50-298

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

May 9, 1997

Mr. G. R. Horn
Sr. Vice President of Energy Supply
Nebraska Public Power District
1414 15th Street
Columbus, NE 68601

SUBJECT: COOPER NUCLEAR STATION - EVALUATION OF CORE SPRAY PIPING INDICATIONS DURING REFUELING OUTAGE 17 (TAC NO. M95141)

Dear Mr. Horn:

By letter dated May 7, 1997, the Nebraska Public Power District (NPPD) provided the staff with the results of your re-examination of three indications on the core spray piping at the Cooper Nuclear Station (CNS). The three indications had been previously identified during the 1995 refueling outage. By letter dated December 21, 1995, the Nuclear Regulatory Commission (NRC) staff approved your flaw evaluations of those indications, and also approved plant operation for one fuel cycle without repair of the subject flaws. In that letter, the staff stipulated that continued operation beyond one cycle should be supported by re-inspection and re-evaluation of the indications in accordance with the requirements of American Society of Mechanical Engineers (ASME) Code, Section XI, 1989 Edition.

By letter dated November 27, 1996, NPPD notified the NRC that future inspections of the core spray spargers and piping would be performed in accordance with the Boiling Water Reactor Vessel Internals Project guidelines contained in EPRI Report TR-106740, dated July 1996, "BWR Vessel and Internals Project, Core Spray Internals Inspection and Flaw Evaluation Guidelines (BWRVIP-18)." The visual and ultrasonic (UT) re-examinations of the previously identified flaws were performed in accordance with this guidance during the current refueling outage (RFO-17).

In your letter of May 7, 1997, you reported the following results:

Weld Number	Length - 1995 results (in.)	Length - 1997 results (in.)	Projected length at end of next cycle (in.)	Maximum Allowable flaw size
A1	8.9	9.1	10.3	11.8
A21	5.5	5.6	6.8	11 8
B12	1.5	NA	NA	10.7

I/O
DFO1

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In 1995, the apparent indication in the area of the D12 weld was based on a visual examination, which could not be confirmed by UT examination at the time. Your re-examination of that area during RFO-17 failed to identify any flaws, and you have concluded that the previously identified indication in that area is non-relevant. You further concluded that the slight increases reported in the lengths of the indications seen in the areas of the A1 and A21 welds may have resulted from differences in UT examination methods, or from crack growth. If the increases are attributed to crack growth, you concluded that the actual growth rate has been significantly less than the bounding rate assumed in your previous analyses provided in your letters of November 22 and December 18, 1995. You further concluded that the fracture mechanics evaluation submitted in those letters is still applicable, based on the absence of significant growth in the indications, and that the Code allowable flaw sizes will not be exceeded during an additional cycle of operation. You also committed to re-examine the two relevant indications using the BWRVIP-18 guidance at the next refueling outage.

The NRC staff has reviewed your letter of May 7, 1997, and concludes that the previous analyses and assumptions accepted in our letter dated December 21, 1995, are applicable to the current indications, due to the limited crack growth observed over the past cycle. Therefore, we conclude that the flaw sizes at the end of the next fuel cycle will not exceed the critical flaw sizes calculated using the methodologies and the acceptance criteria provided in IWB-3640 and Appendix C of the ASME Boiler and Pressure Vessel Code, Section XI, and that CNS can be safely operated for an additional fuel cycle without repair of the flaws in the areas of the A1 and A21 welds. Plant operation beyond the next fuel cycle should again be supported by the results of re-inspection and re-evaluation of the indications, consistent with the commitments in your May 7, 1997, letter.

In our letter of December 21, 1995, the staff requested you to provide an evaluation of the potential effects of reduced core spray flow due to leakage through the identified cracks, which were conservatively assumed to be through-wall. In addition, you were requested to evaluate the potential impact of loose parts resulting from a break of the core spray line at the identified crack locations. The staff has reviewed your evaluation of those potential effects, provided by letter dated March 29, 1996, and has concluded that these effects would result in a negligible increase in the peak cladding temperature calculated for the Cooper Nuclear Station. However, this review was based on the crack lengths reported in 1995 and on the Maximum Average Planar Linear Heat Generation Rates for the past fuel cycle. You are therefore requested to submit for staff review an updated analysis to confirm that these conclusions are valid for the next fuel cycle. This submittal is requested within 3 months of plant restart from the current refueling outage.

This completes the NRC staff review for TAC No. M95141. If you have any questions regarding this issue, please contact me at (301) 415-1336.

Sincerely,

James R. Hall

James R. Hall, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

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cc: See next page

This completes the NRC staff review for TAC No. M95141. If you have any questions regarding this issue, please contact me at (301) 415-1330.

Sincerely,

ORIGINAL SIGNED BY:

James R. Hall, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

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Cooper Nuclear Station

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