Omaha Public Power District

1623 HARNEY OMAHA, NEBRASKA 68102 TELEPHONE 536-4000 AREA CODE 402

June 26, 1979

Director of Nuclear Reactor Regulation
ATTN: Mr. Robert W. Reid, Chief
Operating Reactors Branch No. 4
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Reference: Docket No. 50-285

Gentlemen:

In response to discussions held with members of your staff, in connection with the review of the Omaha Public Power District's inservice inspection plan, the following justification provides a basis for ensuring the leak tightness of valves HCV-347, HCV-348, SI-196, SI-195, SI-194, SI-207, SI-208, SI-199, SI-198, SI-197, SI-211, SI-212, SI-202, SI-201, SI-200, SI-216, SI-215, SI-205, SI-204, SI-203, SI-220, and SI-219 at the Fort Calhoun Station.

On the shutdown cooling line from the primary system, any leakage past HCV-348 would create a pressure buildup until safety valve SI-188 lifted (setpoint = 2000 psig). The lifting of the safety valve would be detected in an increase in both temperature and level in the pressurizer quench tank. Leakage through HCV-348 and HCV-347 would be detected in the daily Surveillance Test RLT-3, Reactor Coolant System Leak Rate Test, which monitors the reactor coolant system inventory. Technical Specification 2.1.4(1) states that, if the reactor coolant system leakage exceeds 1 gpm, the source must be identified within 12 hours or the reactor shall be placed in the hot shutdown condition. If the source is not located within 24 hours, the reactor shall be placed in the cold shutdown condition. Technical Specification 2.1.4(2) further states that the reactor shall be placed in the hot shutdown condition within 12 hours if the leakage exceeds 10 gpm. The reactor shall be placed in the cold shutdown condition if the leakage exceeds 10 gpm for 24 hours.

Leakage from the primary reactor coolant system through the primary check valves, SI-208 (SI-212, SI-216, or SI-220), will cause a pressure buildup in the upstream piping until it reaches 440 psig. At that time, the pressure control valve, PCV-2949 (PCV-2969, PCV-2929, or PCV-2909), will open, routing the leakage through a leakage cooler and flow element, and then to the chemical and volume control system or waste disposal system. The leakage would be detected in performing the daily Reactor Coolant System Leak Rate Test as described above, and by the flow indicating device.

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Leakage through the SI tank outlet check valve, SI-207 (SI-211, SI-215, or SI-219), would require a pressure differential on the valve caused by leakage through the primary check valve. Leakage in this case would cause a pressure and water level rise in the corresponding SI tank, SI-6C (SI-6D, SI-6B, or SI-6A), and actuate an alarm in the control room.

Leakage through any of the high or low pressure SI lines check valves, SI-194 (SI-197, SI-200, or SI-203), SI-195 (SI-198, SI-201, or SI-204), or SI-196 (SI-199, SI-202, or SI-205), would be identified by a loss of level in the corresponding SI tank. A low level alarm would actuate in the control room.

If non of the above conditions exist, leakage through these isolation valves is not occurring. These conditions are adequate to monitor leakage past the subject valves and, therefore, leak testing in accordance with ASME Section XI is not believed to be necessary.

Sincerely,

W. C. Andrews

for T. E. Short
Assistant General Manager

TES/KJM/BJH:jmm

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