzle inner radii areas from 100 percent to 25 percent of the nozzles for each nozzle type during each 10-year interval. This code case was conditionally approved in RG 1.147, Revision 17. For application of ASME Code Case N-702, the licensee is required to address the conditions specified in RG 1.147, Revision 17 for ASME Code Case N-702. The condition specified in RG 1.147, Revision 17 states, in part:

> The applicability of Code Case N-702 must be shown by demonstrating that the criteria in Section 5.0 of NRC Safety Evaluation regarding [Boiling Water Reactor Vessel and Internals Project] BWRVIP-108 dated December 19, 2007 ([ADAMS Accession No.] ML073600374) or Section 5.0 of NRC Safety Evaluation regarding BWRVIP-241 dated April 19, 2013 ([ADAMS Accession No.] ML13071A240) are met. The evaluation demonstrating the applicability of the Code Case shall be reviewed and approved by the NRC prior to the application of the Code Case.

BWRVIP-108, "BWR Vessel and Internals Project, Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Inner Radii" (Not publicly available; proprietary information) and BWRVIP-241, "BWR Vessel and Internals Project, Probabilistic Fracture Mechanics [PFM] Evaluation for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii" (ADAMS Accession No. ML11119A043) contain PFM analysis results supporting ASME Code Case N-702. Both reports are for 40 years of operation. BWRVIP-241 contains additional PFM results supporting revision of the evaluation criteria under "Conditions and Limitations" in the safety evaluation (SE) for BWRVIP-108. The SE for BWRVIP-241 accepted the revised criteria.

Recently, the NRC issued a safety evaluation (SE) dated April 26, 2017 (ADAMS Accession No. ML17114A096), on a supplemental document for license renewal (BWRVIP-241, Appendix A, "BWR Nozzle Radii and Nozzle-to-Vessel Welds Demonstration of Compliance with the Technical Information Requirements of the License Renewal Rule (10 CFR 54.21")). This license renewal Appendix A extends the applicability of the BWRVIP-108 and BWRVIP-241 methodologies, and, therefore, ASME Code Case N-702, through the period of extended operation.

ASME Code Case N-702 allows that VT-1 visual examination may be performed in lieu of volumetric examination for Examination Item No. B3.100 nozzle inner radius sections. ASME Code Case N-648-1, as conditionally accepted by RG 1.147, Revision 17, requires that nozzle inner radius examinations must use the allowable flaw length criteria of ASME Code, Table IWB-3512-1, with limiting assumptions on the flaw aspect ratio.

3.2 ASME Code Components Affected

The affected components belong to Examination Category B-D, "Full Penetration Welded Nozzles in Vessels" under Examination Item No. B3.90, "Nozzle-to-Vessel Welds" and B3.100, "Nozzle Inside Radius Section."

Table 1
RPV Nozzle-to-Vessel Welds and Inner Radii Subject to this Request

Identification Number	Description	Total Number	Minimum Number to be examined
N1	Recirculation Outlet	2	1
N2	Recirculation Inlet	10	3
N3	Main Steam Outlet	4	1
N5	Core Spray	2	1
N6	Head Spray	2	1
N7	Head Vent	1	1
N8	Jet Pump Instrumentation	2	1

3.3 Applicable Code Edition and Addenda

This request applies to the fifth and sixth 10-year ISI intervals, in which CNS adopted the 2007 Edition through 2008 Addenda of ASME Code Section XI, as the Code of Record.

3.4 Applicable Code Requirements

ASME Code Section XI, Table IWB-2500-1, Examination Category B-D, requires a volumetric examination of all nozzles with full penetration welds to the vessel shell (or head) and integrally cast nozzles each 10-year interval.

3.5 Licensee's Proposed Alternative

The licensee proposed to implement ASME Code Case N-702 and reduce the ASME Code-required volumetric examinations for all RPV nozzle-to-shell welds and inner radii, to a minimum of 25 percent of the nozzle inner radii and nozzle-to-shell welds, including at least one nozzle from each system and nominal pipe size during each inspection interval. The required examination volume for the reduced set of nozzles remains at 100 percent of that depicted in Figures IWB-2500-7 (a) through (d), as applicable in the ASME Code.

In addition, the licensee stated it may perform a VT-1 visual examination in lieu of a volumetric examination for Category B-D, Item No. 3.100 consistent with ASME Code Case N-648-1, with associated required conditions specified in RG 1.147, Revision 17.

3.6 Licensee's Bases for Alternative

The alternative is based on the PFM results documented in the BWRVIP-241 report. The licensee proposed that it met the evaluation criteria in the SE for BWRVIP-241 as follows:

(1) Maximum RPV Heatup/Cooldown Rate

The maximum RPV heatup/cooldown rate is limited to less than 115 °F/hr (degrees Fahrenheit per hour). CNS Technical Specifications Surveillance Requirement 3.4.9.1, Reactor Coolant System heatup and cooldown rates are limited to a maximum of 100 °F when averaged over any one hour period and thus meets the requirement of Condition 1.

(2) Recirculation Inlet (N2) Nozzles

$(pr/t) / C_{i\text{-}RPV} < 1.15$, where

p = RPV normal operating pressure (per square inch (psi)),
r = RPV inner radius (inch),
t = RPV wall thickness (inch), and
Ci-RPV = 19332.

The CNS result based on the input parameters for this nozzle, per the licensee's submittal, is $(pr/t) / C_{i\text{-}RPV} = 0.85$ ([(1020)(110.4)/6.875]/19332), thus meeting the requirements of Condition 2.

(3) Recirculation Inlet (N2) Nozzles

$[p(r_o^2+r_i^2)/(r_o^2-r_i^2)]/C_{i\text{-}NOZZLE} \leq 1.47$, where

p = RPV normal operating pressure (psi),
r_o = nozzle outer radius (inch),
r_i = nozzle inner radius (inch), and
Ci-NOZZLE = 1637.

The CNS result based on the input parameters for this nozzle, per the licensee's submittal, is $[p(r_o^2+r_i^2)/(r_o^2-r_i^2)]/C_{i\text{-}NOZZLE} = 1.34$ $([1020(10.22^2 + 6.188^2)/(10.22^2 - 6.188^2)]/1637)$, thus meeting the requirements of Condition 3.

(4) Recirculation Outlet (N1) Nozzles

$(pr/t)/C_{o\text{-}RPV} \leq 1.15$, where

p = RPV normal operating pressure (psi),
r = RPV inner radius (inch),
t = RPV wall thickness (inch), and
Co-RPV = 16171.

The CNS result based on the input parameters for this nozzle, per the licensee's submittal, is $(pr/t)/C_{o\text{-}RPV}$ 1.013 ([(1020)(110.4)/6.875]/16171), thus meeting the requirements of Condition 4.

(5) Recirculation Outlet (N1) Nozzles

$[p(r_o^2 + r_i^2)/(r_o^2 - r_i^2)]$/Co-NOZZLE ≤ 1.59, where

p = RPV normal operating pressure (psi),
r_o = nozzle outer radius (inch),
r_i = nozzle inner radius (inch), and
Co-NOZZLE = 1977.

The CNS result based on the input parameters for this nozzle, per the licensee's submittal, is $[p(r_o^2 + r_i^2)/(r_o^2 - r_i^2)]$/Co-NOZZLE = 1.08 ($[1020(21.656^2 + 12.875^2)/(21.656^2 - 12.875^2)]/1977$), thus meeting the requirements of Condition 5.

The licensee's application also states that the licensee performed a plant-specific PFM analysis to supplement the criteria of ASME Code Case N-702 and BWRVIP-241 in order to demonstrate that the probability of failure (PoF) remains acceptable over the period of extended operation. Conservatively, assuming zero inspection for the initial 40 years of operation and examination of 25 percent of the nozzles every interval for the period of extended operation, the evaluation concluded the average PoF for a low temperature overpressure (LTOP) event is 1.675×10^{-10} per year for the nozzle inner radius, and $< 8.33 \times 10^{-13}$ per year for the nozzle-to-shell weld, both of which are less than the NRC safety goal of 5×10^{-6} per year. These probabilities were calculated based on the most limiting nozzle (the N2 inlet nozzle) and thus are bounding for all nozzles which are part of the licensee's request.

3.7 Duration of Proposed Alternative

The fifth 10-year ISI interval for CNS began on April 1, 2016, and the sixth 10-year ISI interval is scheduled to end concurrent with the end of the extended license period on January 18, 2034.

3.8 NRC Staff Evaluation

The licensee proposed an alternative to implement ASME Code Case N-702 for all CNS RPV nozzle-to-vessel shell penetration welds and nozzle inner radii using the criteria in BWRVIP-241.

In general, the applicability of the BWRVIP-241 report to an ASME Code Case N-702 alternative is demonstrated by showing that Criteria 2 through 5 within Section 5.0 of the NRC SE for

BWRVIP-241 are met for the bounding nozzles (recirculation inlet and outlet nozzles), and that Criterion 1 is met for all components included in the proposed alternative.

The NRC staff confirms that Criterion 1 (applicable to all nozzles within the scope of ASME Code Case N-702) is satisfied because CNS Technical Specifications Surveillance Requirement 3.4.9.1 limits the maximum heatup/cooldown rate to less than or equal to 100 °F/hour, well below the 115 °F/hour criterion limit.

For Criteria 2 through 5, the licensee provided plant specific data and its evaluation of the driving force factors, or ratios, using the criteria established in Section 5.0 of the SE for BWRVIP-241. The NRC staff reviewed the licensee's calculations and confirms that they show that Criteria 2 through 5 are satisfied. Therefore, the BWRVIP-241 report applies to CNS, and the basis for using ASME Code Case N-702 is demonstrated for the CNS RPV nozzle-to-vessel welds and inner radii listed in Table 1 above.

The licensee performed a plant-specific PFM analysis to supplement the criteria of ASME Code Case N-702 and BWRVIP-241 in order to demonstrate that the PoF remains acceptable over the period of extended operation. Conservatively, assuming zero inspection for the initial 40 years of operation and examination of 25 percent of the nozzles every interval for the period of extended operation, the evaluation concluded the average PoF for a low temperature overpressure event is 1.675×10^{-10} per year for the nozzle inner radius, and $<8.33 \times 10^{-13}$ per year for the nozzle-to-shell weld. The licensee's evaluation also provided the PoF due to normal operation, or $<8.33 \times 10^{-10}$ per year. The NRC staff finds the licensee's evaluation acceptable since the NRC staff has reviewed the PFM analysis provided and has determined that the PoF due to either LTOP or normal operation is less than the NRC safety goal of 5×10^{-6} per year.

For the Examination Item No. B3.100 nozzle inner radius sections, the NRC staff finds the licensee's proposal to perform VT-1 visual examination in lieu of ultrasonic examination to be acceptable because the licensee will comply with ASME Code Case N-648-1, with associated required conditions specified in RG 1.147, Revision 17.

4.0 CONCLUSION

As set forth above, the NRC staff determines that the licensee has demonstrated that the proposed alternative provides an acceptable 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)(1). Therefore, the NRC staff authorizes the use of RI5-03 at CNS for the fifth and sixth intervals for ASME Category B-D, Item Numbers B3.90 and B3.100 until January 18, 2034.

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

Principal Contributor: J. Jenkins, NRR/DMLR/MVIB

Date: July 31, 2018

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOR THE FIFTH 10-YEAR INSERVICE INSPECTION INTERVAL

RELIEF REQUEST NO. RR5-02

PROPOSED ALTERNATIVE TO UTILIZE ASME CODE CASE N-513-4

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO 50-298

1.0 INTRODUCTION

By letter dated August 17, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17241A048), as supplemented by letter dated March 8, 2018 (ADAMS Accession No. ML18078A264), Nebraska Public Power District (NPPD the licensee), submitted a request in accordance with paragraph 50.55a(z)(2) of Title 10 of the *Code of Federal Regulations* (10 CFR) for a proposed alternative to the requirements of Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) for the Cooper Nuclear Station (CNS). The proposed alternative, Relief Request RR5-02, would allow the licensee to use ASME Code Case N-513-4, "Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Class 2 or 3 Piping Section XI, Division 1," for the evaluation and temporary acceptance of flaws in moderate energy Class 2 and 3 piping in lieu of specified ASME Code requirements for the fifth 10-year inservice inspection (ISI) Interval which began on April 1, 2016, and is scheduled to end on February 28, 2026.

Specifically, pursuant to 10 CFR 50.55a(z)(2), the licensee requested to use the alternative 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.

2.0 REGULATORY EVALUATION

The licensee proposes an alternative to the requirement of ASME Code, Section XI, Articles IWC-3000 and IWD-3000.

Adherence to Section XI of the ASME Code is mandated by 10 CFR 50.55a(g)(4), "Inservice inspection standards requirements for operating plants," which states, in part, that ASME Code Class 1, 2, and 3 components (including supports) will meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in Section XI of the ASME Code.

The regulation 10 CFR 50.55a(z), "Alternatives to codes and standards requirements," states, in part, that alternatives to the requirements of paragraph (g) of 10 CFR 50.55a may be used when authorized by the 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.

Based on the above, and subject to the following technical evaluation, the U.S. Nuclear Regulatory Commission (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

3.1 ASME Code Component(s) Affected

The affected components are ASME Code Class 2 and 3 moderate energy piping systems, as described in ASME Code Case N-513-4, Section 1 "Scope," whose maximum operating temperature does not exceed 200 degrees Fahrenheit (°F) and whose operating pressure does not exceed 275 pounds per square inch gauge (psig).

3.2 Applicable Code Edition and Addenda

The Code of Record for the fifth 10-year ISI interval at CNS is the ASME Code, Section XI, 2007 Edition, through 2008 Addenda. The fifth 10-year ISI at CNS began on April 1, 2016, and is scheduled to end on February 28, 2026.

3.3 Applicable Code Requirement

ASME Code, Section XI, Subarticles IWC-3120 and IWC-3130, require that flaws exceeding the defined acceptance criteria be corrected by repair/replacement activities or evaluated and accepted by analytical evaluation. ASME Code, Section XI, subparagraph IWD-3120(b), requires that components exceeding the acceptance standards of Article IWD-3400 be subject to supplemental examination, or to a repair/replacement activity.

3.4 Reason for Request

The licensee stated that ASME Code Case N-513-3 does not allow evaluation of flaws located away from attaching circumferential piping welds that are in elbows, bent pipe, reducers, expanders, and branch tees. ASME Code Case N-513-3 does not allow evaluation of flaws located in heat exchanger external tubing or piping. ASME Code Case N-513-4 provides guidance for evaluation of flaws in these locations. Moderately degraded piping could require a plant shutdown within the required action statement timeframes to repair observed degradation. The licensee stated, in part, in its letter dated August 17, 2017, that "[p]lant shutdown activities result in additional dose and plant risk that would be inappropriate when a degraded condition is demonstrated to retain adequate margin to complete the component's function."

3.5 Licensee's Proposed Alternative and Basis for Use

The licensee's proposed alternative is to use ASME Code Case N-513-4 for the evaluation and temporary acceptance of flaws in moderate energy Class 2 and 3 piping in lieu of specified ASME Code, Section XI, requirements. In addition, the licensee's proposed alternative includes the determination of an allowable leakage rate by dividing the critical leakage rate by a safety factor of four.

The licensee stated that it will apply ASME Code Case N-513-4 in its entirety for the evaluation of Class 2 and 3 piping flaws at CNS if Code repairs cannot reasonably be completed within the technical specification required time limit.

The licensee stated that limitations in ASME Code Case N-513-3, related to its use on piping components such as elbows, bent pipe, reducers, expanders, and branch tees and external tubing or piping attached to heat exchangers, have been addressed in ASME Code Case N-513-4. The licensee provided a high level overview of the differences between Code Case N-513-3 and Code Case N-513-4 as listed below:

1. Revised the maximum allowable time of use from no longer than 26 months to the next refueling outage.

2. Added applicability to piping elbows, bent pipe, reducers, expanders, and branch tees where the flaw is located more than $(R_0t)^{1/2}$ [where R_0 is the outside pipe radius and t is the evaluation wall thickness] from the centerline of the attaching circumferential piping weld.

3. Expanded use to external tubing or piping attached to heat exchangers.

4. Revised to limit the use to liquid systems.

5. Revised to clarify treatment of Service Level load combinations.

6. Revised to address treatment of flaws in austenitic pipe flux welds.

7. Revised to require minimum wall thickness acceptance criteria to consider longitudinal stress in addition to hoop stress.

8. Other minor editorial changes to improve the clarity of the Code Case.

Enclosure 1, Attachment 2 of the licensee's letter dated August 17, 2017, includes a technical basis document for the fourth revision to N-513, "Proceedings of the ASME 2014 Pressure Vessels & Piping Conference, PVP2014, July 20-24, 2014, Anaheim, California, USA, PVP2014-28355, Technical Basis for Proposed Fourth Revision to ASME Code Case N-513."

The licensee stated that the effects of leakage may impact the operability determination or the plant flooding analyses specified in paragraph 1(f) of ASME Code Case N-513-4. For a leaking flaw, the licensee stated that the allowable leakage rate will be determined by dividing the critical leakage rate by a safety factor of four. The critical leakage rate is determined as the limiting leakage rate that can be tolerated and may be based on the allowable loss of inventory or the maximum leakage that can be tolerated relative to room flooding, among others. The licensee contends that applying a safety factor of four to the critical leakage rate provides quantitative measurable limits, which ensure the operability of the system and early identification of issues that could erode defense-in-depth and lead to adverse consequences.

The licensee stated that ASME Code Case N-513-4 utilizes technical evaluation approaches that are based on principles that are accepted in other Code documents already acceptable to the NRC. The licensee also stated that application of this code case, in concert with safety factors on leakage limits, will maintain acceptable structural and leakage integrity while minimizing plant risk and personnel exposure by minimizing the number of plant transients that could be incurred if degradation is required to be repaired based on ASME Code, Section XI, acceptance criteria only.

3.6 Hardship Justification

As stated, in part, by the licensee in its letter dated August 17, 2017,

> Moderately degraded piping could require a plant shutdown within the required action statement timeframes to repair observed degradation. Plant shutdown activities result in additional dose and plant risk that would be inappropriate when a degraded condition is demonstrated to retain adequate margin to complete the component's function. The use of an acceptable alternative analysis method in lieu of immediate action for a degraded condition will allow NPPD to perform additional extent of condition examinations on the affected systems while allowing time for safe and orderly long term repair actions if necessary. Actions to remove degraded piping from service could have a detrimental overall risk impact by requiring a plant shutdown, thus requiring use of a system that is in standby during normal operation. Accordingly, compliance with the current code requirements results in a hardship without a compensating increase in the level of quality and safety.

3.7 Duration of Proposed Alternative

The licensee stated that the duration of the proposed alternative is the fifth 10-year ISI interval which began on April 1, 2016, and is scheduled to end on February 28, 2026. The licensee stated that if a flaw is evaluated near the end of the interval, and the next refueling outage is in the subsequent interval, the flaw may remain in service until the next refueling outage.

3.8 NRC Staff Evaluation

The NRC staff evaluated the adequacy of the proposed alternative in maintaining the structural integrity of piping components identified in ASME Code Case N-513-4. ASME Code Case N-513-3, which is conditionally approved for use in RG 1.147, Revision 17, provides alternative evaluation criteria for temporary acceptance of flaws, including through-wall flaws, in moderate energy Class 2 and 3 piping. However, ASME Code Case N-513-3 contains limitations that the licensee considers restrictive and could result in an unnecessary plant shutdown. ASME Code Case N-513-3 is limited to straight pipe with provisions for flaws that extend for a short distance, at the pipe to fitting weld, into the fitting. Evaluation criteria for flaws in elbows, bent pipe, reducers, expanders, branch tees and heat exchangers are not included within the scope of ASME Code Case N-513-3. Code Case N-513-4 addresses these aforementioned limitations. Given that the previous revision of this code case (Code Case N-513-3) is conditionally approved for use in RG 1.147, Revision 17, the staff focused its review on the differences between ASME Code Cases N-513-3 and N-513-4. The significant changes in ASME Code Case N-513-4 include: (1) revised temporary acceptance period; (2) added flaw evaluation criteria for elbows, bent pipe, reducers/expanders and branch tees, (3) expanded applicability to heat exchanger tubing or piping, (4) limited use to liquid systems,

(5) clarified treatment of service load combinations, (6) revised treatment of flaws in austenitic pipe flux welds, (7) revised minimum wall thickness acceptance criteria to consider longitudinal stress in addition to hoop stress, and (8) revised leakage monitoring requirements. The NRC staff also evaluated the licensee's proposed limitation on the leakage rate and its hardship justification.

The NRC staff notes that many requirements specified in ASME Code Case N-513-4 are not discussed in this SE, but they should not be considered as less important. As part of the NRC-approved proposed alternative, all requirements in the code case must be followed. Any exceptions or restrictions to the code case that are approved in this SE also need to be followed.

Temporary Acceptance Period

ASME Code Case N-513-3 specifies a temporary acceptance period of a maximum of 26 months. Code Case N-513-3 is accepted for use in RG 1.147, Revision 17, with the following condition, "The repair or replacement activity temporarily deferred under the provisions of this Code Case shall be performed during the next scheduled outage." ASME Code Case N-513-4 includes wording that limits the use of the code case to the next refueling outage. The NRC staff finds that Code Case N-513-4 appropriately addresses the NRC condition on Code Case N-513-3, and, is therefore, acceptable.

Flaw Evaluation Criteria for Elbows, Bent Pipe, Reducers/Expanders and Branch Tees.

Evaluation and acceptance criteria have been added to ASME Code Case N-513-4 for flaws in elbows, bent pipe, reducers, expanders, and branch tees using a simplified approach, which is based on the Second International Piping Integrity Research Group (IPIRG-2) program reported in NUREG/CR-6444 BMI-2192, "Fracture Behavior of Circumferentially Surface-Cracked Elbows," March 1996.

The flaw evaluation methodology in ASME Code Case N-513-4 for piping elbows, bends, reducers and tees, is conducted as if the flaws in these components are in straight pipe by scaling hoop and axial stresses using ASME Code piping design code stress indices and stress intensification factors to account for the stress variations caused by the geometric differences. Equations used in the code case are consistent with the piping design by rule approach in ASME Code, Section III, NC/ND-3600. NUREG/CR-6444 shows that this approach is conservative for calculating stresses used in flaw evaluations in piping elbows and bent pipe. The code case also applies this methodology to reducers, expanders, and branch tees.

The NRC staff finds that the flaw evaluation and acceptance criteria in ASME Code Case N-513-4 for elbows, bent pipe, reducers, expanders, and branch tees is acceptable because the flaw evaluation methods in the code case are consistent with ASME Code, Section XI, ASME Code Section III, design by rule approach and provides a conservative approach as confirmed by comparing the failure moments predicted using this approach to the measured failure moments from the elbow tests for through-wall circumferential flaws conducted as part of the IPIRG-2 program.

Flaw Evaluation in Heat Exchanger Tubing or Piping

ASME Code Case N-513-4 has been revised to include heat exchanger external tubing or piping provided that the flaw is characterized in accordance with Section 2(a) of the code case and leakage is monitored. Section 2(a) requires that the flaw geometry be characterized by volumetric inspection or physical measurement.

The NRC staff determined that the flaw evaluation criteria in ASME Code Case N-513-4 for straight or bent piping, as appropriate, can be applied to heat exchanger external tubing or piping. The staff determined the methods for evaluating flaws in straight pipe are acceptable since they are currently allowed in ASME Code Case N-513-3. For bent pipe, the acceptability is described in Section 3.2.2 above. Therefore, the NRC staff finds inclusion of heat exchanger external tubing or piping in the code case to be acceptable because only heat exchanger tubing flaws that are accessible for characterization and leakage monitoring may be evaluated in accordance with the code case and the code case provides acceptable methods for the evaluation of flaws.

Limit Use to Liquid Systems

Use of ASME Code Case N-513-4 is specifically limited to liquid systems. The NRC staff finds this change acceptable since ASME Code Case N-513 is not intended to apply to air or other compressible fluid systems.

Treatment of Service Load Combinations

Modifications in ASME Code Case N-513-4 now make clear that all service load combinations must be considered in flaw evaluations to determine the most limiting condition, although previously implied in ASME Code Cases N-513-3 (N-513-4 makes this requirement clear). Therefore, the NRC staff finds this change acceptable.

Treatment of Flaws in Austenitic Pipe Flux Welds

Paragraph 3.1(b) of ASME Code Case N-513-4 contains modifications which include a reference to ASME Code Section XI, Appendix C, C-6320, to address flaws in austenitic stainless steel pipe flux welds. The ASME Code, Section XI, Appendix C, C-6000 permits the use of elastic plastic fracture mechanics criteria in lieu of limit load criteria to analyze flaws in stainless steel pipe flux welds. Equation 1 of the code case was also revised to be consistent with ASME Code, Section XI, Appendix C, C-6320, so the equation can be used for flaws in austenitic stainless steel pipe flux welds. The NRC staff finds this acceptable because the modification to the code case now includes the appropriate methods for the evaluation of stainless steel pipe flux welds in accordance with ASME Code, Section XI.

Minimum Wall Thickness Acceptance Criteria to Consider Longitudinal Stress

Although it is unlikely that a minimum wall thickness calculated based on the longitudinal stress would be limiting when compared to a minimum wall thickness calculated based on hoop stress, ASME Code Case N-513-4 includes revisions that require consideration of longitudinal stress in the calculation of minimum wall thickness. Previous versions of the code case only required the use of hoop stress. The NRC staff finds this acceptable because it will ensure that the more limiting of the longitudinal or hoop stress is used to determine minimum wall thickness.

Leakage Monitoring for Through-Wall Flaws

ASME Code Case N-513-3 required through-wall leakage to be observed by daily walkdowns to confirm the analysis conditions used in the evaluation remain valid. ASME Code Case N-513-4 modifies this requirement by continuing to require that leakage be monitored daily but now allows other techniques to be used to monitor leakage such as using visual equipment or leakage detection systems to determine if leakage rates are changing. The NRC staff finds this change acceptable because the code case continues to require through-wall leaks to be monitored daily and the expanded allowable monitoring methods should have no adverse impact.

Leakage Rate

ASME Code Case N-513-3, paragraph 1(d), states, "The provisions of this Case demonstrate the integrity of the item and not the consequences of leakage. It is the responsibility of the Owner to demonstrate system operability considering effects of leakage." ASME Code Case N-513-4 modified the last sentence, now located in paragraph (f), to state, "It is the responsibility of the Owner to consider effects of leakage in demonstrating system operability and performing plant flooding analyses."

The licensee stated that the allowable leakage rate will be determined by dividing the critical leakage rate by a safety factor of four. The critical leakage rate is determined as the limiting leakage rate that can be tolerated and may be based on the allowable loss of inventory or the maximum leakage that can be tolerated relative to room flooding, among others. The licensee contends that applying a safety factor of four to the critical leakage rate, provides quantitative measurable limits which ensure the operability of the system and early identification of issues that could erode defense-in-depth and lead to adverse consequences.

ASME Code Cases N-513-3 and N-513-4 do not contain leakage limits for components with through-wall flaws. The NRC staff finds that the licensee's approach of applying a safety factor of four to the critical leakage rate is acceptable because it will provide sufficient time for corrective measures to be taken before significant increases in leakage erodes defense-in-depth which could lead to adverse consequences.

Hardship Justification

The NRC staff finds that performing a plant shutdown to repair the subject piping would cycle the unit and increase the potential of an unnecessary transient resulting in undue hardship. Additionally, performing certain ASME Code repairs during normal operation would challenge the technical specification "Completion Time" and place the plant at higher safety risk than warranted. Therefore, the NRC staff determined that compliance with the specified ASME Code repair requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

3.9 Technical Evaluation Summary

The NRC staff finds that the proposed alternative will provide reasonable assurance of the structural integrity because: (1) ASME Code Case N-513-4 addresses the NRC condition in RG 1.147 for Revision 3, (2) flaw evaluations in component types added to Revision 4 of the code case are based on acceptable methodologies, and (3) the method for determining the allowable leakage rate is adequate to provide early identification of a significant increase in leakage. In addition, complying with ASME Code, Section XI requirements would result in 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 determined that the proposed alternative, Request No. RR5-02, provides reasonable assurance of structural integrity of the subject components and that complying with IWC-3120, IWC-3130, IWD-3120(b), and IWD-3400, of the ASME Code, Section XI, would result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the 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 the licensee's proposed alternative as described in Request No. RR5-02, to use ASME Code Case N-513-4, at CNS, for the fifth 10-year ISI interval which began on April 1, 2016, and is scheduled to end on February 28, 2026. If the proposed alternative is applied to a flaw near the end of the authorized 10-year ISI interval, and the next refueling outage is in the subsequent interval, the licensee is authorized to continue to apply the proposed alternative to the flaw until the next refueling outage.

The NRC staff notes that approval of this alternative does not imply NRC approval of ASME Code Case N-513-4.

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

Principal Contributor: R. Davis, NRR/DMLR/MPHB

Date: July 31, 2018

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

SAFETY EVALUATION BY THE OFFICE NUCLEAR REGULATION

FOR THE FIFTH 10-YEAR INSERVICE INSPECTION INTERVAL

RELIEF REQUEST NO. RR5-03

ALTERNATE REPAIR FOR RESIDUAL HEAT REMOVAL (RHR)

AND RHR SERVICE WATER BOOSTER PIPING

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

By letter dated August 17, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17241A048), supplemented by letter dated May 16, 2018 (ADAMS Accession No, ML18143B464), Nebraska Public Power District (NPPD, the licensee) requested relief from the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, IWA-3120(b), at Cooper Nuclear Station (CNS). The licensee submitted Relief Request RR5-03 for the U.S. Nuclear Regulatory Commission's (NRC) review and approval for the repair of residual heat removal (RHR) service water booster (RHRSWB) piping using ASME Code Case N-513-4, "Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Class 2 or 3 Piping Section XI, Division I," with exceptions.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(2), the licensee submitted Relief Request RR5-03 on the basis that compliance with the specified ASME Code repair would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The NRC staff notes that the licensee revised the original Relief Request RR5-03 as part of a response to the NRC's request for additional information. Therefore, the NRC's evaluation below is for Relief Request RR5-03, as documented in the licensee's letter dated May 16, 2018. The NRC staff further notes that this is a contingency relief request. The licensee postulated degradation in the RHRSWB piping as part of analyses and technical basis to support the relief request.

2.0 REGULATORY EVALUATION

Adherence to Section XI of the ASME Code is mandated by 10 CFR 50.55a(g)(4), "Inservice inspection standards requirements for operating plants," which states, in part, that ASME Code Class 1, 2, and 3 components will meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in ASME Code, Section XI.

The regulation at 10 CFR 50.55a(z), "Alternatives to codes and standards requirements," states, in part, that:

> Alternatives to the requirements of paragraphs (b) through (h) of [10 CFR 50.55a] or portions thereof 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 applicant or licensee must demonstrate that:
>
> (1) *Acceptable level of quality and safety.* The proposed alternative would provide an acceptable level of quality and safety; or
>
> (2) *Hardship without a compensating increase in quality and safety.* 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 the use of an alternative and the NRC to authorize the proposed alternative.

3.0 TECHNICAL EVALUATION

3.1 ASME Code Component(s) Affected

The affected components are the ASME Code Class 3 RHRSWB piping with a maximum operating pressure of less than or equal to 490 per square inch gauge (psig) and a maximum operating temperature less than 200 degrees Fahrenheit (°F).

The RHR and RHRSWB systems are standby systems that typically operate during testing or plant shutdown. The licensee stated that the systems are designed such that RHRSWB operates at a higher pressure than RHR. Under this design, if there is an internal leak within a RHR heat exchanger, RHRSWB water, which is raw water from the Missouri River, will leak into the RHR system.

As stated in the licensee's letter dated May 16, 2018, "The RHRSWB System is designed to provide an adequate supply of cooling water to the RHR heat exchangers during postulated accident and transient conditions to remove the design RHR System heat load and at adequate pressure to prevent uncontrolled release of fission products to the environment due to a RHR heat exchanger tube failure."

As stated in the licensee's letter dated May 16, 2018, "RHRSWB System at CNS has exhibited a history of degradation similar to raw fresh water systems throughout the nuclear industry. Degradation requiring immediate action to address leakage or observed thinning in the system is generally due to localized corrosion mechanisms."

3.2 Applicable Code Edition and Addenda

The applicable Code of Record for the fifth 10-year inservice inspection (ISI) interval and the ISI program is the 2007 Edition through 2008 Addenda of the ASME Code, Section XI. The fifth ISI interval started on April 1, 2016, and will end on February 28, 2026.

3.3 Applicable ASME Code Requirements

As stated in the licensee's letter dated May 16, 2018, "ASME Code, Section XI, IWD-3120(b) requires that components exceeding the acceptance standards of IWD-3400 be subject to supplemental examination or to a repair/replacement activity.

3.4 Reason for Request

The licensee proposed to use ASME Code Case N-513-4 to perform repair/replacement activities for degraded RHRSWB piping, which has a maximum operating pressure in excess of 275 psig and is the maximum allowed pressure in the code case. The licensee stated that a plant shutdown would be required within the action statement timeframes, as specified in the plant Technical Specifications, if the degraded RHRSWB piping is repaired in accordance with the ASME Code, Section XI. The licensee noted that plant shutdown activities result in additional radiation dose and plant risk that would be inappropriate when a degraded pipe condition is demonstrated to retain adequate margin to complete the component's function. The licensee noted that the use of an acceptable alternative analysis method in lieu of the immediate repair of the degraded RHRSWB piping will permit additional extent of condition examinations on the affected systems while allowing time for safe and orderly long term repair actions if necessary. ASME Code Case N-513-3 does not allow evaluation of flaws located away from attaching circumferential piping welds that are in elbows, bent pipe, reducers, expanders, and branch tees. Code Case N-513-3 also does not allow evaluation of flaws located in heat exchanger external tubing or piping. Code Case N-513-4 provides guidance for evaluation of flaws in these locations.

3.5 Proposed Alternative

The licensee proposed to apply ASME Code Case N-513-4 to disposition the degraded RHRSWB piping having a maximum operating pressure of 490 psig in lieu of repair/replacement activities in accordance with the ASME Code, Section XI..

3.6 Basis for Use

The licensee stated that the ASME recognized that relatively small flaws could remain in service without risk to the structural integrity of a piping system and developed ASME Code Case N-513. NRC approval of ASME Code Case N-513-3 in Regulatory Guide (RG) 1.147, Revision 17, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," allows temporary acceptance of partial through-wall or through-wall flaws provided that all NRC-imposed conditions on the code case are met. The temporary acceptance period is the time to the next scheduled refueling outage. ASME Code Case N-513-3 requires the Owner to demonstrate system operability due to leakage. ASME Code Case N-513-3 is not applicable to piping components such as elbows, bent pipe, reducers, expanders, and branch tees and external tubing or piping attached to heat exchangers. The ASME approved Code

Case N-513-4 to expand use on these pipe locations and to revise several other areas of the code case.

The licensee stated that it will follow all requirements of the ASME Code Case N-513-4 with a few exceptions as discussed below. The NRC staff notes that the discussion below follows the organizational structure of the ASME Code Case N-513-4. The licensee also provided hardship justification as discussed below.

Scope

ASME Code Case N-513-4, Section 1, "Scope," limits its application to certain pipe components and operating conditions. Paragraph 1(c) of Section 1 limits the application to piping with a maximum operating pressure not exceeding 275 psig. The maximum operating pressure of the RHRSWB system is 490 psig. Therefore, the licensee requested relief from the pressure limits of paragraph 1(c).

The licensee stated that it will evaluate each leak using the plant operability evaluation process in order to satisfy paragraph 1(f) of Section 1. The licensee's evaluation will consider requirements or commitments established for the piping system, continued degradation and potential consequences, operating experience, and engineering judgment. The licensee will consider but is not limited to system makeup capacity, containment integrity with the leak not isolated, effects on adjacent equipment, and the potential for room flooding.

The licensee explained that the effects of leakage may impact the operability determination or the plant flooding analyses specified in paragraph 1(f) of ASME Code Case N-513-4. The licensee will determine the allowable leakage rate for a leaking flaw by dividing the critical leakage rate by a safety factor of four. The critical leakage rate is determined as the lowest leakage rate that can be tolerated and may be based on the allowable loss of inventory or the maximum leakage that can be tolerated relative to room flooding. The licensee noted that the safety factor of four on leakage is based upon ASME Code Case N-705, "Evaluation Criteria for Temporary Acceptance of Degradation in Moderate Energy Class 2 or 3 Vessels and Tanks Section XI, Division 1," dated October 12, 2006, which the NRC has accepted without condition in RG 1.147, Revision 17. Paragraph 2.2(e) of ASME Code Case N-705 requires a safety factor of two on flaw size when estimating the flaw size from the leakage rate. This corresponds to a safety factor of four on leakage for nonplanar flaws.

The licensee contended that although the use of a safety factor to determine an unknown flaw is considered conservative when the actual flaw size is known, this approach is deemed acceptable based upon the precedent of ASME Code Case N-705. The licensee noted that the subject alternative does not propose to use any portion of ASME Code Case N-705 and that its citation is intended only to provide technical basis for the safety factor on leakage.

In the submittal dated May 16, 2018, the licensee stated that if ASME Code Case N-513-4 is applied to a leaking flaw in the RHRSWB system for leakage greater than 5 gallons per minute (gpm), the leakage shall be stopped throughout the temporary acceptance period by the use of engineered mechanical clamping. The licensee further stated that the engineered mechanical clamping shall be sufficient to withstand the maximum operating pressure and removable such that the frequent periodic inspections defined in paragraph 2(e) of ASME Code Case N-513-4 may be performed.

Procedure

ASME Code Case N-513-4, Section 2, "Procedure," provides provisions for flaw characterization and periodic inspections. The licensee stated that its proposed alternative does not take exception to Section 2 of ASME Code Case N-513-4. The licensee further stated that the periodic inspection interval per paragraph 2(e) of ASME Code Case N-513-4 will demonstrate that a leaking flaw continues to meet the flaw acceptance criteria and that the flaw will not grow to an unacceptable size.

The licensee reported that during the temporary acceptance period, leaking flaws will be monitored daily, as required by paragraph 2(f) of ASME Code Case N-513-4, to confirm the analysis conditions used in the flaw evaluation remain valid. The licensee noted that significant change in the leakage rate would require reinspection, per paragraph 2(f) of the code case, and the licensee will perform any required reinspection.

Flaw Evaluation

ASME Code Case N-513-4, Section 3, "Flaw Evaluation," provides provisions for the evaluation of detected flaws. The proposed alternative does not take exception to Section 3 of ASME Code Case N-513-4. The licensee stated that allowable flaw sizes calculated by both the linear elastic fracture mechanics and branch reinforcement methods used in the code case were smaller than would be expected. The licensee evaluated the effects of jet thrust force and found that there was little difference in jet thrust force for a 0.50-inch diameter flaw size at an operating pressure of 275 psig versus 490 psig.

Acceptance Criteria

ASME Code Case N-513-4, Section 4, "Acceptance Criteria," provides provisions for the acceptance of flaws. The proposed alternative does not take exception to Section 4 of ASME Code Case N-513-4.

Augmented Examination

ASME Code Case N-513-4, Section 5, "Augmented Examination," provides provisions for the extent of condition examinations. In the submittal dated May 16, 2018, the licensee stated that with regard to augmented examination process as described in Section 5 of the code case, a sample size of at least five of the most susceptible and accessible locations will be examined within 30 days of detecting the original flaw, as required by paragraph 5(a) of ASME Code Case N-513-4. The licensee further stated that it will examine additional locations as specified in the requirements of paragraph 5(b) as it applies to paragraph 5(a).

Mandatory Appendix I

ASME Code Case N-513-4, Mandatory Appendix I, "Relations for Fm, Fb, and F for Through-Wall Flaws," provides provisions for the flaw evaluation. The proposed alternative does not take exception to Mandatory Appendix I of ASME Code Case N-513-4.

Hardship

The licensee stated that performing an ASME Code repair could have a detrimental impact on the overall risk by requiring a plant shutdown, thus requiring use of a piping system that is in standby during normal operation. The application of ASME Code Case N-513-4, along with the additional commitments associated with this relief request, will maintain acceptable structural and leakage integrity while minimizing plant risk and personnel exposure by minimizing the number of plant transients that could be incurred if degradation is required to be repaired based on the requirements of the ASME Code, Section XI. The licensee concluded that compliance with the current ASME Code requirements results in a hardship without a compensating increase in the level of quality and safety.

3.7 Duration of Proposed Alternative

The licensee stated in its submittal dated May 16, 2018, that an ASME Code, "Section XI, compliant repair/replacement will be completed prior to exceeding the next scheduled refueling outage, or allowable flaw size, or leakage in excess of 5 gpm, whichever comes first. This relief request will be applied for the duration of the fifth 10-year ISI interval. If a flaw is evaluated near the end of the interval and the next refueling outage is in the subsequent interval, the flaw may remain in service under this relief request until the next refueling outage."

3.8 NRC Staff Evaluation

The NRC staff evaluated the licensee's proposed alternative with respect to the provisions of ASME Code Case N-513-4, which has not been approved for use by the NRC. For clarity, the NRC staff's evaluation of the proposed alternative will follow the organizational structure of ASME Code Case N-513-4. The NRC staff's evaluation is limited to CNS only and does not constitute a generic review of the code case.

Scope

In its submittal dated May 16, 2018, the licensee provided the isometric drawings showing pipe support locations, pipe routing, nominal pipe size, outside diameter, pipe schedule, and wall thickness of various RHRSWB pipe segments. Based on the review of the information provided, the NRC staff determined that the scope of the subject piping is appropriately defined and identified. Therefore, the NRC staff finds that the subject piping satisfies paragraph 1(a) of ASME Code Case N-513-4.

Paragraph 1(c) of ASME Code Case N-513-4 limits the maximum operating pressure to 275 psig. To justify the use of this code case on the RHRSWB piping with a maximum operating pressure of 490 psig, the licensee demonstrated in Enclosure 2 of the May 16, 2018, submittal that (1) the structural integrity of the RHRSWB piping is achieved by a flaw evaluation, (2) the structural integrity of the subject piping will be maintained by periodically inspecting and monitoring, and (3) the leaking flaw and/or leak rate are limited to within the allowable value. Based on the flaw evaluation, periodic inspections and monitoring, and the leakage limit, the NRC staff finds the proposed alternative is acceptable with respect to paragraph 1(c).

Paragraph 1(f) of ASME Code Case N-513-4 requires that "the Owner consider the effects of leakage in demonstrating system operability and performing plant flooding analyses." To address the NRC staff's question regarding what action will be taken before the leakage limit of 5 gpm is reached, the licensee explained that identified leaks will be quantified and entered into the CNS Corrective Action Program and assessed for system operability. The licensee will evaluate the leakage based on its plant-specific operability evaluation process as discussed above. The NRC staff finds it acceptable because the licensee's operability evaluation will consider system makeup capacity, containment integrity with the leak not isolated, effects on adjacent equipment, and the potential for room flooding.

The NRC staff notes that the licensee's flooding analysis showed that during normal operation, sump pumps in the four reactor building quad rooms are sized to handle leakage from the reactor building structure, which includes the torus area. The licensee stated that two 50 gpm sump pumps in each quad room have the capacity to pump a total of 100 gpm of leakage from each quad room of the reactor building structure. The sump pumps in the Emergency Condensate Storage Room in the control building basement have a total pumping capacity of 100 gpm. The NRC staff finds that the sump pumps in the reactor building have sufficient capacity to manage a 5-gpm leak.

The NRC staff noted that as a part of the flooding analysis, the licensee identified the corresponding coolant flow from postulated pipe breaks in each of the above equipment areas, based on system pressure (adjusting for head loss and elevation difference), break area, and discharge coefficient. The NRC staff finds that all the licensee's analyzed breaks are adequately derived based on equipment available to safely shutdown and maintain the reactor in a safe shutdown condition. The NRC staff verified that the allowable break flows evaluated in the licensee's flooding analysis are significantly greater than the leakage limit of 5 gpm. The NRC staff finds that the proposed 5 gpm leakage limit is acceptable because a margin exists between the 5 gpm limit and the allowable break flow in the equipment areas.

The NRC staff asked for the minimum leak rate and the flow rate that would cause the RHRSWB to not perform its intended function, and the average normal flow rate in the subject pipe. The licensee stated that each RHRSWB pump is verified to produce a flow rate of up to 4000 gpm as part of quarterly inservice testing and biennial performance testing. The RHRSWB System flow assumed in the analyses is 4000 gpm per pump with one pump operating in one loop. The NRC staff notes that, according to the licensee, RHR heat exchanger thermal performance is based on reduced tube side flow of 3500 gpm. The NRC staff determined that there is a margin of 500 gpm (4000 gpm – 3500 gpm) exists between the 4000 gpm that the RHRSWB system can deliver and a required coolant flow of 3500 gpm at the RHR heat exchanger tube side. The NRC staff finds that a RHRSWB piping leakage of 5 gpm is much less than the margin of 500 gpm. Therefore, a leak of 5 gpm has a minimal impact to RHR heat exchanger performance. The NRC staff also recognizes that the leakage limit of 5 gpm is within the uncertainty of the flow instrumentation and within the capacity of RHRSWB system to provide the required flow.

The NRC staff asked the licensee how the leakage in the RHRSWB piping can be detected and how the operators in the control room are notified. The licensee explained that a leak in the RHRSWB piping can be detected and communicated to the control room in many ways. The CNS Operations department conducts operator tours that vary from once per week (full torus floor area) to twice daily in the reactor building RHR heat exchanger room and control building basement. In addition to operator tours, plant walkdowns conducted by various departments, such as Radiation Protection, Maintenance, System Engineering, and management observations, provide additional opportunities for a leak to be detected and reported.

The licensee noted that the minimum detectable leakage can vary from a small weep to a steady stream, depending on the size of the leak opening. Generally, leaks from the RHRSWB piping would be expected to be observed at a low point below the leak such as on the floor in these locations. The RHRSWB system is operated on an intermittent basis (minimum quarterly) so the probability of detecting a leak would be greatest when the RHRSWB system is in operation. Other times, the piping is continuously pressurized at the service water operating pressure of approximately 30 to 50 psig. Larger leaks could be detected by the radwaste operators by observing changes in the floor drain and/or equipment drain collector tank levels. The licensee stated that engagement by CNS personnel provides reasonable assurance that a leak would be visually identified and reported to the control room in a timely manner. The NRC staff finds that the licensee does have adequate means and reasonable intervals to monitor the subject piping and report the leak before reaching the limit of 5 gpm.

The NRC staff reviewed the pipe support locations on the piping isometric drawings and the jet force loads provided in Enclosure 2 of Attachment 3 to the submittal dated May 16, 2018. The NRC staff confirmed that, should the leakage occur, the existing pipe supports will be able to support the jet force loads from the allowable flaw size.

The NRC staff determined that the licensee provided reasonable assurance that leakage from the RHRSWB piping will be detected early based on various indications, so that the operators can take appropriate corrective actions. The NRC staff finds that the licensee has satisfied paragraph 1(f) because it has considered effects of leakage and will implement a leakage limit of 5 gpm in demonstrating system operability.

Based on above evaluation, the NRC staff finds that the proposed alternative satisfies the requirements of Section 1 of ASME Code Case N-513-4, except paragraph 1(c). However, the licensee has provided adequate justifications for the deviation from paragraph 1(c).

Procedure

The NRC staff determines that the proposed alternative will follow and does not take exception to Section 2 of ASME Code Case N-513-4. Therefore, the NRC staff finds that the proposed alternative is acceptable with respect to Section 2 of ASME Code Case N-513-4.

Flaw Evaluation

The NRC staff notes that Table 4 in Enclosure 2 of Attachment 3 to the submittal dated May 16, 2018, provides generic allowable flaw sizes, not specific to CNS. The licensee explained that Table 4 is meant to illustrate the effect of pressure on the allowable through-wall flaw size. The licensee stated that the plant-specific allowable flaw size is a function of the applied loading at the specific location for which this code case will be applied. The licensee noted that the exact location for which the code case will be applied cannot be known at the time of submittal. The licensee further stated that the size of a flaw will be limited to a size that does not exceed the acceptance criteria of the flaw evaluation or a flaw size resulting in a leakage rate of greater than 5 gpm, whichever is less. Therefore, the NRC staff finds that it is acceptable for the licensee to provide generic allowable flaw sizes as a reference for comparison purposes. The NRC staff notes that with a higher operating pressure of 490 psig, the allowable flaw size based on structural consideration will be smaller than those at the code case limit of 275 psig. On the other hand, the allowable minimum pipe wall thickness will be higher for the 490 psig than for the 275 psig case. The NRC staff finds that the smaller allowable flaw size and higher minimum wall thickness are conservative and it is appropriate to compensate for the higher operating pressure. However, the NRC staff notes that the final allowable flaw size will be appropriately limited to a size that does not exceed the acceptance criteria of the flaw evaluation or a flaw size resulting in a leakage rate of greater than 5 gpm, whichever is less.

In addition, the licensee stated that if leakage does occur in the future, it will follow the requirements in the code case to analyze the flaw based on plant-specific parameters.

Based on the above, the NRC staff finds that the proposed alternative will follow and does not take exception to Section 3 of ASME Code Case N-513-4. Therefore, the proposed alternative is acceptable with respect to Section 3 of ASME Code Case N-513-4.

Acceptance Criteria

The NRC staff finds that the proposed alternative will follow and does not take exception to Section 4 of ASME Code Case N-513-4. Therefore, the proposed alternative is acceptable with respect to Section 4 of ASME Code Case N-513-4.

Augmented Examination

ASME Code Case N-513-4, Paragraph 5(a) requires, in part, that:

> A sample size of at least five of the most susceptible and accessible locations, or, if fewer than five, all susceptible and accessible locations shall be examined within 30 days of detecting the flaw.

Paragraph 5(b) requires that:

> When a flaw is detected, an additional sample of the same size as defined in [paragraph 5](a) shall be examined

In the submittal dated May 16, 2018, the licensee stated that it will examine the same number of pipe locations as required by paragraph 5(a) of ASME Code Case N-513-4. In addition, the licensee stated that it will examine additional locations as specified in the requirements of paragraph 5(b) as it applies to 5(a).

The NRC staff finds that the licensee has satisfied the provisions of Section 5 of ASME Code Case N-513-4 and, therefore, the proposed augmented examination is acceptable.

Mandatory Appendix I

The NRC staff finds that the proposed alternative will follow and does not take exception to the Mandatory Appendix I of ASME Code Case N-513-4. Therefore, the proposed alternative is acceptable with respect to Mandatory Appendix I of ASME Code Case N-513-4.

Hardship Justification

The NRC staff evaluated the technical basis of this request against the criteria contained in 10 CFR 50.55a(z)(2). The NRC staff notes that performing the specified ASME Code compliant repairs will require a plant shutdown, which will lead to unnecessary plant transients and additional radiation dose. The plant shutdown is undesirable in terms of plant safety because it increases loads on the systems and components. In addition, the ASME Code compliant repair of the subject piping would not significantly increase plant quality or safety. The NRC staff, therefore, finds that requiring an ASME Code compliant repair is a hardship or unusual difficulty without a compensating increase in plant quality or safety.

In summary, the NRC staff finds that the proposed alternative will provide reasonable assurance of the structural integrity of the RHRSWB piping because (1) the licensee will follow the requirements of N-513-4 with exceptions for which the licensee has provided appropriate justifications, (2) the licensee will perform flaw evaluations in combination with periodic inspections to ensure that the flaw will not exceed the allowable per the code case, and (3) the licensee will implement a leakage limit of five gpm.

4.0 CONCLUSION

As set forth above, the NRC staff determines that the proposed alternative as documented in the submittal dated May 16, 2018, provides reasonable assurance of structural integrity of the RHRSWB piping. The NRC staff concludes that complying with the specified ASME Code requirements 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 Relief Request RR5-03, as documented in the submittal dated May 16, 2018, for the fifth 10-year ISI interval on the basis that an ASME Code, Section XI compliant repair/replacement will be completed prior to exceeding the next scheduled refueling outage, or allowable flaw size, or leakage in excess of 5 gpm, whichever comes first.

The NRC staff further concludes that if a flaw is evaluated near the end of the interval and the next refueling outage is in the subsequent interval, the flaw may remain in service under this relief request until the next refueling outage.

The NRC staff notes that the authorization of Relief Request RR5-03 does not imply NRC approval of ASME Code Case N-513-4.

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

Principal Contributor: J. Tsao, NRR/DMLR/MPHB

Date of issuance: July 31, 2018

SUBJECT: COOPER NUCLEAR STATION – REQUESTS FOR RELIEF ASSOCIATED WITH THE FIFTH 10-YEAR INSERVICE INSPECTION INTERVAL PROGRAM (CAC NOS. MG0175 THROUGH MG0179; EPIDS L-2017-LLR-0062 THROUGH L-2017-LLR-0066) DATED JULY 31, 2018

DISTRIBUTION:
PUBLIC
PM File Copy
RidsACRS_MailCTR Resource
RidsNrrDorlLpl4 Resource
RidsNrrPMCooper Resource
RidsNrrLAPBlechman Resource
RidsRgn4MailCenter Resource
RidsNrrDmlrMvib Resource
RidsNrrDmlrMphb Resource
RidsNrrDssSnpb Resource
RidsNrrDssSrxb Resource
JJenkins, NRR
RDavis, NRR
JTsao, NRR
APatel, NRR
JBowen, OEDO
LBurkhart, OEDO

ADAMS Accession No. ML18183A325 *by e-mail

OFFICE	NRR/DORL/LPL4/PM	NRR/DORL/LPL4/LA	NRR/DMLR/MVIB/BC*	NRR/DMLR/MPHB/BC*
NAME	TWengert	PBlechman w/comments	SRuffin/DAlley	DAlley/SRuffin
DATE	7/17/18	7/11/18	4/24/18 4/28/18 5/24/18	3/25/18 5/24/18
OFFICE	NRR/DSS/SNPB*	NRR/DORL/LPL4/BC		
NAME	RLukes	RPascarelli		
DATE	5/1/18	7/31/18		