July 23, 1985

Docket No. 50-285

Mr. R. L. Andrews
Division Manager - Nuclear Production
Omaha Public Power District
1623 Harney Street
Omaha, Nebraska 68102

Dear Mr. Andrews:

In conducting our review of your April 1, 1982, July 1, 1982, December 30, 1982 and August 2, 1983 submittals relating to Performance Testing of Relief and Safety Valves, NUREG-0737, Item II.D.1, at the Fort Calhoun Station, Unit No. 1, we have determined that we will need the additional information identified in the enclosure to continue our review.

In order for us to maintain our review schedule, your response is requested within 30 days of your receipt of this letter. If you cannot respond in the time frame requested, please contact your project manager to negotiate a mutually agreeable response date.

The reporting and/or recordkeeping requirements contained in this letter affect fewer than ten respondents; therefore, OMB clearance is not required under P.L. 96-511.

Please contact the NRC project manager, E. G. Tourigny, if you have any questions concerning this request.

Sincerely,

Edward J. Butcher, Acting Chief
Operating Reactors Branch No. 3
Division of Licensing

Enclosure:
Request for
Additional Information

cc w/enclosure:
See next page

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Mr. R. L. Andrews
Omaha Public Power District

Fort Calhoun Station
Unit No. 1

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REQUEST FOR ADDITIONAL INFORMATION

TMI ACTION NUREG-0737 (II.D.1)

FOR

FORT CALHOUN

DOCKET NO.: 50-285

JUNE 1985

SAFETY EVALUATION QUESTIONS
TMI ACTION NUREG-0737 II.D.1
FOR FORT CALHOUN

Questions related to the selection of transients and valve inlet conditions:

1. The Combustion Engineering Report on operability of PORVs in CE Plants indicated that the limiting inlet fluid conditions during low temperature pressurization transients is a water discharge event. The CE Inlet Fluid Conditions Report stated that the pressurizer water solid condition and resulting PORV liquid discharge case was chosen for the cold overpressurization event since it gave the most severe pressurization transients. The report further states that a steam bubble can also exist in the pressurizer during low temperature operation whereby the PORV could lift on steam. No low pressure steam tests were performed by EPRI on the Dresser PORV. Provide verification that the Fort Calhoun PORVs will operate satisfactorily on low pressure steam.

2. The Fort Calhoun submittal did not discuss the feedline break event. NUREG-0737 II.D.1 requires that the transients of Regulatory Guide 1.70 Revision 2 be considered. The feedline break is included in these transients. Discuss the feedline break event and state whether or not is is applicable to Fort Calhoun; or provide peak pressure, pressurization rate, temperature, discharge flow rate and expected fluid. Demonstrate safety and PORV functionability for this event, and consideration of this event in the discharge piping analysis.

Questions related to valve operability:

3. The Fort Calhoun nuclear plant utilizes Dresser 31533VX-30 PORV valves. The model number indicates that the valves contain the older obsolete internals. Most plants using this valve have upgraded their valves to the type 2 internals. The EPRI tests were conducted with

the type 2 internals. The EPRI PWR Safety and Relief Valve justification report indicates that as of August 1981 the licensee had not purchased the parts necessary to upgrade their valves to the type 2 internals. The manufacturer indicated that all plants using this valve are expected to make the modification. Since the EPRI tests were conducted with the type 2 internals, the licensee should either make the modification or justify that the tests demonstrate acceptable performance of the plant valve.

4. NUREG 0737, Item II.D.1 requires that the plant-specific PORV Control Circuitry be qualified for design-basis transients and accidents. Provide information which demonstrates that this requirement has been fulfilled.

5. The Safety Valve test data identified in the submittal as applicable to the Fort Calhoun plant are based on steam flow with loop seal internals in the safety valve and two acceptable ring settings. Provide verification that the Fort Calhoun Safety Valves are equipped with the loop seal internals and identify the ring settings used.

6. The December 30, 1982 submittal states that the PORV flange loads are less than those measured in the EPRI tests and, therefore, the valves are not expected to stick open. Provide confirming information that the calculated piping loads (post-modification) at the safety and PORV valve flanges are acceptable when compared to the EPRI test program loads.

   Thermal expansion of the pressurizer causing displacement of the piping nozzles and thermal expansion of the piping from the nozzles to the valves can contribute to the bending moment induced in the valve body. The submittal does not make clear what loads were considered in calculating the bending moments applied to the plant safety valves and PORVs. Provide additional discussion comparing the measured moment on the tested valves to the calculated induced moments from all effects including those described above on the plant specific valves. Verify that the bending moments would have no adverse effect on the operability of the plant valves.

7. Dresser Industries, the manufacturer of the Ft.Calhoun PORV, wrote a letter to Metropolitan Edison Co. in March 1976 warning that the PORV block valve should be kept closed when reactor coolant system pressure is below 1000 psig to avoid damaging the PORV disk and seat by steam wirecutting. The EPRI program data indicates that the Dresser PORV was successfully tested on water at pressures in the 500-900 psig range. Steam testing at lower pressures was not performed. Each EPRI test sequence was initiated with a valve where disk and seat were in excellent condition, which may not be representative of the condition of the Dresser PORV as routinelyplaced in service at Ft. Calhoun. The recommendation made by Dresser that the PORV be isolated at pressures lower than 1000 psi would seen to preclude the use of the PORV for low temperatures overpressure protection of the reactor vessel. Explain whether the Dresser recommendation or a modification of it will be followed to prevent damage to the disk and seat from steam wirecutting or provide details of tests performed since the March 1976 letter that demonstrate that such precautions are unnecessary.

Questions related to the block valves:

8. The submittal did not address operation of the PORV block valves. The EPRI test data do not include any test data for the 2-1/2 inch Crane gate valve or the Limitorque SMB-00-7.5 operator which according to the EPRI block Valve Report (R. C. Youngdahl Valve Package, June 1, 1982) is the combination used as a PORV block valve at the Fort Calhoun plant. NUREG-0737 Part II.D.1 states in part that each PWR licensee should provide evidence supported by test that the block or isolation valves between the pressurizer and each power-operated relief valve can be operated, closed, and opened for all fluid conditions expected under operating and accident conditions. Provide information on how this requirement is met.

Questions on thermohydraulic analysis:

9. The submittal does not identify the method or computer program used for thermohydraulic analysis or how the method or computer program was verified. To allow for an evaluation of the analysis, identify the method or computer program and how it was verified.

Identify the important parameters and the rationale for their selection. This should include a description of the method or computer program used to generate fluid pressures and moments over time and how the program or computer program calculates resulting fluid forces on the system. Fluid conditions assumed for the analysis should be provided such as: location and initial temperature of water in loop seal, peak pressure, pressurization ratio, back pressure, temperature, fluid range, and number and type of valves actuated.

Because the ASME Code requires derating of the safety valves to 90% of expected flow capacity, the safety valve analysis should be based on 111% of flow rating unless otherwise justified. Information should be provided explaining how the derating of the safety valves was handled and the method used to establish flow rates for the safety valves and PORVs in the analysis.

Questions on structural analysis:

10. The April 1, 1982 submittal states that a code comparable to RELAP5 would be used to verify the adequacy of the valve discharge piping. The December 30, 1982 submittal states that the evaluation of thermal and dynamic stresses on piping system and supports were completed. The conclusions of this evaluation were that the PORV inlet piping were within the applicable code limits but that the Safety Valve inlet piping may exceed the elastic limit during loop seal discharge unless pipe restraints were modified and that both the PORV and Safety valve discharge piping would be stressed beyond design valves upon valve actuation.

The August 2, 1983 submittal stated that several modifications were made to the PORV discharge piping and supports to reduce possible stresses to acceptable levels. It was also stated that an analysis was being made on the safety valve inlet and discharge piping and modifications were planned for completion during the 1984 refueling outage.

Provide verification that the as-modified piping and supports have been analyzed. Identify the analytical method used and explain how the method has been verified. Identify the multi-valve opening sequences used to produce the worst case loading on the piping and supports. Identify the load combinations considered in the analysis and the allowable stress limits used. Load combinations and acceptance criteria were recommended in the EPRI PWR Safety and Relief Valve Test Program Guide for application of Valve Test Program Results. If other load combinations and criteria are appropriate, the rationale for their selection should be provided.