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May 27, 1988
LIC-88-384

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Reference: Docket No. 50-285

Gentlemen:

SUBJECT: Response to Request for Additional Information Concerning NUREG-0737, Item II.D.1.

The Omaha Public Power District (OPPD) received reference 20 which detailed the NRC staff and it's consultant's review of NUREG 0737 Item II.D.1, Performance Testing of Relief and Safety Valves for Fort Calhoun Station.

OPPD has compiled the majority of the requested information and has summarized it in the attachment to this letter. As discussed with OPPD's Project Manager, Tony Bournia on April 27, 1988, the response to items 6 and 12 will be forwarded by June 30, 1988. Also, additional information is being prepared by vendors for responses to items 9 and 11. It will take approximately 8 weeks to complete these responses.

We believe that this additional information, in conjunction with the information to be submitted, as specified in the Attachment, will serve to address your reviewer's concerns.

Should you have any additional questions, please do not hesitate to contact us.

Sincerely,

R. L. Jaworski for
R. L. Andrews
Division Manager
Nuclear Production

RLA/me

cc: LeBoeuf, Lamb, Leiby & MacRae
R. D. Martin, NRC Regional Administrator
A. Bournia, NRC Project Manager
P. H. Harrell, NRC Senior Resident Inspector

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ATTACHMENT

NRC Question 1:

In Reference 12, the Fort Calhoun licensee, Omaha Public Power District (OPPD) stated that the bending moment on the Fort Calhoun PORV was less than the maximum bending moment for the test valve but no value was given which would allow a quantitative evaluation to be made.

<u>OPPD Response 1:</u>

The maximum moment calculated to be acting on the PORV's was 1918 foot-pounds or 23,016 inch-pounds. This is less than the bending moment for the test valve in the EPRI tests of 25,500 inch-pounds. The OPPD PORV's are therefore bounded by EPRI tests.

NRC Question 2:

The maximum expected backpressure for the PORVs was not provided which would allow the applicability of the EPRI tests to be verified.

<u>OPPD Response 2:</u>

The maximum backpressure at the discharge of the PORV's was calculated to be 400 psia. This is less than the backpressure measured at the discharge of the PORV's in the appropriate EPRI tests. This verifies the applicability of the EPRI tests.

NRC Question 3:

The safety val[illegible] loop seal temperature with the modified inlet piping that was used i[illegible] analysis of the SV piping was not provided.

<u>OPPD Response 3:</u>

The temperatures of the loop seals upstream of the safety valves used in the final analysis of the SV piping were based on a full scale test of the loop seal piping configuration. The report on that test provided the basis for a paper published by the American Institute of Chemical Engineers and presented at an AIChE Heat Transfer Symposium. The temperatures used in the SV piping analysis ranged from 265° F next to the valve to 500° F next to the steam/water interface with 640° F inlet steam.

The proper use of these temperatures was separately checked and independently reviewed by a qualified consulting engineering firm with experience in performing thermal-hydraulic analyses.

NRC Question 4:

The EPRI/CE tests on the Dresser 31533VX-30 PORV show a very high probability that this valve will not close on demand when used with a cold loop seal. (For the two 103-105° F loop seal tests, one required 70 sec before the valve closed and the other required 90 sec and closing of the block valve before the test valve closed.) The CE submittal, CEN-213 (Reference 17), indicates that this valve is acceptable for plant use with the cold loop seal. The NRC staff does not concur with this conclusion. Based on the available data, it is the staff position that the Dresser 31533VX-30 PORV at Fort Calhoun is not qualified because of the delayed closure problem with the cold loop seal. OPPD's proposed action and schedule are required in order to resolve this issue.

OPPD Response to Question 4:

As part of this project, OPPD modified the loop seals upstream of the PORV's to increase their temperatures. The temperatures used in the final analysis of the PORV piping ranged from 130° F next to the PORV's to 600° F at the steam water interface. These temperatures are lower than those expected based on the full scale loop seal temperature test referenced in the previous question. Lower temperatures were used in the analysis for conservatism (lower temperatures produce larger forces downstream). The actual temperatures the loop seal will rise to are expected to be above 200° F at the seat of the valve.

Although these temperatures are not as high as those in the EPRI tests that had closure immediately upon demand, they are considerably higher than the loop seal temperatures in the tests when immediate closure did not occur. OPPD's position is they are sufficiently high to insure that adequate PORV closing can be achieved. This position is supported by two events at Fort Calhoun during which the PORV's did perform satisfactorily to terminate overpressure transients which resulted from a loss-of-load. The first event occurred on May 28, 1976, and the second occurred on July 2, 1986.

Prior to both events, the unit was operating at a nominal pressurizer pressure of 2100 psia. When the loss of load occurred, the pressure increased rapidly to 2400 psia where the PORV's were actuated. The pressure increase terminated at this point and a sharp decrease was initiated.

During the first event, the pressure decreased to 1950 psia due to the normal pressurizer level and pressure decreases which occur following the reactor trip and due to blowdown from the PORV's. During the second event, the pressure decreased to 1725 psia for the same reasons and due to the main feedwater controls remaining opened for a longer period of time. All available data indicated that the PORV's successfully closed for both transients. The quench tank rupture disk remained intact. The pressurizer pressure then increased back up to approximately 2100 psia and stabilized at that level.

The PORV's operated as designed to terminate the overpressure transient which followed these loss-of-load events. In both cases, the pressure decrease was terminated above the Safety Injection Actuation Signal setpoint. Based on these operating experiences and the increased loop seal temperatures, the PORV operation is considered to be adequate. OPPD has no plans for further actions to address this issue.

NRC Question 5:

Test information showing that the Crane block valves and their Limitorque SMB-00-7.5 operators will open and close under all possible conditions at Fort Calhoun was not provided. OPPD's response dated March 1, 1986 (Reference 15) to a request for additional information dated July 23, 1985 (Reference 14, question 8) was that (1) the probability of PORV failure is low (-10^{-3}/year), (2) that block valve operability is not a safety issue, and (3) that the Marshall Station block valve tests adequately show their operability.

The arguments presented by OPPD are not acceptable for the following reasons: (1) Based on the EPRI Dresser PORV cold loop seal test data, there is a very high probability that the Fort Calhoun PORVs will fail to close during a transient when the PORVs are required to operate, and (2) NUREG-0737 (Item II.D.1) requires the licensee to show block valve operability under all expected flow conditions.

The licensee response to question #8 in Reference 15 is considered to be unresponsive. The problems encountered with Westinghouse block valves failing to close against operating pressures which were identified during the block valve test (Reference 18) raised a safety concern for all untested block valves and their operators that must be addressed. OPPD must explain specifically how the Marshall Station block valve test data applied to the Fort Calhoun Crane block valves and their operators, or provide test data for the Fort Calhoun Crane block valves and their operators. It should be noted that manufacturer's calculations are not sufficient to show the block valve operators provide sufficient torque to close the valves.

OPPD's Response to Question #5:

Question 5 concerns the operability of our motor operated PORV block valves (HCV-150 and 151). Additional time is needed to respond to this question. A response will be provided by June 30, 1938.

NRC Question 6:

The thermal-hydraulic analyses of the SV/PORV piping system, referred to in Reference 15, was done using the RELAP5/MOD1 code. Three cases were analyzed, 1, both PORVs opening at the same time (the PORVs have the same set pressure), 2, and 3, each SV opening alone (the SVs have staggered set pressures of 2485 psig and 2530 psig). No case was run where both safety valves lift simultaneously. During the loop seal test, the opening pressures for the valves ranged from +1.4 to +5.5% of set pressure, and during the steam tests and opening pressures ranged from -2.6 to +1.4% of set pressure. Since the set pressures of the two Fort Calhoun safety valves are only 1.8% apart, which is within the expected range of lift pressures for the safety valves, it is just as likely for the two valves to lift at the same pressure as for them to lift at different pressures. Therefore, a case with both safety valves lifting simultaneously must be run or OPPD must justify not running it because it is bounded by one of the other cases.

OPPD's Response to Question 6:

Although OPPD's position is that the chances of both valves opening simultaneously is extremely small, we are currently investigating the analyses already performed to determine if they bound the simultaneous opening event. This is thought to be feasible due to the particular piping configuration exis.] at Fort Calhoun. If the review finds that it is bounded, OPPD will p.ovide the NRC with adequate information to justify the position. If the review finds that the existing analyses will not bound the simultaneous opening event, OPPD will provide the NRC with that information and our proposed action and schedule at that point in time. This review should be completed by June 30, 1988.

NRC Question 7:

The licensee did not identify the codes or standards used in the recent analyses of the SV and PORV piping from the pressurizer to the pressurizer relief tank, did not identify the allowable stresses used for the piping/support, and did not provide a comparison of the piping and support stresses/loads with the allowable stresses/loads.

OPPD's Response to Question 7:

Piping upstream of the safety valves and upstream of the PORV's were analyzed to USA Standard B31.7 Nuclear Power Piping Code (1969 Edition 1971 Addenda). Piping downstream of those valves to the quench tank was analyzed to ASME Section III Subsection NC (Class 2), 1974. The codes used in these analyses were equivalent to the original design criteria for the plant. Other information requested in this question will be addressed in the summary report of the analyses discussed in OPPD's response to Question 11.

NRC Question 8:

The submittal states that "normal loads" were considered in the SV/PORV piping and supports in addition to valve opening fluid transient condi'' ~s. The licensee did not identify what "normal loads" were incluaca and how these loads combined. Reference 18 indicates the load combinations that should be applied and how they are to be combined. The licensee did not show how these load combinations were considered in the piping analyses, or justify what was used if it was different than that recommended in Reference 18.

OPPD's Response to Question 8:

This question is requesting information concerning the structural portion of OPPD's analyses. This information will be addressed in the summary report of the analyses discussed in OPPD's response to Question 11.

Attachment (Continued)

NRC Question 9:

The submittal states that RELAP5/MOD1 was used in the thermal-hydraulic analysis and FORCE was used to predict the piping loads that result. To allow for a complete evaluation of the methods used and the results obtained from the thermal-hydraulic analysis, a discussion that contains at least the following information should have been provided.

a. Identification of important parameters used in the analysis and rationale for their selection. These include peak pressure, peak pressurization rate, valve flow rate, valve opening time, loop seal temperature and other fluid conditions for the cases analyzed. Use of the ASME rated valve flow rate is acceptable because the measured flow rate was only 99-104% of rated flow at 3% accumulation.

b. Information on the model used is needed and how well it adheres to the guidelines in Reference 19. These include control volume length, calculation time step used, number of control volumes, and initial pipe conditions before the transient is run. Reference 4 recommends using control volume lengths between 0.5 and 1.0 ft and calculational time step size limited by the mass transport Courant limit for loop seal conditions. (The maximum time step used in the Reference 4 analyses was 2×10^{-4} sec.) If these recommendations were not adhered to, justification of what was used is required.

c. Provide a sketch of the thermal-hydraulic model used showing control volume sizes and locations. A copy of the thermal-hydraulic analysis report should also be provided.

<u>OPPD's Response to Question 9</u>:

This question is requesting information about the RELAP5 analysis that is quite detailed in nature. OPPD is currently in the process of compiling a summary of the analyses that will address the concerns stated in the question. Given the extensive nature of some of the items, it is expected to take approximately 8 weeks to provide an adequate response to this question. This response also depends on the results of the review discussed in the response to Question 6.

NRC Question 10:

Verification of the FORCE Code was not provided. This verification should compare predicted vs measured loads using EPRI SV/PORV test data and conditions. If other data or a standard problem was used to verify FORCE, the licensee must demonstrate that it is applicable.

<u>OPPD's Response to Question 10</u>:

Verification of the FORCE code was performed by the Boeing computer Services (BCS) Division of the Boeing Company under their Quality Assurance Program. Verification test results are retained by BCS. Reverification by OPPD was not considered necessary and was not performed since the program was executed on Boeing's system. Attached are copies of BCS memos addressing this certification.

NRC Question 11:

The submittal states that a structural analysis of the SV/PORV piping system has been conducted using TPIPE, but did not present details of the analysis. To allow for a complete evaluation of the methods used and results obtained from the structural analysis, please provide reports containing at least the following information:

a. Verification of TPIPE for use of dynamic piping structural analyses such as these.

b. How the FORCE calculation loads are applied.

c. A description of methods used to model supports, the pressurizer and relief tank connections, the safety valve bonnet assemblies, and the PORV actuator. Other code input information such as lumped mass spacing, calculation time step, damping factor, and cutoff frequency are also requested. Cutoff frequencies of less than 100 Hz need to be justified if used, and the lumped mass spacing and calculation time step should be consistent with the 100 Hz cutoff frequency or justification provided for the values used.

d. An evaluation of the results of the structural analysis, including a description of modifications made as a result of earlier stress analyses.

e. A sketch of the structural model showing lumped mass locations, pipe sizes, and application points of fluid forces.

f. A copy of the structural analysis report.

OPPD's Response to Question 11:

As with verification of the "FORCE" code mentioned in Question 10, verification of the TPIPE code was performed by the Boeing Computer Services (BCS) Division of the Boeing Company under their Quality Assurance Program. Verification test results are retained by BCS. Reverification by OPPD was not considered necessary and was not performed since the program was executed on Boeing's system. Attached are copies of BCS memos addressing this certification.

As with the data requested in Question 9 for the RELAP5 analysis, the data requested for the dynamic piping structural analysis are quite extensive and detailed in nature. And as with the RELAP5 analyses, OPPD is currently in the process of compiling a summary of the structural analyses that will address the concerns of Quesiton 11. It is expected to take approximately 8 weeks to provide an adequate response to this quesiton. This response also depends on the results of the review discussed in the response to Question 6.

NRC Question 12:

NUREG-0737, Item II.D.1 requires that the plant specific PORV Control Circuitry be qualified for design-basis transients and accidents. OPPD's response to this was, "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 PORVs are located at the PORVs inside containment. For the Fort Calhoun Station, the transients which might challenge the PORVs, 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 PORVs 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."

The licensee's statement is considered evasive since it does not address the pertinent requirements of NUREG-0737, Item II.D.1, namely, accidents and transients inside the containment that subject the PORV circuitry to harsh environment during which the PORV may operate.

The staff has agreed that meeting the licensing requirements of 10 CFR 50.49 for this circuitry is satisfactory and that specific testing per NUREG-0737 requirement is not required. Therefore verify whether the PORV control circuitry has been reviewed and accepted under the requirements of 10 CFR 50.49.

If the PORV circuitry has not been qualified to the requirements of 10 CFR 50.49, provide information to demonstrate that the control circuitry is qualified per the guidance provided in Reg. Guide 1.89, Revision 1, Appendix E.

As an alternative, the staff has determined that the requirements of NUREG-0737 regarding the qualification of the PORV control circuitry may be satisfied if one or more of the following conditions is met.

a. The PORVs are not required to perform a safety function to mitigate the effects of any design basis event in the harsh environment, and failure in the harsh environment will not adversely impact safety functions or mislead the operator (PORVs will not experience any spurious actuations and, if emergency operating procedures do not specifically prohibit use of PORVs in accident mitigation, it must be ascertained that PORVs can be closed under harsh environment conditions).

b. The PORVs are required to perform a safety function to mitigate the effects of a specific event, but are not subjected to a harsh environment as a result of that event.

c. The PORVs perform their function before being exposed to the harsh environment, and the adequacy of the time margin provided is justified; subsequent failure of the PORVs as a result of the harsh environment will not degrade other safety functions or mislead the operator (PORVs will not experience any spurious actuations and, if emergency operating procedures do not specifically prohibit use of PORVs in accident mitigation, it must be ascertained that PORVs can be closed under harsh environment conditions).

d. The safety function can be accomplished by some other designated equipment that has been adequately qualified and satisfies the single-failure criterion.

OPPD Response to Question 12:

Additional time is required to fully respond to this question. OPPD will respond to this question by June 30, 1988.

REFERENCES

1. TMI-Lessons Learned Task Force Status Report and Short-Term Recommendations, NUREG-0578, July 1979.

2. Clarification of TMI Action Plan Requirements, NUREG-0737, November 1980.

3. R. C. Youngdahl ltr. to H. R. Denton, Submittal of PWR Valve Test Report, EPRI NP-2628-SR, December 1982

4. EPRI Plan for Performance Testing of PWR Safety and Relief Valves, July 1980.

5. EPRI PWR Safety and Relief Valve Test Program, Valve Selection/Justification Report, EPRI NP-2292, December 1982.

6. EPRI PWR Safety and Relief Valve Test Program, Test Condition Justification Report, EPRI NP-2460, December 1982.

7. Valve Inlet Fluid Conditions for Pressurizer Safety and Relief Valves in Combustion Engineering-Design Plants, EPRI NP-2318, December 1982.

8. EPRI PWR Safety and Relief Valve Test Program, Safety and Relief Valve Test Report, EPRI NP-2628-SR, December 1982.

9. EPRI/Marshall Electric Motor Operated Block Valve, EPRI NP-2514-LD, July 1982.

10. Letter W. C. Jones, OPPD, to R. A. Clark, NRC, "Safety and Relief Valve Test Program," LIC-82-138, April 1, 1982.

11. Letter W. C. Jones, OPPD, to R. A. Clark, NRC, "NUREG-0737, Item II.D.1, Relief and Safety Valve Test Program," LIC-82-253, July 1, 1982.

12. Letter W. C. Jones, OPPD, to R. A. Clark, NRC, "NUREG-0737, Item II.D.1, Relief and Safety Valve Test Program," LIC-82-415, December 30, 1982.

13. Letter W. C. Jones, OPPD, to R. A. Clark, NRC, "NUREG-0737, Item II.D.1, Relief and Safety Valve Test Program," LIC-82-182, August 2, 1982.

14. Letter E. J. Butcher, NRC to R. L. Andrews, OPPD, "Request for Additional Information Regarding NUREG-0737, Item II.D.1 concerning Performance Testing of Relief and Safety Valves," July 23, 1985.

15. Letter R. L. Andrews, OPPD, to A. C. Thadani, NRC, "Additional Information on Performance Testing of Relief and Safety Valve Testing, NURG-0737, Item II.D.1," LIC-86-083, March 1, 1986.

16. C-E Owner's Group, Summary Report on the Operability of Pressurizer Safety Valves in C-E Designed Plants, CEN-227, December 1982.

17. C-E Owner's Group, Summary Report on the Operability of Power Operated Relief Valves in C-E Designed Plants, CEN-213, June 1982.

18. EPRI PWR Safety and Relief Valve Test Program, Guide for Application of Valve Test Program Results to Plant-Specific Evaluations, Revision 2, Interim Report, July 1982.

19. Application of RELAP5/MOD1 for Calculation of Safety and Relief Valve Discharge Piping Hydrodynamic Loads, EPRI-2479, December 1982.

20. Letter from NRC (A. Bournia) to OPPD (R. L. Andrews), dated February 17, 1988.