November 30, 2001

Mr. David L. Wilson
Vice President of Nuclear Energy
Nebraska Public Power District
P. O. Box 98
Brownville, NE 68321

SUBJECT: COOPER NUCLEAR STATION - INSERVICE INSPECTION RELIEF REQUEST NO. RI-06, REVISION 2, FOR REACTOR PRESSURE VESSEL WELDS (TAC NO. MB2003)

Dear Mr. Wilson:

By letter dated April 24, 2001, the Nebraska Public Power District (the licensee) requested relief from the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI for the Cooper Nuclear Station (CNS) third 10-year inservice inspection (ISI) requirements. Relief was requested from requirements in Table IWB-2500-1, for Categories B-A, Items B1.11, B1.12, B1.21, B1.22, and B1.30 volumetric examination requirements for beltline region welds of the ASME Code. The ISI code of record for CNS third 10-year ISI interval is 1989 Edition of Section XI of the Code.

The licensee stated that the plant configuration limits or prevents access to Code specified reactor pressure vessel shell welds and prevents performance of the Code required examinations. Therefore, pursuant to 10 CFR 50.55a(g)(5)(iii), the licensee proposes an alternative to examine only the accessible portions of the reactor welds in lieu of impractical Code required examinations.

The Nuclear Regulatory Commission (NRC) staff has reviewed the licensee's application. Based on the review, the NRC staff concludes that the alternative proposed by the licensee for the third 10-year ISI interval at CNS will provide an acceptable level of quality and safety. Therefore, based on previously completed examinations according to 10 CFR 50.55a(g)(6)(ii)(A), and pursuant to 10 CFR 50.55a(a)(3)(i), the requested relief is authorized by law for the third ISI interval at CNS.

The NRC staff''s evaluation and conclusions are contained in the enclosed safety evaluation (SE). Should you have questions regarding this SE, please contact Mr. Mohan C. 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/encl: See next page

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*See previous concurrence
No significant changes from SE input
**No legal objection

ACCESSION NO.: ML013190737

OFFICE	PDIV-1/PM	PDIV-1/LA	EMCB/SC*	OGC/NLO**	PDIV-1/SC
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DATE	11/19/01	11/16/01	09/26/01	11/27/01	11/29/01

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 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 10 CFR 50.55a(g), except where specific relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). Section 10 CFR 50.55a(a)(3) of Title 10 of the *Code of Federal Regulations* (10 CFR) states in part that alternatives to the requirements of paragraph (g) may be used, when authorized by the Director of the Office of Nuclear Reactor Regulation, 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 ISI code of record for Cooper Nuclear Station (CNS) third 10-year ISI interval is the 1989 Edition of Section XI of the ASME Code.

2.0 LICENSEE'S REQUEST FOR RELIEF

By letter dated April 24, 2001, the Nebraska Public Power District (licensee) requested relief from the Table IWB-2500-1, Category B-A, Item B1.11, B1.12, B1.21, B1.22 and B1.30 volumetric examination requirements for beltline region welds of the ASME Code. The licensee provided the following information supporting the relief request.

2.1 Relief Requested

Relief Request No. RI-06, Revision 2, regarding Table IWB-2500-1 requirements for Item Nos. B1.11, B1.12, B1.21, B1.22, B1.30.

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

"Table IWB-2500-1, Category B-A, Item B1.11 and B1.12 requires a volumetric examination of all beltline region shell circumferential and longitudinal welds.

Table IWB-2500-1, Category B-A, Item B1.21 and B1.22 requires a volumetric examination of accessible lengths of all lower head circumferential and meridional welds.

Table IWB-2500-1, Category B-A, Item B1.30 requires a volumetric examination of 100% of the length of the shell to flange weld."

2.3 Licensee's Proposed Alternative to Code (as stated)

"In accordance with 10 CFR 50.55a(g)(5)(iii), CNS proposes to examine the accessible portions of the reactor vessel welds in lieu of the impractical Code required examinations."

2.4 Licensee's Basis for Relief (as stated)

"The Cooper Nuclear Station construction permit was issued before the effective date of implementation for ASME Section XI and thus the plant was not designed to meet the requirements of inservice inspection; therefore, 100% compliance is not feasible or practicable.

The CRD and instrument penetrations prevent direct access to most of the bottom head. Circumferential weld HMD-BB-1 is located inside the skirt and is inaccessible for examination. Portions of the Bottom Head Meridional welds, HMB-BB-1, HMB-BB-2, HMB-BB-3, HMB-BB-4, HMB-BB-5, HMB-BB-6 are located inside the vessel skirt and are inaccessible for examination. The accessible portions of these welds will be examined in conjunction with the examination of the bottom head circumferential weld, HMC-BB-1. Access to weld HMC-BB-1 is limited due to the proximity of the vessel skirt. The configuration limits scanning with the 60 degree probe. The total composite coverage is approximately 86%.

Access to the reactor vessel shell welds from the exterior is limited. Below the top of the biological shield, most of the reactor vessel is insulated with permanent reflective insulation and surrounded by a concrete biological shield. Penetrations through the biological shield provide limited access to some welds. The annular space between the inside diameter of the insulation and the outside diameter of the reactor vessel is a nominal 2 inches. There is no working space to remove the insulation panels from the vessel, which precludes both direct and remote examination of the outside surface.

In accordance with 10 CFR 50.55a(g)(6)(ii)(A), an examination of the Reactor Vessel shell welds was performed during RFO-18 using PDI qualified procedures (see Relief Request RI-04) and the GERIS 2000 ID Scanner. Supplemental manual examinations were performed to the extent practical."

3.0 <u>EVALUATION</u>

Section 10 CFR 50.55a(g)(6)(ii)(A)(2) requires all licensees to augment their reactor pressure vessel (RPV) examinations by implementing once, as part of the ISI interval in effect on September 8, 1992, the examination requirements for RPV shell welds specified in Item B1.10 of Examination Category B-A, "Pressure Retaining Welds in Reactor Vessel," in Table IWB-2500-1 of subsection IWB of the 1989 Edition of Section XI, Division 1, of the ASME Code, subject to the conditions specified in 10 CFR 50.55a(g)(6)(ii)(A)(3) and (4). The licensee is requesting NRC staff approval to examine the accessible portions of the RPV welds in lieu of the Code required 100 percent coverage. The licensee's interrogation coverage of the accessible weld portions totaled 86 percent of the total length of the RPV shell welds.

The primary interference in gaining access to the welds from the outer diameter of the RPV is the limited access of 2 inches which precludes removing the insulation. Inside the RPV, access is limited to the shell welds due to the proximity of reactor internals such as feedwater spargers, core spray spargers, alignment pins, downcomers, support lugs, etc. Staff experience with remote, automated ultrasonic testing equipment has shown that these interferences would be impractical to remove to obtain the required coverage. In addition, performance of remote, automated ultrasonic testing with GERIS 2000 ID equipment is a difficult and intensive examination which requires extensive planning, scheduling, and resources.

Welds in the vessel that are subjected to high neutron fluence levels, are important to interrogate due to Irradiation Assisted Stress Corrosion Cracking (IASCC). The welds in the beltline region receive the highest neutron fluence and are most susceptible to cracking. The licensee's data indicates that there are two longitudinal welds in this region where no coverage was obtained, i.e., VLC-BB-1 (shell ring 3) and VLB-BA-2 (shell ring 2). The other two longitudinal welds in shell ring 3 have 95 percent and 98 percent coverage and, in shell ring 2, 49.6 percent and 80.8 percent coverage. The NRC staff concludes that there is sufficient coverage of the other welds in the high fluence regions to be representative of any IASCC phenomena occurring.

Based on the percentage of weld volume examined both from the interior and exterior of the RPV, the NRC staff finds that any patterns of degradation would be detected and the licensee has performed the examination to the extent practical. On this basis, the NRC staff has determined that reasonable assurance of structural integrity of the vessel will be provided by the examinations performed and that the alternative provides an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(g)(6)(ii)(A)(5) and 10 CFR 50.55a(a)(3)(i), the licensee's proposed alternative would provide an acceptable level of quality and safety and may be authorized for the third 10-year ISI interval.

4.0 CONCLUSION

Based on the discussion above, the NRC staff has concluded that the alternative proposed in Relief Request No. RI-06, Revision 2, for the third 10-year ISI interval at CNS will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR50.55a(g)(6)(ii)(A)(5) and 10 CFR 50.55a(a)(3)(i), the NRC staff finds the proposed alternative for the third 10-year ISI interval acceptable and may be authorized by law.

Principal Contributor: T. Steingass

Date: November 30, 2001