Omaha Public Power District
1623 Harney Omaha, Nebraska 68102-2247
402/536-4000

March 1, 1986
LIC-86-083

Mr. Ashok C. Thadani, Project Director
PWR Project Directorate #8
Division of PWR Licensing - B
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, DC 20555

References: (1) Docket No. 50-285
(2) Letter OPPD to NRC dated April 1, 1982 (LIC-82-138)
(3) Letter OPPD to NRC dated July 1, 1982 (LIC-82-253)
(4) Letter OPPD to NRC dated December 30, 1982 (LIC-82-415)
(5) Letter OPPD to NRC dated August 2, 1983 (LIC-83-182)
(6) Letter NRC to OPPD dated July 23, 1985
(7) Letter OPPD to NRC dated September 3, 1985 (LIC-85-387)

Dear Mr. Thadani:

Additional Information on Performance Testing of Relief and Safety Valve Testing, NUREG-0737 Item II.D.1

The Omaha Public Power District (OPPD) received Reference 6, requesting additional information relating to the subject. As was noted in Reference 6, this information was to continue a review of References 2, 3, 4, and 5. Accordingly, please find attached OPPD's response to the Reference 6 questions. If you have additional questions concerning this issue, please do not hesitate to contact us.

Sincerely,

J K Gasper

R. L. Andrews for
Division Manger
Nuclear Production

RLA/DJM/me

cc: LeBoeuf, Lamb, Leiby & MacRae
1333 New Hampshire Ave., N.W.
Washington, DC 20036

E. G. Tourigny, NRC Project Manager
P. H. Harrell, NRC Senior Resident Inspector

A046
1/1

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 over-pressurization 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.*

OPPD RESPONSE TO QUESTION #1

OPPD PORV's are manufactured by Dresser. Low pressure tests with steam inlet were recently conducted by the manufacturer. Results of the tests verified satisfactory operation with inlet pressures above 100 psi. Below that pressure, the weight of some internal parts could not be overcome by the steam pressure. Cold over-pressurization trip setpoints for the PORV's at Fort Calhoun never fall below 400 psia (Reference Operating Manual, Technical Data Book, Figure III). Since the Fort Calhoun setpoints are well within the range of steam inlet pressures for which the valve has been tested, satisfactory operability has been adequately verified.

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 it 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.*

OPPD RESPONSE TO QUESTION #2

In considering various transients for review of PORV and safety valve operability, it was determined that the feed line break was not the limiting transient. The Fort Calhoun Station, USAR Section 14.10 specifically states "rupture of a main steamline, discussed in section 14.12, represents an upper limit for such an accident". Thus, the feedline break event was not considered applicable as related to PORV or safety valve operability. The loss of load analysis described in our April 1, 1982, submittal was found to be the limiting transient with respect to RCS pressure excursions and resulting peak pressure.

OPPD RESPONSE TO QUESTION #2 (Continued)

From the USAR, Sections 14.9 and 14.10; for the loss of load transient, the primary system pressure could increase at an average rate of about 60 psi/sec. and, assuming the PORV's did not open, result in a peak pressure of 2530 psia with the safety valves opened. The piping analysis, RELAP5, assumed pressure ramp rates which exceed that identified in the USAR for the loss of load transient.

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 tvpe 2 internals. The EPRI Safety 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.*

OPPD RESPONSE TO QUESTION #3

Type 2 PORV internals were purchased under OPPD, P.O. 60127, dated December 22, 1981. They were installed under OPPD Modification Request No. FC-80-35. Field work was completed May 9, 1984.

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.*

OPPD RESPONSE TO QUESTION #4

The control circuitry for the PORV is, for the most part, located outside of the containment building, in the switchgear and control rooms. As such, it would not be subjected to a harsh environment. The solenoid valves which open the PORV's are located at the PORV's inside containment. For the Fort Calhoun Station, the transients which might challenge the PORV's, namely loss of load or loss of feedwater flow, do not create a harsh environment in the containment. In the highly unlikely event that both PORV's failed to open when challenged, either of the two safety valves could provide more than enough capacity to handle the amount of steam that would be generated. This was described in our April 1, 1982, submittal.

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.*

OPPD RESPONSE TO QUESTION #5

The OPPD safety valve internals are designed for steam service. The valve manufacturer has stated that the primary purpose for changing to loop seal internals is to insure leak tightness. The difference between the internals is the material used. The dimensions of the two sets are the same so the operation of the valve is not affected. The valves at the Fort Calhoun Station have operated satisfactorily since they were first installed during the initial plant erection. They have provided over-pressure protection for the pressurizer without leaking since then. During each refueling outage the valves are removed, overhauled, and their operability is tested. The ring settings are: upper= -115, lower= -14. Also, OPPD does not feel that replacement of the internals is warranted since leakage occurring after an SV lift is of less concern than leakage during normal operation. This is particularly true since an SV lift is a very low probability event; i.e., actuation of the PORV's is sufficient to mitigate all expected overpressure transients at Fort Calhoun.

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 PORV's. 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.

OPPD RESPONSE TO QUESTION #6

PORV's - Piping upstream of the PORV's was not structurally modified. Directly downstream of the PORV's, an anchor restraint is attached to the pipe preventing downstream bending moments and loads from being transmitted to the valve. Therefore, the December 30, 1982 submittal remains current in its assertion that the PORV flange loads are less than those measured in the EPRI tests.

OPPD RESPONSE TO QUESTION #6 (Continued)

Safety Valves - Flange loads on the safety valves were calculated considering the thermal expansion of the pressurizer and the piping between the quench tank and the nozzles. Also considered was the dead weight loading and the loading due to either valve lifting and passage of the loop seal through the downstream piping.

Total flange loads were transmitted to the valve manufacturer, Crosby Valve and Gauge Co., who evaluated the effect that loading would have on the valve operability. Their report stated that the valves would not be adversely affected by those loads and would remain operable.

7. *Dresser Industries, the manufacturer of the Fort Calhoun PORV, wrote a letter to Metropolitan Edison Co., in March 1976 warning the 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 routinely placed in service at Fort Calhoun. The recommendation made by Dresser that the PORV be isolated at pressures lower than 1000 psi would seem to preclude the use of the PORV for low temperature 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.*

OPPD RESPONSE TO QUESTION #7

The PORV's at the Fort Calhoun Station are installed on water filled loop seals which prevent them from being subjected to a steam wirecutting process.

8. *The submittal did not address operation of the PORV block valves. The EPRI test data do not include any test data for the 2½ 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.*

OPPD RESPONSE TO QUESTION #8

In a letter to Mr. Harold Denton, Director, Nuclear Power Reactor Regulations, from Mr. R. C. Youngdahl, Chairman, EPRI Research Advisory Committee, dated July 24, 1981, the subject of block valve operation was addressed.

The Omaha Public Power District supports the EPRI position stated in that letter. It was concluded by the PWR utilities that "sufficient evidence (supported by test data) is available to demonstrate block valve operability." It was also concluded that additional block valve tests were not necessary because:

1. The probability of a relief valve failing to perform its intended function is low (on the order of 10^{-3}/year). This is the same magnitude as other small break loss-of-coolant accident (LOCA) initiators. Operability of the associated block valve does not increase the probability of a small break LOCA. Based on test data accumulated, the TMI valve failure was an isolated occurrence and not a generic problem associated with all relief valves.

2. The operability of block valves is not a safety issue. Plant procedures implemented since the TMI accident provide methods for safely shutting down a plant in the event of a small break LOCA.

3. Results of block valve tests performed at the Marshall Station have provided sufficient information to address valve operability.

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 of 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 PORV's in the analysis.

OPPD RESPONSE TO QUESTION #9

The thermohydraulic analysis was performed using the computer code RELAP5, mod 4. RELAP5 has been verified for use in the type of analysis by EPRI (EPRI Report NP-2479-LD). Fluid conditions for the analysis started at a pressure just below the set pressure with a rate of increase (pressurization rate) used provided to OPPD by the NSSS supplier, Combustion Engineering. The temperature of water in the loop seal was determined by testing by OPPD. A full size model of the loop seal was built and subjected to simulated plant conditions. Temperature readings taken were then used in the analysis. These were later verified under actual plant operation. Downstream conditions were calculated by the RELAP5 code in response to the valve opening. The analyzed valve actuation sequence consisted of 3 independent events.

1) Both PORV's opening simultaneously; and

2 & 3) Each safety valve opening alone.

This was based on the fact that both PORV's have the same setpoint which is lower than that of either safety valve. The safety valves have different setpoints. Also, CE analysis for Fort Calhoun has shown that any one of the 4 valves can mitigate a worse case pressurization rate in the primary system. The safety valve flow rate assumed in the analysis was 100% of the rated flow capacity rather than the more conservative 111% theoretical flow. Other assumptions in the analysis are judged to be conservative enough to make up for any unconservative effect the flow assumption may have.

A good example is the damping factor used in the structural analysis of the pipe. Our analysis assumed only one-half percent of critical damping which results in significantly greater loads and stresses than a more realistic damping factor of two or five percent. OPPD is confident that the reduction in loads as a result of increased damping would more than counterbalance any increase in loads due to higher flow rates.

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 values 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.

10. (continued)

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.

OPPD RESPONSE TO QUESTION #10

The thermohydraulic analysis utilized the RELAP5 computer code to generate transient fluid conditions in the discharge piping during a valve discharge event. The forces on the pipe due to these transients were calculated using the computer code FORCE marketed by the BOEING Computer Services Company. These forces were then input to the piping analysis computer code TPIPE which generated piping stress and hanger loads and verifies their acceptability against a code equivalent to the code of record used for the Fort Calhoun Plant. This analysis was performed on the entire piping system for each of the three events listed above in the response to question 9 with the piping in the post modification configuration it is now in for the PORV's and the safety valves. The analysis was performed at OPPD and was then sent to Impell Corp. in Walnut Creek, California where a check of the analysis and a separate independent review of the analysis were performed to verify the entire analysis.