Omaha Public Power District

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September 3, 1982
LIC-82-318

Mr. Robert A. Clark, Chief
U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Division of Licensing
Operating Reactors Branch No. 3
Washington, D.C. 20555

Reference: Docket No. 50-285

Dear Mr. Clark:

10 CFR 50, Appendix J
Containment Leakage Testing Program

The Commission's letter to the Omaha Public Power District, dated July 23, 1982, transmitted the Commission's safety evaluation report (SER) regarding the containment leakage testing program at the Fort Calhoun Station. The SER identified three open items that are to be resolved prior to issuance of the District's license amendment. The District's response to these open items is provided below.

The first open item concerns the leak testing of the containment personnel access door seals and, specifically, the methodology for extrapolating the leakage results determined at the 5 psig test pressure to a leakage rate equivalent to a test at 60 psig. Appendix A to the Commission's letter dated July 23, 1982 details an extrapolation methodology, different from that proposed by the District, that is acceptable to the Commission for this purpose. The District has reviewed the Appendix A methodology and has determined that in order to make an accurate assessment of the acceptability of the criteria for the Fort Calhoun Station, further detailed analysis is required. Additionally, the District has requested copies of three of the references identified in Appendix A from our Project Manager to supplement the District's review and analysis of this methodology. Pending further review of Appendix A and receipt of the requested information, the District expects to provide the Commission with a response to this item by November 1, 1982.

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The other two open items concern Type C testing of the isolation valves associated with containment mechanical penetrations M-3 and M-44. The isolation valves associated with penetration M-3, the charging pump discharge line, will not be tested in accordance with Appendix J, Type C testing, as justified in the District's letter to the Commission dated May 15, 1980 and the current Technical Specifications. These valves and associated piping are part of a closed system which is connected to the reactor coolant system and the valves do not automatically close upon receipt of an accident signal. However, the principal justification for not leak testing the M-3 isolation valves is the fact that the pressure seen by these valves in the direction of flow toward the containment building will always be much greater than the maximum design basis accident (i.e., LOCA) pressure in containment of approximately 60 psig. Since all charging pumps are automatically started upon receipt of a Safety Injection Actuation Signal (SIAS) and produce a normal discharge pressure of approximately 2100 psig in the subject line, the leakage of contaminated containment air through this path is precluded.

Concerning the testing of the isolation valves associated with penetration M-44, the "Pressurizer Dead Weight Tester," the District is presently evaluating the possibility of capping this 1/8" instrument line inside the containment. The District no longer utilizes this instrument line, thus capping the line would preclude the necessity of leak testing the penetration isolation valves. The results of this evaluation and the schedule for completing the work, if applicable, will be provided to the Commission with the District's response to the first open item discussed above.

Sincerely,

W. C. Jones
Division Manager
Production Operations

WCJ/TLP:jmm

cc: LeBoeuf, Lamb, Leiby & MacRae
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