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Nebraska Public Power District

NLS960007
March 29, 1996

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

Subject: Impact of Core Spray Line Crack Indications
Cooper Nuclear Station, NRC Docket No. 50-298, License No. DPR-46

References:
1. Letter (No. NLS950244) to USNRC Document Control Desk from J. H. Mueller (NPPD) dated December 18, 1995: Follow-up Information to IE Bulletin 80-13 Response; Visual Inspection of Core Spray Spargers
2. Letter to G. R. Horn (NPPD) from J. R. Hall (US NRC) dated December 21, 1995: Cooper Nuclear Station - Evaluation of Core Spray Piping Indications (TAC No. M94097)

Gentlemen:

The purpose of this letter is to submit to the Nuclear Regulatory Commission (NRC) additional information regarding the potential impact of postulated cracks in the Core Spray lines for Cooper Nuclear Station (CNS). The Nebraska Public Power District (District) committed, by letter dated December 18, 1995 (Reference 1) to provide this information as a follow-up to the NRC's Safety Evaluation Report (Reference 2). The District committed in Reference 1 to forward this information to the NRC by February 8, 1996. However, as discussed with and agreed to by the NRC CNS Senior Project Manager, submittal of this letter by the end of March 1996 was acceptable in order to address minor technical issues.

The District has completed a review of the effect of potential leakage through crack indications which have been observed in both the "A" and "B" core spray lines. The review assumes a through wall crack with a length equal to the maximum predicted cycle 17 growth and a width of 10 mils. The review also considered the unrelated leakage from the T-box vent holes associated with the core spray piping/core shroud penetrations. There is one vent hole for each T-box and one T-box for each core spray line. The total leakage flow rate for the two T-box vent holes combined is conservatively estimated to be 20.2 gpm.

For the flaws in lines "A" and "B," the total potential leakage rate has been calculated to be 40.2 gpm. For line "A," the calculated potential leakage rate consists of 20.8 gpm for sleeve weld #1 and 13.8 gpm for sleeve weld #21. For line "B," the calculated potential leakage rate consists of 5.6 gpm from T-box weld #12. With the addition of the vent hole flows (20.2 gpm), the total leakage rate for both loops sums up to 60.4 gpm. This translates to less than 1.5% of total core spray flow rate.

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A change of 10% core spray flow during the worst case Design Basis Analysis (DBA) Loss of Coolant Accident (LOCA), which consists of an 80% recirculation line break, is conservatively estimated to yield a 20 ^{0}F change in Peak Clad Temperature (PCT). A change of 1.5% core spray flow is conservatively estimated to yield an increase of 3 ^{0}F PCT during the DBA LOCA. For CNS, the thermal-mechanical Maximum Average Planer Linear Heat Generation Rates (MAPLHGR) bound the DBA LOCA. The design PCT for the most limiting DBA LOCA is 2,200 ^{0}F LOCA at an MAPLHGR of approximately 13.8 kilowatts per foot (kW/ft). For Cycle 17, the limiting value MAPLHGR is 12.93 kW/ft, which conservatively translates to a PCT value of approximately 2,150 ^{0}F. When adding the additional 3 ^{0}F to the PCT range for the current cycle, there is considerable margin between the resultant PCT and the design PCT. Therefore, the evaluation indicates that the postulated leakage would not adversely affect core spray performance and that such leakage would not affect plant operating limits.

The District has performed an analysis to evaluate the safety concerns associated with a postulated piping failure which results in loose piping parts falling into the vessel annulus region. The analysis evaluated four major concerns associated with loose parts: The potential for fuel bundle flow blockage and consequent fuel damage, the potential for fretting wear of fuel cladding, the potential for interference with control rod operation, and the potential for corrosion or chemical reaction with other reactor materials.

The evaluation results indicate that a single 360 degree (full circumferential) through-wall crack would not result in loose pieces and the sparger would remain attached to the shroud. Further, in the unlikely event that multiple failures were to occur and a piece was to break loose, the evaluation concludes that the loose part would result in no significant fuel bundle flow blockage, no potential for interference with control rod operation and no potential for corrosion or adverse chemical reaction with other reactor materials. As for the potential for fretting wear of fuel cladding, possible fuel rod leaks would be detected by the off-gas system so that appropriate action can be taken to maintain the off-site radiation release within acceptable limits. Any such cladding damage would only be an operational concern and not a safety concern to the magnitude of the other three concerns identified above.

If you have any questions, or require any additional information, please contact me.

Sincerely,

John H. Mueller

John H. Mueller
Site Manager

/dnm
Attachment

cc: Senior Project Manager
USNRC - NRR Project Directorate IV-1

Senior Resident Inspector
USNRC - Cooper Nuclear Station

Regional Administrator
USNRC - Region IV

NPG Distribution

Correspondence No: NLS960007

The following table identifies those actions committed to by the District in this document. Any other actions discussed in the submittal represent intended or planned actions by the District. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify the Licensing Manager at Cooper Nuclear Station of any questions regarding this document or any associated regulatory commitments.

COMMITMENT	COMMITTED DATE OR OUTAGE
None	N/A