Nebraska Public Power District

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September 22, 1981

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U.S. NUCLEAR REGULATORY COMMISSION

Mr. Thomas M. Novak, Assistant Director
for Operating Reactors
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Response to Unresolved Safety Issue A-10, BWR Nozzle Cracking

Dear Mr. Novak:

By letter dated July 15, 1981, you reviewed our commitments concerning the subject issue and requested additional justification for our position. The following discussion and analysis is being provided as that justification.

An examination of recent CNS plant startups for variance in the FW nozzle temperature between loops receiving RWCU flow and loops not receiving RWCU flow was performed. Two FW nozzles are monitored for temperatures, one nozzle receiving cleanup flow and one nozzle not receiving cleanup flow. A maximum temperature variance of less than 10°F between FW loops and an average temperature difference of less than 5°F between FW loops was observed throughout the startups. These temperatures are measured by RTD's located on B and D FW nozzles adjacent to the safe end to nozzle weld.

CNS has a Westinghouse Turbine Generator which has a different turbine control system and considerably different startup characteristics than a General Electric Turbine Generator. The Digital Electro Hydraulic Control System used for turbine control is programmed by the operators to bypass to the condenser significant amounts of steam early in the reactor startup and thus provide a steady and stable reactor pressure increase. With steam being bypassed to the condenser and with reactor temperature and pressure increasing, FW flow becomes very large as compared to RWCU flow. This operational characteristic contributes to the small temperature difference between the FW loops.

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An additional contribution to the small temperature difference in FW loops is found in the FW control system. During all startups FW flow is controlled by the FW startup flow control valves. Throughout most of the startup the FW startup flow control valves are operated in an automatic mode to maintain a constant reactor vessel level and a stable FW flow. The FW heaters are cascading type heaters and are fully operational during startup. Normal FW heating does not commence until extraction steam is established and the turbine roll is started. All of the above plant characteristics and operational methods contribute to the small FW nozzle temperature difference.

Rerouting the RWCU System to all Feedwater (FW) nozzles will require installation of a crosstie line between the RWCU and the HPCI Systems in the steam tunnel. A check valve would be needed in this crosstie line to maintain the double valve integrity of the Primary Containment.

All piping and valves would be Class I N (Nuclear). A seismic and hanger analysis of the crosstie line to determine loadings, hanger and restraint location would be required. This analysis should also determine any effect the crosstie would have on the HPCI System. A thermal analysis of the (4 inch) RWCU crosstie line to the (14 inch) HPCI discharge line would be necessary. A thermal sleeve or mixing tee may be needed to reduce the thermal stress on the HPCI piping. Additional inservice inspection (NDE) of the HPCI and the crosstie line may be necessary. Although the basic design change is relatively simple, requiring one check valve and approximately 30 ft. of pipe, the seismic analysis, the hanger analysis and design, the thermal analysis, and procurement of the Class I Nuclear piping and components results in a quite costly design change. In our opinion, this design change is not justified by the small benefit that would be realized from rerouting RWCU to both FW loops. As previously stated, recent startups have shown a maximum temperature difference of less than 10°F throughout the startup for a FW nozzle receiving RWCU flow and a FW nozzle not receiving RWCU flow.

In NUREG 0619, the NRC expressed concerns to the Utilities and to General Electric that the elimination of the CRD return line to the vessel may potentially degrade the total plant capability to supply water to the vessel during emergency conditions. Even though this capability may not have been required as part of the original plant design, a satisfactory equivalent capability must be retained by the facility. A generic analysis of BWR CRD Systems performed by General Electric revealed that for Cooper Station the decay and residual heat from the fuel after a scram would result in a boil off from the core of approximately 135 gpm. NUREG 0619 required all facilities to test the CRD System flow capability to attain a flow greater than the decay and residual heat boil off rate.

Special Test Procedure 81-1 was performed during the 1981 Refueling and Maintenance Outage to demonstrate the CRD System flow capacity. With two (2) CRD pumps in operation, a flow capacity in excess of 160 gpm was achieved at reactor pressure above 1000 psig. With one CRD pump in operation, a flow capacity of 140 gpm was achieved at reactor pressure of 990 psig. This result demonstrates that the CRD System retains the flow capability to supply water to the vessel in excess of the flow required by NUREG 0619. Special Test Procedure 81-1 is available for examination at the plant site.

Presently there are no plans to reroute the CRD return line back to the reactor vessel through the RWCU or FW Systems. The CRD System has demonstrated good and reliable operation and STP 81-1 demonstrated flow capacity in excess of NUREG 0619 requirements. Rerouting would require an expensive and significant design change to the CRD System and is not justified at Cooper Station.

In regards to Paragraph 8.2 of NUREG 0619, only items in Section 8.2(4) are applicable to Cooper Station and the following responses are provided:

8.2(4) Licensees and applicants who choose to cut and cap the CRDRL nozzle without rerouting of the CRDRL: In addition to the final PT of the nozzle, the return flow capacity demonstration and the post-modification CRD-system-performance test, the following requirement must be met:

The CRD system modifications of Section 8.1(4)(a') through (4)(c') must be accomplished and plant maintenance procedures must be changed to include flushing the normal drive-movement exhaust-water header and cleaning the filters in the insert and exhaust lines if carbon steel piping is retained. These filters are to be retained in the HCU to prevent corrosion products from being carried into the CRD mechanisms.

8.1(4) (a') Equalizing valves between the cooling water header and the normal drive movement exhaust water header.

(b') Flush ports at high and low points of the normal drive movement exhaust water header piping run if carbon steel piping is retained.

(c') Replacement of carbon steel pipe in flow stabilizer loop with stainless steel and rerouting directly to the cooling-water header.

Response:

Cooper Station cut and capped the CRDRL nozzle in 1978. A PT of the nozzle and vessel wall was satisfactorily completed. Since Cooper Station cut and capped the CRDRL nozzle in 1978 prior to the restrictions of NUREG 0619, a pre-modification CRD system performance test was performed. A return flow capacity test was completed during the 1981 Refueling and Maintenance Outage.

Equalizing valves have been installed between the cooling water header and the normal drive movement exhaust water header. All carbon steel pipe was replaced with stainless steel pipe in areas of reverse flow so flush ports are not required. The carbon steel pipe in the flow stabilizer loop was replaced with stainless steel and is routed to the cooling water header.

If there are any questions concerning this response, please contact P. Ballinger at Cooper Nuclear Station.

Sincerely,

Jay M. Pilant

Jay M. Pilant
Division Manager of Licensing
and Quality Assurance

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