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LIC-89-874

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-137
Washington, DC 20555

References: 1. Docket 50-285
2. Letter from NRC (A. Bournia) to OPPD (K. J. Morris) dated July 24, 1989

Gentlemen:

SUBJECT: OPPD Response to NRC Finding Contained in Technical Evaluation Report, NUREG-0737, Item II. D. 2, "Performance Testing of Relief and Safety Valves," for Fort Calhoun Station (TAC No. 445582)

Enclosed please find Omaha Public Power District's (OPPD) response to concerns outlined in Reference 2.

If you should have any questions, please do not hesitate to contact me or members of my staff.

Sincerely,

K. J. Morris
Division Manager
Nuclear Operations

KJM/pjc

Attachment: Response to NRC Technical Evaluation Report (TAC No. 445582) NUREG-0737, Item II. D. 2, "Performance Testing of Relief and Safety Valves," for Fort Calhoun Station

c: 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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OPPD Response to Reference 2, (NRC) TECHNICAL EVALUATION REPORT, NUREG-0737, ITEM II.D.2,"PERFORMANCE TESTING OF RELIEF AND SAFETY VALVES", FOR FORT CALHOUN STATION (TAC NO. 445582)

References:
1) Docket No. 50-285
2) Letter from NRC (A. Bournia) to OPPD (K. J. Morris) dated July 24, 1989
3) NRC Generic Letter 89-10
4) NUREG-0578 and NUREG-0737

OPPD's review of Reference 2 disclosed a number of items which require further action or response on OPPD's part to resolve. We have summarized for each item required by Reference 4, the open issues (concerns) and our responses to them:

ITEM 1:

Conduct testing to qualify reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents.

CONCERN:

The part of Item 1 that requires tests be conducted to qualify reactor coolant system relief valves was not met for the PORVs or the PORV block valves. The test results demonstrated that the Power Operated Relief Valves (PORVs) might not close properly following operation; therefore, they were not demonstrated to operate reliably. Data from tests that are applicable to the Fort Calhoun block valves was not presented.

RESPONSE:

a) PORV Block Valves: Because of Fort Calhoun's unique block valves no test results are available for demonstrating operability under conditions such as might occur during PORV discharge. Test results which are available for other valve and motor operator combinations indicate a considerable scatter in the theoretical valve factors calculated from actual torques required to lose and seat the tested valves. Calculations were performed using valve factors based on manufacturer's revised recommendations and the test results for other valves to determine required operator torque for valve closure to demonstrate operability. The operability of the block valves is judged to be a low risk issue for the following reasons: 1) Fort Calhoun is operated with the PORV block valves open and therefore their operability is not required to protect against overpressurizing the RCS; 2) In the event that one should be required to isolate a stuck open PORV, the differential pressure and fluid momentum forces across the valve would initially be insignificant, and therefore not interfere with movement of the valve; 3) Since the valve would already be in motion when these forces became significant, there would be less resistance to closing against sliding friction than if static friction were being overcome; 4) Although complete closure might not be attained, a reduction in coolant outflow would result; 5) The maximum flow area of the PORVs is less than that assumed in the plant design for a Small Break loss of coolant accident (LOCA).

Since testing of motor operated valves (MOVs) is part of the resolution of another issue (Reference 3), we are investigating participation in a MOV owners group; otherwise we may independently have testing done on valves similar to our block valves. Completion of the Reference 3 activities is required by the end of 1995, with utility commitments due by the end of 1989. A decision to test, replace, or modify the PORV block valves will be incorporated into our response to Reference 3.

b) PORV operation with 200 F Loop Seal: Test data indicated that cold loop seal water delayed closing of the PORVs, potentially resulting in excessive loss of coolant inventory and/or drop in reactor coolant loop pressure. OPPD's actual experience with these valves, as noted in previous responses, shows that no unacceptable operating condition resulted from any delayed closure associated with the original loopseal or with the present loopseal. The actual loopseal configuration contains less water than the original design, which increases the margin against an unacceptable delay (in closure). The delayed closure of the PORVs is bounded by the Updated Safety Analysis Report (USAR) Section 14.22 "Reactor Coolant System Depressurization Incident," which assumes two PORVs to be inadvertently opened while operating at rated thermal power. The analysis results for this event show that about 7 seconds elapse at which time the reactor trips, and that no departure from nucleate boiling DNB occurs even though the PORVs are assumed to remain open. Therefore, the concern regarding PORV closure is not a safety issue.

ITEM 2:

Determine valve expected operating conditions through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Rev. 2.

CONCERN:

The part of Item 2 that requires the expected valve operating conditions to be determined for the transients and accidents listed in Regulatory Guide 1.70, Rev. 2 was not met for the feedwater line break (FWLB). This is because conditions for the FWLB were not considered.

RESPONSE:

OPPD has analyzed the FWLB event and the results show that the pressurizer (PZR) will not fill with water due to the resulting volume expansion. This analysis used the Combustion Engineering System Excursion Code (CESEC) model and methodology which was previously reviewed and approved by the NRC, and has been used in our licensing submittals of reload analyses since the early 1980s. Therefore, only steam discharge is anticipated during a FWLB and the conditions chosen bound the appropriate transients and accidents.

ITEM 3:

Choose the single failure such that the dynamic forces on the safety and relief valves are maximized.

No open issues.

ITEM 4:

Use the highest test pressure predicted by conventional safety analysis procedures.

CONCERN:

The part of Item 4 that requires the highest predicted pressure be chosen for the tests was not met for the PORV block valves because the Fort Calhoun block valves were not tested.

RESPONSE:

See item 1a above.

ITEM 5:

Include in the relief and safety valve qualification program the qualification of the associated control circuitry.

No open issues.

ITEM 6:

Provide test data for NRC staff review and evaluation including criteria for success or failure of valves tested.

CONCERN:

The part of Item 6 that requires test data be provided for the NRC staff review and evaluation, including criteria for success or failure of valves tested, was not met for the PORV block valves. This is because the plant specific valves were not tested by EPRI, and OPPD did not provide adequate justification that EPRI test results are applicable to the Fort Calhoun block valves.

RESPONSE:

See item 1a above.

ITEM 7:

Submit a correlation or other evidence to substantiate the valves tested in a generic test program demonstrate the functionability of as-installed primary relief and safety valves. This correlation must show that the test conditions used are equivalent to expected operating and accident conditions as prescribed in the Final Safety Analysis Report (FSAR). The effect of as-built relief and safety valve discharge piping on valve operability must also be considered.

CONCERN:

The following parts of Item 7 were not met.

a. The part of Item 7 that requires the Licensee to submit a correlation or other evidence to substantiate the valves tested in a generic test program demonstrate the functionability of as-installed primary relief and safety valves was not met for the PORV block valves. The Licensee has not provided evidence to indicate the valves tested by EPRI are representative of the untested block valves at Fort Calhoun. The Licensee's conclusions regarding the adequacy of the plant operator were based on calculations and not supported directly by test data.

b. The part of Item 7 that requires showing the test conditions are equivalent to those prescribed in the FSAR was not met for water discharge through the safety valves, PORVs, and PORV block valves. Conclusive evidence was not provided to show the PORVs and safety valves will not be required to pass water during a feedline break event. Evidence was not provided that block valve tests were completed under conditions that bound the Fort Calhoun plant specific conditions.

RESPONSE:

a) See item 1a above.

b) See item 2 above.

ITEM 8:

Qualify the plant specific safety and relief valve piping and supports by comparing to test data and/or performing appropriate analysis.

CONCERN:

The part of Item 8 that requires qualification of the piping and supports was not met. This is because, based on the information supplied by the Licensee, it cannot be concluded the thermal-hydraulic and structural analyses were adequate to qualify the pressurizer safety and relief valve piping and supports.

RESPONSE:

a) The time step size used in the analysis for Fort Calhoun was allowed to vary under the optimizing scheme of RELAP 5, rather than being held constant as was done for the EPRI analysis work. EG&G has identified this as being inconsistent with the EPRI studies. OPPD believes, however, that we were consistent with the intent of the EPRI work which demonstrated a need for a very small time step size during the critical period of loop seal passage. The optimizing routine of RELAP 5 allows the time step to increase in size during periods when rates of change in fluid properties

are less critical. In addition, EG&G noted that some of the segment lengths used in our model were less than the 0.5 ft. used in the EPRI work. Shorter segments would suggest the need for smaller time steps than established by the EPRI studies. In our model, only a small percentage of the total number of segments were less than 0.5 ft. and these segments were nearly all upstream of the valves, where flashing and water hammer are insignificant. For the downstream piping, the only segments which were significantly less than 0.5 ft., were for component 122, which is a straight length of pipe, 5 ft. long, located in the discharge pipe for one of the safety valves (SVs). The segment lengths are: one segment at 0.196 ft. and ten segments at 0.412 ft. This section of pipe is only a small portion of the overall piping network. A review of the force time history plots for this component shows peak unbalanced segment forces of 20 Kips to be acting on it during the brief period of loopseal passage. This compares well to the peak forces of 16 Kips in component 99, which is located in a similar discharge line for the other SV where segment lengths are all greater than 0.5 ft. Therefore, the use of segment lengths less than 0.5 ft. does not affect the analysis conclusions.

b) Seismic Stresses were not combined with Thermal Hydraulic Stresses: The pipe stress summary does not combine seismic with thermal hydraulic stresses in all cases. The seismic induced pipe stresses for this system are small (5.2 ksi maximum) and it is improbable that peak seismic stresses could occur simultaneously with peak thermal hydraulic (T-H) stresses since the peaks are statistically independent and the T-H peak is of extremely short duration. An SRSS combination of seismic and T-H stresses only increase the present stress summary values by about 2 ksi maximum, which is insignificant in comparison to the allowable stresses. Therefore, the omission of combined seismic and T-H stresses in the pipe stress summary does not affect the analysis conclusion.

c) Pipe Restraint Allowables based on Design Basis Criteria: Pipe restraint loads were combined in accordance with EPRI guidelines, but stresses were compared to Design Basis Allowables in lieu of EPRI suggested values. The EPRI guidelines acknowledge the use of Design Basis Criteria as being acceptable in Note (1), Table 4.4-2 of the EPRI document. With respect to seismic events, Fort Calhoun's Design Basis Earthquake (DBE) is equivalent to a Safe Shutdown Earthquake (SSE), therefore our use of DBE loads is consistent with the use of SSE loads.

New Issues not itemized in Reference 4:

ITEM A:

<u>CONCERN:</u>

The tests demonstrated the need for inspection and maintenance to maintain the operability of the safety valves. Therefore, the Licensee must develop procedures requiring inspection and maintenance of the safety valves following each lift which involves loop seal or water discharge. They must be incorporated into the plant operating procedures or licensing documents.

RESPONSE:

Procedures to inspect SVs following lift events: Procedures will be implemented, as recommended, to inspect the SVs following each lift event that might occur in the future. These procedures will be in place by the end of the upcoming 1990 refueling outage.

ITEM B:

CONCERN:

With respect to the PORVs, it was concluded that unless the heavier springs recommended by Dresser are installed under the main and pilot disks or OPPD can demonstrate that PORV leakage is not a problem at low pressures, the plant valve is not considered operable below 100 psig.

RESPONSE:

Low Pressure Performance of PORVs: The manufacturers' recommendation to replace the springs was to avoid seat and disk damage which might occur due to leakage at low pressures with the original springs. Damage to the seating surfaces could result in continuous leakage at all pressures, which in turn would cause additional damage. Also, if the amount of leakage were significant, temperature sensors (downstream of the PORVs) would go into alarm. These conditions would be unacceptable. Actual operating experience with the existing valves and springs has demonstrated that no unacceptable leakage occurs during transient low pressure or normal steady state operation. We have discussed this matter with the valve manufacturer and have jointly concluded that replacement of the springs is not warranted. However, we have ordered new springs for installation if future maintenance requiring disassembly is required.

STATUS SUMMARY OF ITEMS:

ITEM 1:

a) Open item to be resolved by closure of Reference 3 (NRC Generic Letter 89-10).

b) Considered closed.

ITEM 2:

No open issues.

ITEM 3:

No open issues.

ITEM 4:

To be resolved with Item 1a above.

ITEM 5:

No open issues.

ITEM 6:

To be resolved with Item 1a above.

ITEM 7:

a) To be resolved with Item 1a above.

b) Considered closed with Item 2 above.

ITEM 8:

a) Considered closed.

b) Considered closed.

c) Considered closed.

ITEM A:

The procedure to inspect SVs following any future lift events is to be in place by end of next refueling outage.

ITEM B:

Considered closed.