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NLS2004091
August 26, 2004

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

Subject: Response to Request for Additional Information Regarding Risk-Informed Relief Request RI-34
Cooper Nuclear Station, NRC Docket No. 50-298, DPR-46

Reference:
1. Letter to R. Edington (Nebraska Public Power District) from U.S. Nuclear Regulatory Commission dated June 17, 2004, "Request for Additional Information Regarding Risk-Informed Relief Request RI-34 (TAC No. MC2351)."
2. Letter to U. S. Nuclear Regulatory Commission from S. Minahan (Nebraska Public Power District) dated March 11, 2004, "Risk-Informed Inservice Inspection Program (Relief Request RI-34)" (NLS2004023).

The purpose of this letter is for the Nebraska Public Power District (NPPD) to respond to the Request for Additional Information provided in Reference 1 by the Nuclear Regulatory Commission (NRC) regarding the previously submitted Relief Request of Reference 2. Attachment 1 provides a revision to RI-34 as requested by the NRC.

Question 1: *In the Applicable Time Period Section of Attachment 1, the licensee requested approval of the proposed RI-ISI program at CNS for the remainder of the third ten-year interval of the ISI Program, beginning with the last outage of the third period, and for the fourth ten-year ISI interval, which will begin on March 1, 2006. This is not consistent with the current NRC regulatory requirements that the ISI program needs to be updated every 10 years. As the proposed RI-ISI program is a part of the ISI program, it also needs to be updated every 10 years and submitted to the NRC consistent with the current American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI requirements. Therefore, the licensee's relief request (RI-34) should be revised to indicate that the subject relief request applies only to the third ten-year interval of the ISI program beginning from the third period. A separate relief request should be submitted to NRC for implementing the proposed RI-ISI program in the fourth 10-year interval of the ISI program.*

Response: The revised RI-34 Relief Request is provided in Attachment 1 which requests approval only for the remainder of the third ten-year Inservice Inspection interval.

COOPER NUCLEAR STATION
P.O. Box 98 / Brownville, NE 68321-0098
Telephone: (402) 825-3811 / Fax: (402) 825-5211
www.nppd.com

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Question 2: *In the Basis For Relief Section of Attachment 1, the licensee stated that the RI-ISI application was also conducted in a manner consistent with ASME Code Case N-578 "Risk-Informed Requirements for Class 1, 2, and 3 Piping, Method B." The staff notes that Code Case N-578 has not been endorsed by NRC in the Regulatory Guide 1.147. Therefore, the licensee should limit the application of Code Case N-578 to only the portion that was approved by NRC as referenced in Electric Power Research Institute (EPRI) Topical Report (TR) TR-112657.*

Response: ASME Code Case N-578 is not the basis for the RI-ISI relief request. The RI-ISI application at Cooper Nuclear Station (CNS) was conducted strictly in accordance with EPRI TR-112657. The referenced statement is simply meant to point out that the requirements implemented in the RI-ISI application at CNS per EPRI TR-112657 are generally consistent with Code Case N-578.

Question 3: *In Section 3, Risk-Informed Process, the licensee stated that a deviation to the EPRI RI-ISI methodology has been implemented in the failure potential assessment for the potential for thermal stratification, cycling and striping (TASCS). For clarification, provide confirmation to the following two items pertaining to the assessment of TASCS:*

a. *Confirm that the methodology for assessing TASCS in the CNS RI-ISI program is identical to the materials reliability program (MRP) methodology in EPRI TR-000701* [sic], *"Interim Thermal Fatigue Management Guideline (MRP-24)," January 2001.*

b. *The licensee stated that the final MRP guidance on the subject of TASCS will be incorporated into the CNS RI-ISI application if different from the criteria used. Confirm that only the portion of the final MRP guidance that are reviewed and approved by NRC will be incorporated into the CNS RI-ISI program.*

Response 3a: The methodology provided in EPRI Technical Report 1000701 (MRP-24) was written as an interim guideline for the evaluation of pressurized water reactors to assure that leakage would not occur in safety injection lines and drain/excess letdown lines. As such, the methodology is not strictly applicable to CNS, a boiling water reactor (BWR). However, the underlying methodology used for assessing TASCS at CNS is consistent with MRP-24.

Response 3b: Final MRP guidance is not currently available. However, CNS will incorporate the applicable NRC-approved final guidance of MRP-24 into the RI-ISI program for assessing TASCS.

Question 4: *In Section 3.5.2, Program Relief Requests, the licensee stated in note 2 to the relief request of RI-20, Rev. 1 that the subject Relief Request can be modified or withdrawn dependent upon the results of the upcoming examination. The staff notes that the subject relief request addresses the issue pertains to partial surface* *examination coverage of weld RVD-BF-14 in the ISI program. In the RI-ISI program this weld is selected for volumetric examination instead of surface examination. Therefore, this relief request should be withdrawn because it is no longer applicable to the inspection of weld RVD-BF-14 in the RI-ISI program. A separate relief request for volumetric examination of this weld should be submitted when needed.*

Response: NPPD agrees that the subject relief request should be withdrawn after NRC approval of the RI-ISI relief request, with a new relief request submitted for RVD-BF-14, if necessary.

Question 5: *In Table 3.3, Failure Potential Assessment Summary, intergranular stress corrosion cracking (IGSCC) is identified as a potential failure mechanism in 6 elements of the nuclear boiler (NB) system. In Table 3.5, those elements susceptible to IGSCC are assigned to Category 4 or 6 (for elements with no degradation mechanism). Discuss what method will be used for inspecting those elements in Category 4 that are susceptible to IGSCC. In addition, in note 2 to Table 3.5, it is stated that one of the augmented inspected (IGSCC) welds is being credited for RI-ISI program. Provide reason and justification for allowing such a credit.*

Response: Of the six welds in the NB system, five are classified as Category A locations per the plant's Generic Letter (GL) 88-01 Program, and the remaining location (Control Rod Drive (CRD) return line nozzle cap weld) is classified as Category D. Four of the five 88-01 Category A welds are classified as Risk Category 4 locations for RI-ISI purposes, based on a high consequence ranking and low failure potential. The CRD return line nozzle cap weld is Risk Category 4 (2). Per EPRI TR-112657, Rev. B-A (Section 2.4), the examination of welds identified as Category A inspection locations is subsumed by the RI-ISI Program. These welds are treated like any other Risk Category 4 location provided no other damage mechanisms are present, and are subject to the same volumetric examination.

In regard to Note 2 of Table 3.5 (Note 3 to Tables 5-1 and 5-2 are similar), the NRC has previously accepted crediting the augmented inspection program examinations to satisfy EPRI TR-112657 selection requirements[1]. In this case, the examination performed on the Category D CRD return line nozzle cap weld for the CNS GL 88-01 Program, is credited to meet the selection requirement for Risk Category 4 (2) and one of the four welds was selected to meet the requirements for Risk Category 4.

1. The use of augmented inspection program examinations to meet EPRI TR-112657 selection requirements is described in a letter to the NRC from J. Knubel (New York Power Authority), dated May 8, 2000, "Revised Risk-Informed Inservice Inspection (RI-ISI) Program." This position was accepted by the NRC as documented on Page 4 of the RI-ISI Safety Evaluation "Risk-Informed Inservice Inspection Program James A Fitzpatrick Power Plant," dated September 12, 2000.

Question 6: *In Table 3.3, many plant systems did not have any potential failure mechanism. This is consistent with Table 3.4, which shows that the majority of the elements selected for examination are in Category 4. Provide detailed discussion regarding how the elements in Category 4 are selected for inspection and what examination method will be used for each selected element.*

Response: Per Risk Category 4 requirements, a 10% sampling of the inspection locations was selected for examination in each of the applicable systems. It should be noted that in the NB system, a 10% sampling was selected for examination in both Risk Category 4 (2) and Risk Category 4. This resulted in the only Risk Category 4 (2) location (CRD return line nozzle cap weld that is classified as Category D per the plant's GL 88-01 Program) being selected for examination, as well as one of the four Risk Category 4 locations.

The Risk Category 4 selections were distributed among representative structural discontinuities in each system factoring in worker exposure concerns and access considerations. A volumetric examination will be performed in all cases.

Question 7: *In Table 3.3, crevice corrosion is identified as the only potential failure mechanism in reactor recirculation system and core spray system. Discuss what inspection method will be used for detecting this failure mechanism including qualification/demonstration of the inspection method and personnel.*

Response: Section 4 of EPRI TR-112657, Rev. B-A provides guidance on examination volumes and methods and generally recommends ultrasonic examination as the inspection method of choice. In particular, Section 4.2.2 provides typical configurations and examination volumes for locations potentially susceptible to crevice corrosion (CC) cracking. None of the creviced locations in the program are scheduled for examination in Refueling Outage 22. CNS is following the development of appropriate examination techniques for crevices. Prior to implementing the specific examination, CNS will ensure that the vendor's examination procedures and qualifications will reliably detect crevice corrosion for the specific configurations present at CNS.

Question 8: *In Table 3.3, IGSCC is identified as a potential failure mechanism only in the NB system. Discuss and provide reasons why stainless steel components in other systems are not considered susceptible to IGSCC. Even for Category A welds which are subsumed by the RI-ISI program should be considered as susceptible to IGSCC, Category A welds are more resistant to IGSCC; however, they are not immune to IGSCC.*

Response: CNS implemented a major piping replacement project in 1985 in response to IGSCC concerns. Replacement piping was installed in the susceptible systems with the necessary material properties (e.g., low carbon content) as to render them resistant (Generic Letter 88-01 Category A) to IGSCC.

Certain dissimilar metal nozzle-to-safe end welds at CNS have Alloy 182 buttering with Alloy 82 corrosion resistant cladding (CRC) and Induction Heating Stress Improvement (IHSI). Per NUREG-0313 Rev. 2, section 2.1.1 (3), this configuration satisfies the criteria for Category A.

With the exception of the CRD return line nozzle cap weld in the NB system that is classified as Category D per the CNS GL 88-01 Program, the other stainless steel welds are classified as Category A. In accordance with EPRI TR-112657 Rev. B-A (Section 2.4), stainless steel piping welds identified as Category A are considered resistant to IGSCC and are assigned a low failure potential provided no other damage mechanisms are present. As such, the examination of welds identified as Category A inspection locations is subsumed by the RI-ISI Program. In these cases, IGSCC is not assigned as a damage mechanism for RI-ISI purposes.

Question 9: *Describe in detail how the assessment of potential failure mechanisms for various systems as provided in Table 3.3 was performed, and also identify all deviations from the approved guidelines in EPRI TR-112657. The staff notes that the potential failure mechanisms identified for systems at CNS are substantially less than that at similar boiling water reactors.*

Response: Potential failure mechanisms were assessed for the various systems as provided in Table 3.3 by a thorough review of relevant plant documentation combined with communication with utility personnel. Piping class boundaries were identified from system flow diagrams. Information on piping dimensions and materials was obtained from the CNS ISI weld database. Operating temperatures were obtained from isometric drawings and piping specifications. Piping geometry was obtained from isometric drawings. Normal and upset operating conditions for the systems were evaluated from design criteria documents, plant operating procedures, and correspondence with plant technical personnel. Water chemistry for fluid sources was obtained from the CNS chemistry procedure. Insulation information was obtained from the thermal insulation specification. Susceptibility to IGSCC and flow-accelerated corrosion (FAC) was determined in accordance with the plant's Generic Letter 88-01 and FAC Program documents. The only deviation from the approved guidelines in EPRI TR-112657 was taken with respect to TASCS, as discussed in the response to Question 3 above.

Comparing the potential failure mechanisms at CNS to previous BWR applications performed at Hope Creek, Monticello, Duane Arnold, Fermi, Brunswick, Pilgrim and Perry yields the following insights:

The primary degradation mechanisms found in BWRs are IGSCC, FAC, CC, TASCS and thermal transients (TT). For IGSCC, element susceptibility is determined based upon the category assigned in the plant's GL 88-01 Program. At CNS, the welds in the plant's Generic Letter 88-01 Program are classified as Category A with the exception of the CRD return line nozzle cap weld in the NB system that is classified as Category D. Therefore, CNS has a much lower number of non-Category A locations than most previous applications. In accordance with EPRI TR-112657, stainless steel piping welds identified as Category A are considered resistant to IGSCC and are assigned a low failure potential provided no other damage mechanisms are present. As such, the examination of welds identified as Category A inspection locations is subsumed by the RI-ISI Program. FAC susceptibility is determined based upon components currently being monitored under the plant's FAC Program. The review shows that CNS has a comparable number of FAC susceptible locations to previous applications. The review also shows that CNS has a comparable number of CC susceptible locations (thermal sleeves located in oxygenated fluid at high temperature) to previous applications. The Main Steam and Feedwater piping affected by TT is also comparable to previous applications (note that some piping classified as High Pressure Coolant Injection system at other plants is classified as part of the Main Steam system at CNS). The Feedwater system at CNS is not affected by TASCS due to the high initial flow rate used per plant operational procedures. TT is not a problem in the Residual Heat Removal (RHR) system at CNS (as it is at some other plants) since this system is pre-heated prior to shutdown cooling operations. This pre-heating procedure also precludes TT in the RHR system from shutdown cooling return flow (a problem at some other plants).

For the reasons stated (primarily the large number of GL 88-01 Category A IGSCC locations), it is reasonable that CNS has fewer elements susceptible to the failure mechanisms evaluated than previous applications.

Should you have any questions concerning this matter, please contact Mr. Paul Fleming at (402) 825-2774.

Sincerely,

Randall K. Edington

Randall K. Edington
Vice President - Nuclear and
Chief Nuclear Officer
/wrv

Attachment

cc: Regional Administrator w/attachment
USNRC - Region IV

Senior Project Manager w/attachment
USNRC - NRR Project Directorate IV-1

Senior Resident Inspector w/attachment
USNRC

NPG Distribution w/o attachment

Records w/attachment

ATTACHMENT 1
RELIEF REQUEST NUMBER: RI-34 Revision 1

COMPONENT IDENTIFICATION

Code Classes:	1 and 2
References:	IWB-2500, IWC-2500, Table IWB-2500-1, Table IWC-2500-1
Examination Categories:	B-F, B-J, C-F-1, and C-F-2
Item Numbers:	B5.10, B5.20, B5.130, B5.140, B9.1 0, B9.20, B9.30, B9.40, C5.50, and C5.80.
Description:	Risk-Informed Inservice Inspection (RI-ISI).
Component Numbers:	All Class 1 and 2 pressure retaining piping welds

APPLICABLE CODE EDITION AND ADDENDA

1989 Edition, No Addenda

CODE REQUIREMENT

ASME Section XI (1989 Edition), IWB-2500 (a) states:

> Components shall be examined and tested as specified in Table IWB-2500-1. The method of examination for the components and parts of the pressure retaining boundaries shall comply with those tabulated in Table IWB-2500-1 except where alternate examination methods are used that meet the requirements of IWA-2240.

Table IWB-2500-1, Categories B-F and B-J requires 100% and 25% respectively of the total number of non-exempt welds.

ASME Section XI (1989 Edition), IWC-2500 (a) states:

> Components shall be examined and pressure tested as specified in Table IWC-2500-1. The method of examination for the components and parts of the pressure retaining boundaries shall comply with those tabulated in Table IWC-2500-1, except where alternate examination methods are used that meet the requirements of IWA-2240.

Table IWC-2500-1, Categories C-F-1 and C-F-2 require 7.5%, but not less than 28 welds to be selected for examination. Note- Cooper Nuclear Station (CNS) does not have any Category C-F-1 welds.

In addition, both Tables (IWB-2500-1 and IWC-2500-1) reference figures that convey the examination volume for each configuration that could be encountered.

BASIS FOR RELIEF

The scope for ASME Section XI Inservice Inspection (ISI) programs is largely based on deterministic results contained in design stress reports. These reports are normally very conservative and may not be an accurate representation of failure potential. Service experience has shown that failures are due to either corrosion or fatigue and typically occur in areas not included in the plant's ISI program. Consequently, nuclear plants are devoting significant resources to inspection programs that provide minimum benefit.

As an alternative, significant industry attention has been devoted to the application of risk-informed selection criteria in order to determine the scope of ISI programs at nuclear power plants. Electric Power Research Institute (EPRI) studies indicate that the application of these techniques will allow operating nuclear plants to reduce the examination scope of current ISI programs by as much as 60% to 80%, significantly reduce costs, and continue to maintain high nuclear plant safety standards.

NPPD has applied the methodology of EPRI Topical Report TR-112657 in the development of the proposed CNS RI-ISI Program (see Enclosure 1 to this Attachment). The RI-ISI application was also conducted in a manner consistent with ASME Code Case N-578 "Risk-Informed Requirements for Class 1, 2, and 3 Piping, Method B." The use of this methodology for the selection and subsequent examination of Class 1 and 2 piping welds will provide an acceptable level of quality and safety.

Relief is requested in accordance with 10CFR50.55a(a)(3)(i). The Nuclear Regulatory Commission has previously approved several RI-ISI Programs based on methodology contained in EPRI Topical Report TR-112657, Revision B-A. A similar RI-ISI submittal has been recently approved for Salem, Units 1 and 2.[2]

PROPOSED ALTERNATE PROVISIONS

As an alternative to existing ASME Section XI requirements for piping weld selection and examination volumes, NPPD will implement the alternative RI-ISI program described in Enclosure 1.

APPLICABLE TIME PERIOD

Approval of this alternative is requested for the remainder of the third ten-year interval of the ISI Program for CNS, beginning with the last outage (RFO 22) of the third period.

2. Letter from J. Clifford (NRC) to R. Anderson (PSEG Nuclear), dated October 1, 2003, TAC NOS. MB7537 and MB7538).

Correspondence Number: NLS2004091

The following table identifies those actions committed to by Nebraska Public Power District (NPPD) in this document. Any other actions discussed in the submittal represent intended or planned actions by NPPD. They are described for information only and are not regulatory commitments. Please notify the Licensing & Regulatory Affairs Manager at Cooper Nuclear Station of any questions regarding this document or any associated regulatory commitments.

COMMITMENT	COMMITTED DATE OR OUTAGE
CNS will incorporate the applicable NRC-approved final guidance of MRP-24 into the RI-ISI program for assessing TASCS.	Within 12 months of NRC acceptance of MRP-24.