September 18, 2001

Mr. J. H. Swailes
Vice President of Nuclear Energy
Nebraska Public Power District
P. O. Box 98
Brownville, NE 68321

SUBJECT: COOPER NUCLEAR STATION - INSERVICE INSPECTION RELIEF REQUEST RI-26, 28, 30, 31, 32, AND RC-07 (TAC NO. MB1415)

Dear Mr. Swailes:

In a letter dated December 4, 2000, as supplemented on July 16, 2001, Nebraska Public Power District (NPPD/the licensee) requested relief (Relief Request RI-26, 28, 30, 31, 32 and RC-07) from certain American Society of Mechanical Engineers Boiler and Pressure Vessel Code (Code) requirements. The NRC staff has evaluated your request and determined that Relief Requests RI-26, RI-28 and RC-07 are acceptable. Accordingly, the NRC staff concludes that compliance with the specified Code requirements for RI-26 and RI-28 would result in hardship or unusual hardship without a compensating increase in the level of quality and safety, and therefore, the proposed alternatives are authorized pursuant to a 10 CFR 50.55a(a)(3)(ii) for the third 10-year ISI interval. For RC-07 the NRC staff concludes that the proposed alternative provides an acceptable level of quality and safety, and therefore, the proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i). The NRC staff has denied RI-30 as it request deviations from requirements in 10 CFR Part 50 which requires an exemption rather than a relief request. Relief Request RI-31 is not necessary due to a recent rule change. By letter dated July 16, 2001, the licensee withdrew RI-32.

A copy of the NRC's safety evaluation is enclosed. This completes the technical review for TAC No. MB1415. If you have any questions, please contact Mohan Thadani at 301-415-1476.

Sincerely,

/RA/

Robert A. Gramm, Chief, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-298

Enclosure: Safety Evaluation

cc w/encls: See next page

September 18, 2001

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

INSERVICE AND CONTAINMENT INSPECTION RELIEF

REQUESTS RI-26, 28, 30, 31, AND 32

COOPER NUCLEAR STATION

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NUMBER 50-298

1.0 INTRODUCTION

The inservice inspection (ISI) of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) Class 1, Class 2 and Class 3 components is to be performed in accordance with Section XI of the ASME Code and applicable edition and addenda as required by Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(g), except where specific written relief has been granted by the Nuclear Regulatory Commission (NRC) pursuant to 10 CFR 50.55a(g)(6)(i). Section 10 CFR 50.55a(a)(3) states in part that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if the licensee demonstrates that: (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), 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 the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The inservice inspection code of record for Cooper Nuclear Station's (CNS) third 10-year ISI interval is the 1989 Edition of Section XI of the ASME Boiler and Pressure Vessel Code.

By letter dated December 4, 2000 (Reference 1), the Nebraska Public Power District (NPPD/the licensee) requested relief from successive ultrasonic testing (UT) inspection frequencies of reactor vessel shell welds (RI-26), periodic monitoring of mid-wall slag inclusions in the reactor vessel shell welds found during GERIS 2000 UT testing (RI-28), UT of components for single side access (RI-30), UT length sizing acceptance criteria (RI-31), qualification of NDE personnel to CP-189 (RI-32), and performing the VT-3 visual examination at the end of the interval for Items E1.12 and E1.20 (RC-07).

2.0 RELIEF REQUEST RI-26, SUCCESSIVE INSPECTIONS OF REACTOR PRESSURE VESSEL WELDS, ITEM NUMBERS B1.11, B1.12, B.1.21, B1.22, B1.30

2.1 Code Requirements for which Relief is Requested

IWB-2420(a) requires the sequence of component examinations established during the first interval be repeated each successive inspection interval to the extent practical. During the first interval, ASME Section XI, 1974 Edition through Summer 1975 addenda required a percentage of each reactor pressure vessel longitudinal/circumferential welds be tested. In the second interval, the governing code was ASME Section XI, 1980 Edition through the Winter 1981 Addenda. This code required the examination of only one beltline longitudinal and circumferential weld. During the first period of the third interval the requirements were 10 CFR 50.55a(g)(6)(ii)(A) which required 100 percent of the reactor pressure vessel longitudinal and circumferential welds be ultrasonically examined under an augmented inspection program.

2.2 Licensee's Proposed Alternative to Code

Pursuant to 10 CFR 50.55a(a)(3)(ii) the licensee proposes to schedule future reactor vessel shell weld examinations on a 10-year inspection frequency based on the newly established reactor pressure vessel baseline examination they completed during the first period of the third interval. The next examination of the reactor pressure vessel would be performed the first period of the fourth interval.

2.3 Licensee's Basis for Relief (as stated)

"In accordance with ASME Section XI, it is permissible to defer reactor pressure vessel examinations to the end of an interval (no more than 10 years between inspections). In the case of CNS, the subject reactor pressure vessel welds were examined to the extent practical in the first period of the third inspection interval. In previous intervals, the subject reactor pressure vessel shell welds were inaccessible for examination. Because a new baseline has been established in the first period of the third interval and the intent of the Code is to ensure that inspections are performed at least once each 10 years, it is appropriate to schedule the subject successive examinations for the first period of the fourth interval.

Performing the vessel shell weld inspections in accordance with the periodic inspection criteria results in the following:

1. Mobilizing the inspection device in each period of the third interval.
2. Additional outage activities to receive, transport to the refueling floor, set up, calibrate, install in the vessel, remove, decontaminate, repackage, and demobilize the inspection tool.
3. Increases in plant radiation exposure.
4. Increases in potential for personnel contaminations.
5. Increases risk of personnel hazards associated with setup/takedown hazards.
6. Increases in radioactive waste."

2.4 Evaluation

Section 50.55a(g)(6)(ii)(A)(3) imposed augmented examination requirements for reactor pressure vessels. Section 50.55a(g)(6)(ii)(A)(3) allowed licensees with fewer than 40 months remaining in the inservice inspection interval in effect on September 8, 1992, to defer the required augmented reactor vessel examination to the first period of the next inspection interval. Section 50.55a(g)(6)(ii)(A)(3)(iii) states that: "If the deferred augmented examination is used as a substitute for the normally scheduled reactor vessel shell weld examination, subsequent reactor vessel shell weld examinations must be performed during the first period of the successive inspection intervals." In this case, the rule requires successive reactor pressure vessel weld examinations (100 percent) to be performed in the same period of the next interval, 10 years later.

For CNS, it is more practical to perform these reactor pressure vessel examinations at one time due to the complexity of remote automated ultrasonic equipment and challenging/limited access within a reactor pressure vessel. What the licensee is proposing is consistent with 50.55a(g)(6)(ii)(A)(3)(iii). Requiring the licensee to maintain its original ASME Code successive examination schedule would result in the next examination of the reactor vessel shell welds being conducted on a frequency greater than that required by the regulations. This would result in hardship without a compensating increase in the level of quality and safety.

Additionally, minimizing the number of augmented examinations to once each interval within the vessel actually is desirable. Automated ultrasonic equipment is extremely complex with many parts. Therefore, fewer entries into the reactor vessel internal areas provides for less risk of unretrievable, lost parts that may cause damage during operation.

2.5 Conclusion

Based on the discussion above, the NRC staff concludes that compliance with the specified ASME Code requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the proposed alternative RI-26 is authorized for the third 10 year ISI interval.

3.0 RELIEF REQUEST RI-28, INSPECTION OF REACTOR PRESSURE WELDS VLA-BA-3 AND VLC-BB-2

3.1 Code Requirements for which Relief is Requested (as stated)

"IWB-2420(b) requires that if flaw indications or relevant conditions are evaluated in accordance with IWB-3132.4 or IWB-3142.4, respectively, and the component qualifies as acceptable for continued service, the areas containing such flaw indications or relevant conditions shall be reexamined during the next three inspection periods listed in the schedules of the inspection programs of IWB-2410."

3.2 Licensee's Proposed Alternative to Code

Pursuant to 10 CFR 50.55a(a)(3)(ii), the licensee's proposed alternative is to defer successive inspections for three (3) periods of five (5) mid-wall slag inclusions found during the RFO-18 GERIS 2000 automated examinations of the reactor pressure vessel. The next scheduled examination would be performed during the first period of the fourth interval.

3.3 Licensee's Basis for Relief (as stated)

"The five indications have been dispositioned as slag deposits which remained in the welds as a result of insufficient back gouging and/or cleaning of the weld root area during the fabrication process. The vertical seams in this Combustion Engineering vessel are a double "U" type weld configuration. The nature of these indications is similar to indications seen by General Electric in other Boiling Water Reactor Pressure Vessel assembly weld examinations and to indications previously identified in the Feedwater and Main Steam vessel to nozzle welds at CNS.

The construction radiographs for welds VLA-BA-3 and VLC-BB2 were reviewed. However, the indications could not be identified on the film. Indications from thin slag inclusions in thick walled vessels are not always apparent on radiographs but they do have sufficient reflectivity to be characterized by ultrasonic examination. The stress distribution in the mid-wall of heavy plate is lower than the surface stress. Based on industry experience, it is highly unlikely that service induced flaws would develop in the vessel mid-wall. Based on this investigation, it is concluded that the flaws are fabrication related and are not service induced.

Fracture mechanics evaluations were performed in accordance with the requirements of IWB-3600. The results demonstrated that the flaws are acceptable for continued operation through the end of life as defined in the current license for CNS. Since the accessible Category B-A Reactor Vessel Shell welds were examined during RFO-18 to the extent practical, no additional shell weld examinations were required. However, successive examinations are required by Code to monitor the subject flaws for growth during the next three periods.

The GERIS 2000 system used to examine the reactor pressure vessel (RPV) shell welds from the inside of the vessel, provided increased coverage, access to previously unexamined areas, and better detection and resolution capabilities for flaws that may exist in the examination regions. As a result, fabrication flaws have been detected that exceed Section XI acceptance criteria. CNS has investigated the subject flaws to the extent practical and has determined that the flaws have existed since the fabrication of the vessel. Based on the fracture mechanics evaluation, it is reasonable to conclude that the subject flaws are stable and do not require monitoring on an increased frequency.

The GERIS 2000 system has been used at many other plants to perform RPV examinations. As a result it has been contaminated by its frequent use. Requiring CNS to perform the subject inspections on an increased frequency will result in the following:

1. Required additional activities to receive, transport to the refueling floor, set up, calibrate, install in the vessel, remove, decontaminate, repackage, and demobilize the inspection tool.
2. Increases in plant radiation exposures.
3. Increases the potential for personnel contaminations.
4. Increases safety concerns associated with setup/takedown hazards.
5. Increases in radioactive waste."

3.4 <u>Evaluation</u>

Five flaw indications were detected during RFO-18 that exceeded the acceptance standards of Table IWB-3510-1. Two of the flaw indications are located in vertical weld VLA-BA-3 and the other three are located in vertical weld VLC-BB-2. The two flaw indications located in vertical weld VLA-BA-3 were previously found by manual ultrasonic examination techniques from the outside diameter in the fall of 1995. The flaw evaluations performed at that time determined that these flaws were mid-wall reflectors, acceptable by ASME Section XI. The five flaw indications identified were subsequently plotted, evaluated and determined to be in the mid-wall region of the weld root during the RFO-18 GERIS 2000 automated examinations.

The construction radiographs for welds VLA-BA-3 and VLC-BB-2 were reviewed by the licensee but the indications could not be identified on the film. Fracture mechanics evaluations were performed by the licensee in accordance with the requirements of IWB-3600 with the results demonstrating that the flaws are acceptable for continued operation through the end of life as defined in the current license for the licensee.

The NRC staff agrees that in some instances, the orientation of a flaw may be such that it is not detectable by radiography and yet by ultrasonics due to the angle of incidence. Mid-wall laminates that are parallel to the outer diameter and inner diameter are particularly difficult for radiography to detect. Recognizing that there is no difference in the stress distribution and morphology of mid wall slag inclusions between a pressurized water reactor and a boiling water reactor (BWR), the NRC staff concludes that these are non-threatening flaws as defined below.

The NRC staff's "Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report," dated July 28, 1998 concluded: "The BWRVIP proposed alternative criteria to the ASME Code for IWA-2420, Successive Examinations, would eliminate these examinations for "non-threatening" flaws, e.g., such as embedded flaws from material manufacturing or vessel fabrications which experience negligible or no growth during the design life of the vessel, provided that the following conditions are met:

1. The flaw is characterized as subsurface,
2. The NDE technique and evaluation that detected and characterized the flaw as originating from material manufacture or vessel fabrication is documented in a flaw evaluation report,
3. The vessel containing the flaw is acceptable for continued service in accordance with IWB-3600 and the flaw is demonstrated acceptable for the intended service life of the vessel."

The NRC staff recognizes that these mid-wall slag inclusions have experienced over twenty (20) years of service with no flaw growth either to the inner or outer wall of the vessel. Based on the service experience to date, the NRC staff expects only negligible growth over the design life of the vessel. Therefore, compliance with the specified Code requirement would not provide an increase in assurance of structural integrity commensurate with the burden associated with Code compliance.

3.5 Conclusion

Based on the discussion above, the NRC staff concludes that compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the proposed alternative RI-28 is authorized for the third 10 year ISI interval.

4.0 RELIEF REQUEST RI-30, SINGLE SIDE ACCESS FOR UT EXAMINATION, ITEM NOs B1.11, B1.12, B1.21, B1.22, B3.90, B5.10, B5.130, B8.10, B9.11, B9.31, B10.10, C1.10, C1.20, C1.30, C2.21, C2.22, C2.32, C5.51, C5.61

4.1 Code Requirements for Which Relief is Requested

The licensee is requesting relief from 10 CFR 50.55a(b)(2)(xv)(G)(1), (2) and 10 CFR 50.55a(b)(2)(xvi)(A) which define new requirements for UT coverage and qualification demonstration which affect both piping and reactor pressure vessel examinations.

4.2 Evaluation

This is a Title 10 Part 50 requirement rather than an ASME Code requirement. Therefore, the licensee needs to submit an exemption request per 10 CFR 50.12 for this change.

5.0 RELIEF REQUEST RI-31, REACTOR PRESSURE LENGTH SIZING CRITERIA

5.1 Code Requirements for Which Relief is Requested

ASME Code, Section XI, 1995 Edition, 1996 Addenda, Appendix VIII, Supplement 4, Subparagraph 3.2(b), length sizing acceptance criteria, requires that flaw lengths estimated by ultrasonics, be the true length - 1/4 inch + 1 inch.

5.2 Licensee's Proposed Alternative to Code

In lieu of the length sizing requirements of the ASME Code Section XI, 1995 Edition, 1996 Addenda, Appendix VIII, Supplement 4, Subparagraph 3.2(b), a length sizing acceptance criteria of 0.75 inch RMS error will be used.

5.3 Evaluation

The relief requested from the requirements of subparagraph 3.2(b) is no longer needed because the relevant portion of 10 CFR Sub Part 50.55a was changed. The NRC published in the *Federal Register* dated March 26, 2001 (66 FR 16390), a rule change to 10 CFR 50.55a(b)(2)(xv)(C)(1), which deals with flaw detection criteria. The rule change corrected an earlier administrative error with the length sizing criterion in the regulation. Therefore, the relief sought is no longer required.

5.4 Conclusion

The NRC staff has determined relief is not required as result of the rule change to 10 CFR 50.55a(b)(2)(xv)(C)(1).

6.0 RELIEF REQUEST RI-32, RELIEF FROM CP-189, 1991 EDITION

During a May 25, 2001, conference call the licensee agreed to withdraw relief request RI-32. By letter dated July 16, 2001, the licensee withdrew this relief request.

7.0 RELIEF REQUEST RC-07, RELIEF FROM SUBSECTION IWE OF ASME SECTION XI

In the *Federal Register* dated August 8, 1996 (61 FR 41303), the NRC amended its regulations to incorporate by reference the 1992 edition with 1992 addenda of Subsections IWE and IWL of Section XI of the ASME Boiler and Pressure Vessel Code (Code). Subsections IWE and IWL provide the requirements for inservice inspection (ISI) of Class CC (concrete containment), and Class MC (metallic containment) of light-water cooled power plants. The effective date for the amended rule was September 9, 1996, and it requires the licensees to incorporate the new requirements into its ISI plans and to complete the first containment inspection by September 9, 2001. However, a licensee may propose alternatives to or submit a request for relief from the requirements of the regulation pursuant to 10 CFR 50.55a(a)(3) and (g)(5).

By letter dated December 4, 2000 (Reference 1), NPPD proposed an alternative to the requirements of Subsection IWE of ASME Code, Section XI (Relief Request RC-07) for its CNS. The NRC's findings with respect to authorizing the alternative or denying the proposed request are discussed in this evaluation.

7.1 Code Requirements:

ASME Section XI, Subsection IWE, Table IWE-2500-1, Examination Category E-A, "Containment Surfaces," requires VT-3 examinations be performed 100 percent prior to each Type A test at the end of the interval.

7.2 Code Requirements from Which Relief is requested:

Relief is requested from performing the VT-3 visual examination at the end of the interval for Items E1.12 and E1.20.

7.3 Basis for Relief:

Subsection IWE, Table IWE-2412-1, Inspection Program B, requires approximately one-third of the examinations in each category to be performed during each inspection period. This method of distributing the examinations over the entire interval increases the probability of early detection of a service generated flaw. Since the examination of containment surfaces in accordance with Subsection IWE has only been required since September 1996, the industry does not have significant experience in performing these examinations. If a flaw is identified, additional areas will be inspected and the requirements for successive inspections will be applied. Distributing the examinations over the period will also facilitate development of examination techniques and improvement in examiner training. The ASME has endorsed this alternative in Code Case N-601.

Relief is requested in accordance with 10 CFR 50.55a(a)(3)(i). Distributing the VT-3 visual examinations over the entire inspection interval in accordance with Code Case N-601, provides an adequate level of quality.

7.4 Proposed Alternative Examination:

VT-3 visual examination may be performed at any time during the interval in accordance with Code Case N-601. The requirements for successive inspections in IWE-2420 shall be met.

7.5 Staff Evaluation of RC-07:

In lieu of meeting the requirements of Table IWE-2500-1 (1992 Edition), Category E-A, Items E1.12 and E1.20 that 100 percent of VT-3 visual examinations shall be performed at the end of the interval, the licensee proposed to perform the VT-3 visual examinations on accessible surface areas of the containment in accordance with Code Case N-601.

The NRC staff finds that performing visual examinations on the accessible surfaces of the containment structure during the course of each inspection interval will provide a more uniform method for monitoring containment surfaces than following the requirements Table IWE-2500-1 (1992 Edition). The alternative frequency for performing these visual examinations is based on the recommendation by Code Case N-601 that the VT-3 examinations in Table IWE-2500-1, Category E-A be performed at any time during the interval of inspection. In doing so, the condition of the containment can be assessed periodically rather than once in 10 years. The integrity of the containment can be better monitored between the 10 CFR Part 50, Appendix J testing, and the visual examinations required by Table IWE-2500-1. On this basis, the NRC staff concludes that the alternative proposed by the licensee based on Code Case N-601 provides an acceptable level of quality and safety, and is authorized pursuant to 10 CFR 50.55a(a)(3)(i).

7.6 Conclusion:

Based on the NRC staff's review of the information provided in the request for relief (Relief Requests RC-07), the NRC staff concludes that the licensee's proposed alternative will provide an acceptable level of quality and safety. Therefore, the proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i).

8.0 REFERENCE:

Letter from John H. Swailes, NPPD to NRC, "Inservice Inspection and Containment Inspection Relief Requests, Cooper Nuclear Station," dated December 4, 2000.

Principal Contributors: T. Cheng
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