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TECHNICAL EVALUATION REPORT

TMI ACTION--NUREG-0737 (II.D.1) RELIEF AND SAFETY VALVE TESTING FORT CALHOUN

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TECHNICAL EVALUATION REPORT
TMI ACTION--NUREG-0737 (II.D.1)
RELIEF AND SAFETY VALVE TESTING
FORT CALHOUN
DOCKET NO. 50-285

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ABSTRACT

Light water reactors have experienced a number of occurrences of improper performance of safety and relief valves installed in the primary coolant system. As a result, the authors of NUREG-0578 (TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations) and subsequently NUREG-0737 (Clarification of TMI Action Plan Requirements) recommended that programs be developed and completed which would reevaluate the functional performance capabilities of Pressurized Water Reactor (PWR) safety, relief, and block valves and which would verify the integrity of the piping systems for normal, transient, and accident conditions. This report documents the review of these programs by the Nuclear Regulatory Commission (NRC) and their consultant, EG&G Idaho, Inc. Specifically, this report documents the review of the Fort Calhoun Licensee response to the requirements of NUREG-0578 and NUREG-0737. This review found that the licensee, Omaha Public Power District has not provided an acceptable submittal reconfirming that the General Design Criteria 14, 15, and 30 of Appendix A to 10 CFR 50 have been met.

FIN No. A6492--Evaluation of OR Licensing Actions-NUREG-0737, II.D.1

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1. INTRODUCTION

1.1 Background

Light water reactor experience has included a number of instances of improper performance of relief and safety valves installed in the primary coolant system. There have been instances of valves opening below set pressure, valves opening above set pressure, and valves failing to open or reseat. From these past instances of improper valve performance, it is not known whether they occurred because of a limited qualification of the valve or because of basic unreliability of the valve design. It is known that the failure of a power operated relief valve (PORV) to reseat was a significant contributor to the Three Mile Island (TMI-2) sequence of events. These facts led the task force which prepared NUREG-0578 (Reference 1) and, subsequently, NUREG-0737 (Reference 2) to recommend that programs be developed and executed which would reexamine the functional performance capabilities of Pressurized Water Reactor (PWR) safety, relief, and block valves and which would verify the integrity of the piping systems for normal, transient, and accident conditions. These programs were deemed necessary to reconfirm that the General Design Criteria 14, 15, and 30 of Appendix A to Part 50 of the Code of Federal Regulations, 10 CFR, are indeed satisfied.

1.2 General Design Criteria and NUREG Requirements

General Design Criteria 14, 15, and 30 require that (1) the reactor primary coolant pressure boundary be designed, fabricated, and tested so as to have extremely low probability of abnormal leakage, (2) the reactor coolant system and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions are not exceeded during normal operation or anticipated transient events, and (3) the components which are part of the reactor coolant pressure boundary shall be constructed to the highest quality standards practical.

To reconfirm the integrity of overpressure protection systems and thereby assure that the General Design Criteria are met, the NUREG-0578 position was issued as a requirement in a letter dated September 13, 1979, by the Division of Licensing (DL), Office of Nuclear Reactor Regulation (NRR), to ALL OPERATING NUCLEAR POWER PLANTS. This requirement has since been incorporated as Item II.D.1 of NUREG-0737, Clarification of TMI Action Plan Requirements, which was issued for implementation on October 31, 1980. As stated in the NUREG reports, each pressurized water reactor Licensee or Applicant shall:

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

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.

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

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

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

6. Provide test data for Nuclear Regulatory Commission (NRC) staff review and evaluation, including criteria for success or failure of valves tested.

7. Submit a correlation or other evidence to substantiate that 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 usedare 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 be considered.

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

2. PWR OWNER'S GROUP RELIEF AND SAFETY VALVE PROGRAM

In response to the NUREG requirements previously listed, a group of utilities with PWRs requested the assistance of the Electric Power Research Institute (EPRI) in developing and implementing a generic test program for pressurizer safety valves, power operated relief valves, block valves, and associated piping systems. Omaha Public Power District (OPPD), the owner of Fort Calhoun, was one of the utilities sponsoring the EPRI Valve Test Program. The results of the program, which are contained in a series of reports, were transmitted to the NRC by Reference 3. The applicability of these reports is discussed below.

EPRI developed a plan (Reference 4) for testing PWR safety, relief, and block valves under conditions which bound actual plant operating conditions. EPRI, through the valve manufacturers, identified the valves used in the overpressure protection systems of the participating utilities and representative valves were selected for testing. These valves included a sufficient number of the variable characteristics so that their testing would adequately demonstrate the performance of the valves used by utilities (Reference 5). EPRI, through the Nuclear Steam Supply System (NSSS) vendors, evaluated the FSARs of the participating utilities and arrived at a test matrix which bounded the plant transients for which over pressure protection would be required (Reference 6).

EPRI contracted with Combustion Engineering (CE) to produce a report on the inlet fluid conditions for pressurizer safety and relief valves in CE designed plants (Reference 7). Since Fort Calhoun was designed by CE, this report is relevant to this evaluation.

Several test series were sponsored by EPRI. PORVs and block valves were tested at the Duke Power Company Marshall Steam Station located in Terrell, North Carolina. Additional PORV tests were conducted at the Wyle Laboratories Test Facility located in Norco, California. Safety valves (SVs) were tested at the Combustion Engineering Company, Kressinger

3. PLANT SPECIFIC SUBMITTAL

A preliminary assessment of the adequacy of the overpressure protection system was submitted by OPPD on April 1, 1982 (Reference 10). A later submittal on the operability of the PORVs was transmitted by OPPD on July 1, 1982 (Reference 11). An initial assessment of the Safety Valves and the Pressurizer Safety and Relief Valve Piping was transmitted December 30, 1982 (Reference 12) and additional information was submitted on August 2, 1983 (Reference 13). A request for additional information (Reference 14) was submitted to OPPD by the NRC on July 23, 1985. OPPD responded to this request on March 1, 1986 (Reference 15).

The response of the overpressure protection system to Anticipated Transients Without Scram (ATWS) and the operation of the system during feed and bleed decay heat removal are not considered in this review. Neither the Licensee nor the NRC have evaluated the performance of the system for these events.

4. REVIEW AND EVALUATION

4.1 Valves Tested

Fort Calhoun, a CE designed PWR, utilizes two safety valves, two PORVs, and two PORV block valves in the overpressure protection system. The safety valves are Crosby HB-BP-86 3K6 valves with steam internals. The plant safety valve inlet configuration is being modified to consist of a short liquid filled loop seal similar to that shown in Figure 1; this figure was taken from Reference 16. The PORVs are Dresser 31533VX-30 solenoid actuated, pilot operated valves with a bore diameter of 1-3/32 in. Fort Calhoun uses cold loop seals upstream of the PORVs. The block valves are 2-1/2 in. Crane 787-U gate valves with Limitorque SMB-00-7.5 operators.

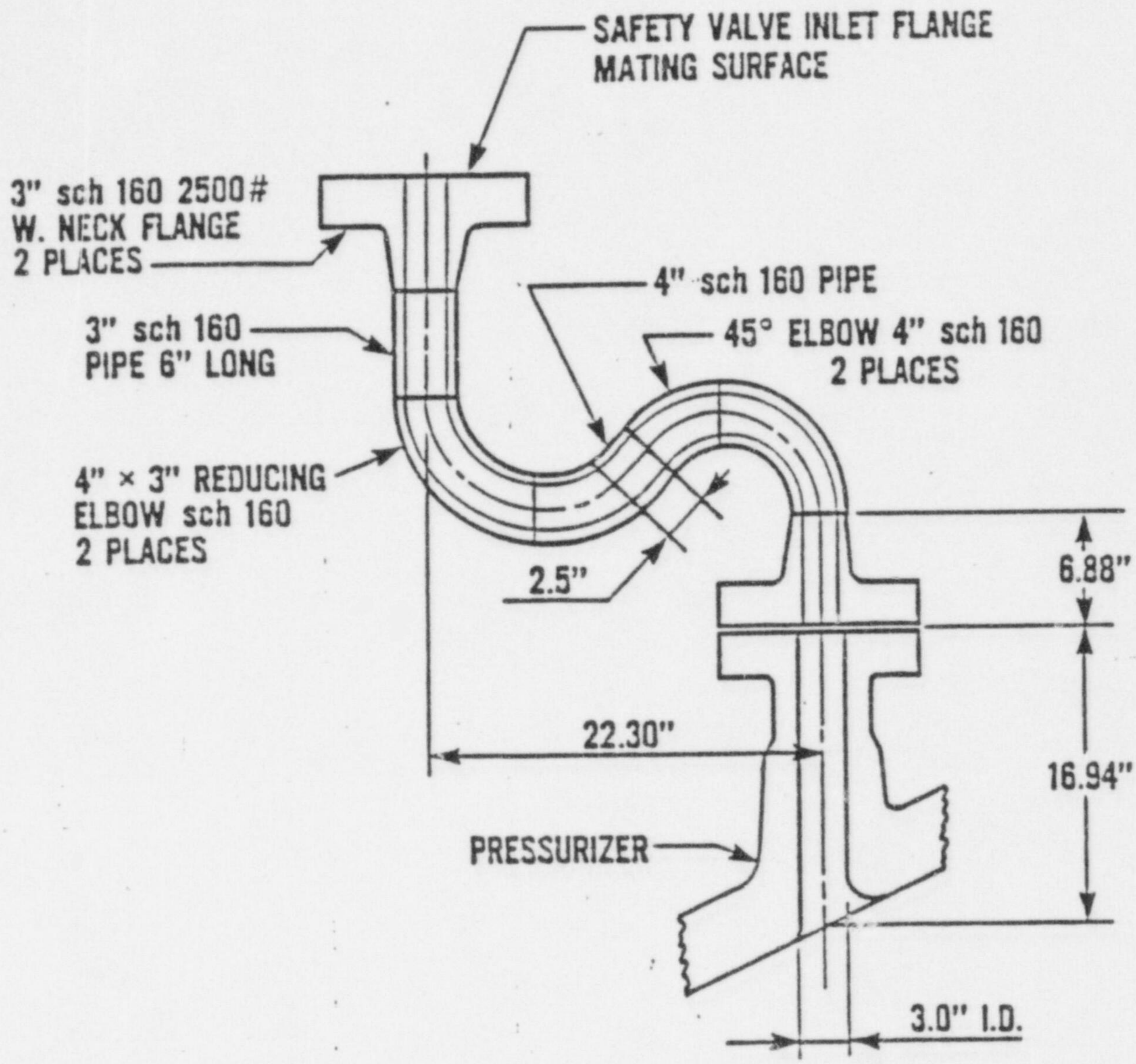
The Crosby 3K6 safety valve was one of the valves tested by EPRI. The plant and test valves are identical. The valve rated flow rate is 212,182 lbm/hr with a set pressure of 2500 psia.

The Dresser PORVs installed at Fort Calhoun were originally of the type 1 design with a 1-3/32 in. bore diameter. The valve tested by EPRI was a 31533VX-30 type 2 design with a 1-5/16 in. diameter. The type 2 design resulted from a need to improve the seat tightness and included modifications to the internals, body, and inlet flange. The body and flange modifications were not of a nature that would affect operability. The Fort Calhoun valves were modified to incorporate the internals of the type 2 design in May 1984. The difference in bore size will only affect valve capacity not operability. The test valve, therefore, is considered an adequate representation of the in-plant valves.

The Crane block valve is a 2-1/2 in. gate valve Model 787-U with a Limitorque SMB-00-7.5 operator. These block valves and operators were not tested by EPRI. A request for additional information was sent to the licensee regarding operability of the block valves on July 23, 1985 (Reference 14). The licensee's response dated March 1, 1986 (Reference 15) was that the block valve test program (Reference 9) was sufficient to demonstrate operability of the untested Crane block valves.

Figure 1

FORT CALHOUN PRESSURIZER SAFETY VALVE INLET PIPING CONFIGURATION[1]



(1) Planned for future modification

(Reference 16). The SV used for tests 516 and 517 had steam internals while the SV used for tests 535 and 537 had loop seal internals. The difference in the SVs with steam and loop seal internals is the use of different valve seat materials (Reference 5); therefore, tests with both internals are valid for evaluating the Fort Calhoun SVs. These tests had peak pressures ranging from 2436 to 2725 psia, pressurization rates from 3 to 267 psi/sec, and backpressures of 507 to 582 psia. These tests bound the Fort Calhoun conditions except for testing of the valve with a filled loop seal.

It is the staff position that Fort Calhoun 3K6 safety valve performance needs to be evaluated with a filled loop seal condition. Four cold loop seal tests were run by EPRI with the Crosby 3K6 SV (Reference 8) that can be applied to the Fort Calhoun conditions and ring settings of -115, -14 (Tests 525, 526, 529, and 536). Fort Calhoun initially had a cold (160°F) SV loop seal inlet which was changed to a short loop seal to reduce the dynamic loads on the piping in the event that the safety valve discharged. No indication of the temperature distribution in the modified loop seals has been provided. However, it is reasonable to assume that the temperature of the modified loop seals will not be colder than the original loop seals. These five loop seal tests were run with the following conditions: pressurization rate range was 3.3 to 220 psi/sec, peak pressure range was 2558 to 2708 psia, peak back-pressure ranging from 471 to 615 psia, and loop seal temperature ranging from 86 to 360°F. These test conditions bound those expected at Fort Calhoun.

Review of the CE inlet conditions report (Reference 7) showed that water did not reach the valve during FSAR transients or an extended high pressure injection (HPI) event. The cutoff head for the Fort Calhoun HPI pumps is below the SV setpoint so that an extended HPI event would not challenge the safety valves.

The CE inlet conditions report (Reference 7) did not list the Feedwater Line Break (FWLB) as a transient which challenged the safety valves or PORVs at Fort Calhoun. Because the FWLB is the limiting transient for delivering high temperature and high pressure water to the safety valve inlet in many PWRs, OPPD was asked a question on the FWLB at Fort Calhoun. OPPD stated in Reference 15 that the Loss of Load (LOLD) is the limiting transient at the Plant. They stated the FWLB is less severe than the Steam Line Break (SLB) and the LOLD bounds both of these accidents. The staff agrees that the FWLB is not as severe as the SLB in terms of break size. However, the SLB represents an overcooling event whereas the FLWB represents an event which is initially an overcooling event followed by an overheating event which can fill the pressurizer and cause high pressure liquid discharge through the safety valves. Therefore, the operability of the safety valves for the specific fluid conditions due to the FWLB must be addressed.

There was a concern that the extended safety valve blowdown (blowdown greater than 5%) observed during the EPRI tests could result in the pressurizer liquid level increasing to the safety valve inlet. CE reanalyzed the LOLD transient assuming a 20% blowdown for the Fort Calhoun plant. Two LOLD transient conditions were analyzed; one was with a normal SV opening and the other assumed a 4 sec delay for steam flow to account for loop seal passage. Other conservative assumptions were also made to maximize pressurizer level swell. The LOLD was chosen because it was the design basis accident which caused the greatest pressurizer pressure for Fort Calhoun. The 20% blowdown is representative since the blowdown observed in the applicable EPRI tests was 13.6 to 21.1%. These analyses showed the pressurizer level did not reach the inlet to the safety valves. Thus, the steam inlet condition was maintained.

The two Dresser PORVs at Fort Calhoun have cold loop seals. The loop seal temperature immediately upstream of the PORVs is between 100-120^{o}F. The peak pressure and pressurization rate for the PORVs during FSAR type transients are 2480 psia and 45 psi/sec, respectively as compared to 2530 psia and 40 psi/sec for the SVs (Reference 17). The maximum expected backpressure for the PORVs at Fort Calhoun was not provided by the licensee as suggested in the EPRI Submittal Guide (Reference 18). Therefore, a complete evaluation of the Fort Calhoun PORVs can not be made.

The test valve was subjected to two cold loop seal simulation tests and fifteen steam tests. In the steam tests, the peak pressure ranged from 2435 psia to 2505 psia. Backpressures ranged from 170 psia to 760 psia. The cold loop seal simulations were run with loop seal temperatures of 104 and 105°F, peak pressure of 2500-2505 psia, and 295-690 psia backpressure. The testing of the Dresser PORV was performed at opening pressures that are above the set pressure for Fort Calhoun for high pressure operation (2415 to 2507 psia versus 2400 psia). The opening pressure also bounded the peak pressure at Fort Calhoun (2480 psia). Except for the uncertainty of the expected peak backpressure, the test conditions are considered representative of the plant conditions.

As with the safety valves, the CE inlet conditions report (Reference 7) indicated that water did not reach the PORV during FSAR transients or an extended HPI event. The cutoff head for the Fort Calhoun HPI pumps is below the PORV setpoint so that an extended HPI event would not challenge the PORVs.

The PORVs are used for low temperature overpressure (LTOP) protection at Fort Calhoun. For low temperature overpressure protection, the valve is required to pass the loop seal water, steam at pressures from 465 to 750 psia, steam to water transition, and liquid at pressures from 465 to 750 psia with temperatures ranging from 100°F to 417°F. The peak pressures noted above are based on analyses that assumed the pressurizer is liquid full (Reference 7). The presence of a steam bubble in the pressurizer, which is the recommended mode of operation during low temperature operation, would limit the peak pressure when the PORV opened on steam, but this condition was not specifically analyzed. Thus, peak pressure during steam discharge was bounded using the liquid full analyses. The loop seal discharge and steam discharge conditions are considered to be adequately represented by the high pressure tests discussed above. Steam to water transition is also considered to be adequately represented by the high pressure transition test, 21-DR-8S/W. Water discharge during a LTOP transient is represented by the low pressure (~690 psia) water tests with fluid temperatures ranging from 112°F to 459°F.

Reference 9 did not provide test data for the Crane block valve at the plant and no test data or justification has been provided by the licensee. The block valve, however, is required to open and close over a range of steam and water conditions. The ability of the block valve operator to provide the required torque to open or close the valve should be verified. Since the PORV tested showed problems closing with cold loop seal conditions it is important that Fort Calhoun's block valves be able to function under all conditions. The licensee's response in Reference 15 is considered inadequate.

For the Fort Calhoun safety valves and PORVs, the test sequences and analyses described above, which demonstrate that the test conditions bounded the conditions for the plant valves, verify that Items 2 and 4 of Section 1.2 have been met (except for the expected peak backpressure for the PORVs and the fluid inlet conditions for the Feedwater Line Break). Items 2 and 4 require that conditions for the operational occurrences have been determined and the highest predicted pressures were chosen for the test. The part of Item 7, which requires showing that the test conditions are equivalent to conditions prescribed in the FSAR, is also met.

The licensee did not provide any test data or analyses for the plant block valves. Therefore Items 2, 4 and 7 of Section 1.2 have not been met with respect to the block valves.

4.3 Valve Operability

As discussed in the previous section, the Crosby 3K6 safety valves at Fort Calhoun are required to operate with a liquid filled loop seal followed by steam and/or liquid water. The Dresser PORVs are required to operate with a cold liquid filled loop seal followed by steam or liquid and also for LTOP conditions with the liquid filled loop seal followed by either steam or liquid. The fluid inlet conditions for the FWLB have not been provided, but otherwise the EPRI test program tested the Crosby 3K6 SV and the Dresser PORV for the required range of conditions. The Crane block valves are required to fully open and close for all possible plant flow and pressure conditions.

CE stated (Reference 16) that tests 516, 517, 535 and 537 are the most appropriate tests for Fort Calhoun. These four tests are steam only, long inlet piping tests where the test safety valve opened at 2435 to 2530 psia (-2.6 to +1.2% of the design set pressure of 2500 psia), had stable behavior, and closed with 13.6 to 21.1% blowdown. The test SV flow rates experienced during these four tests ranged from 99 to 104% of rated flow rate at 3% accumulation and 103 to 109% of rated flow rate at 6% accumulation. The ring settings for these tests were (-95, -14) and (-115, -14) and are similar to those at the plant (Reference 15).

The loop seal tests (525, 526, 529, and 536) are also applicable to the evaluation of the Fort Calhoun safety valves, particularly since they were all run with ring settings of -115, and -14). All but the last experienced chatter during loop seal discharge and stabilized during steam flow through the valve. The last test (536) was stable during loop seal discharge. Chatter during loop seal passage is expected, and causes water-hammer type pressure oscillations, but apparently did not damage the stellite valve disk and seat (used for loop seal applications) enough to cause the valve to malfunction. For these four loop seal tests, the opening pressure ranged from 2536 to 2637 psia (+1.4 to +5.5% above the 2500 psia set pressure), the blowdown ranged from 17.7 to 19.9%, the rated flow rate ranged from 96 to 100% at 3% accumulation and from 104 to 105% at 6% accumulation, and the lift ranged from 105 to 110%. The SV operation during these tests demonstrated that the plant SVs should operate stably. No tests were performed on a 3K6 SV with steam internals in a loop seal configuration. However, a steam/water transition test (428) and several water tests (431a and b, 435, and 438) indicate that the 3K6 SV may experience more leakage following discharge of water. These tests indicate that inspection and maintenance are important to the continued reliable operability of Crosby Model 3K6 safety valve.

A maximum bending moment of 161,500 in-lb was applied to the 3K6 valve discharge flange during test 441 without impairing valve operation. This bounds the maximum expected bending moment of 133,000 in-lb at the plant.

For a test to be an adequate demonstration of safety valve stability, the test inlet piping pressure drop should exceed the plant pressure drop. This is clearly the situation with Fort Calhoun. The plant valves were originally mounted on short liquid filled loop seal piping (6.31 ft) and all the long inlet piping 3K6 tests were performed with a >11 ft inlet piping configuration. The plant predicted inlet piping presssure drop is 325 to 339 psi; the pressure drop predicted for test 517 is 346 psi; therefore, the plant SV should be more stable than the test SV.

As noted above, the valve blowdown for the 3K6 valve during the applicable tests ranged from 13.6% to 21.1%. The Fort Calhoun LOLD analysis with 20% blowdown showed that the pressurizer liquid level would not reach the safety valve inlet. This approximates the blowdown observed in the test. Also, the hot leg remained subcooled during the LOLD analysis with the extended blowdown indicating adequate core cooling was maintained.

The Dresser PORV opened and closed on demand for all nonloop seal tests. During the warm loop seal test (22-DR-9W/W), with 321°F water at the valve inlet, the valve opened on demand but remained open for two seconds upon de-energizing the valve. During the cold loop seal tests (16-DR-6W and 24-DR-6W/W) with 103-105°F water at the valve inlet, the valve opened on demand but remained open for 70 sec for one test and until the block valve was closed for the other test (~90 sec). The Fort Calhoun loop seal temperature is in the 100-120°F range according to Reference 12.

Inspection of the valve after testing at the Marshall Steam Station showed the bellows had several welds partially fail. The failure did not affect valve performance and the manufacturer concluded the failure did not have a potential impact on valve performance. The bellows was replaced and did not fail during any of the additional test series.

A bending moment of 25,500 in-lb was induced on the discharge flange of the test valve without impairing operability. According to the licensee, the maximum bending moment calculated for the Fort Calhoun PORVs is less than this value (Reference 12). The licensee did not provide the moment on the PORV when requested (Reference 15).

The Fort Calhoun PORVs are pilot operated valves that use system pressure to hold the disk tight against the seat. At one point Dresser Industries recommended the block valve be closed at system pressures below 1000 psig to avoid steam wirecutting of the PORV disk and seat. Testing by Dresser later showed the 1000 psig pressure limit to be overly conservative and that the PORV as designed was qualified to system pressures of 100 psig. Below 100 psig the deadweight of the lever on the pilot valve was sufficient to keep the valve open. Dresser recommends that heavier springs be used under the main and pilot disks to ensure closure if the plant is to operate at pressures below 100 psig. Without the heavier springs recommended by Dresser, the PORV should not be used at system pressure below 100 psig. Fort Calhoun has modified its PORVs (Reference 15) to include type 2 internals but has not installed the heavier springs. Therefore, the PORVs may leak during low pressure operation while they provide low temperature overpressure protection. The staff position regarding the PORVs is that since they are installed and available for use, they should be modified as recommended by Dresser.

The licensees submittal regarding operability of the PORVs is not considered adequate. The performance of the PORV during EPRI tests, under the full range of expected inlet conditions, indicates that the PORVs may not close properly with the cold loop seals. The installation of type 2 internals will improve operability at pressures greater than 100 psig, if the cold loop seal were not a problem. In addition, until the heavier springs are installed under the main and pilot disks the valve is not considered operable below 100 psig.

The PORV block valve must be capable of closing over a range of steam and water conditions. The Crane block valve with its Limitorque SMB-00-7.5 operator were not tested by EPRI. The licensee's response in Reference 15 is considered inadequate considering the fact that the PORVs are not qualified.

NUREG-0737 Item II.D.1 requires qualification of associated control circuitry as part of the safety/relief valve qualification. Fort Calhoun PORV control circuitry is primarily outside the containment and not subject to severe service conditions; however, the solenoid valves are in containment. The licensee states that operation of its PORVs is not critical to the plant safety and that a stuck open PORV is within the small LOCA class accident which has been analyzed.

It is the NRC staff's position that the PORV control circuitry must be qualified to perform its required function for any potential environment that it may by exposed to, and that OPPD has not satisfied the requirements of NUREG-737 Item II.D.1 with respect to qualification of the PORV control circitry Therefore, if OPPD takes any exception with respect to qualification of the PORV control circuitry to the harsh environment for equipment important to safety, they must demonstrate that the equipment (PORV) is not required to perform a safety function to mitigate the effects of any design basis accident when exposed to the environment caused by that accident, and any equipment failure in any mode in the harsh environment will not adversily impact safety functions or mislead the operator.

The information provided by the licensee demonstrates that the safety valves will operate satisfactorily provided that OPPD develops procedures to inspect and maintain the safety valves following each lift. The licensee has conducted tests to qualify the safety valves (Item 1 of Section 1.2), and has considered the effect of discharge piping on safety valve operability (Item 7 of Section 1.2).

However, this information does not demonstrate that the PORVs, PORV control circuitry, and PORV block valves are reliably operable.

4.4 Piping and Support Evaluation

The licensee stated that the SV and PORV inlet piping and the piping between the valve discharge flanges and the pressurizer relief tank were analyzed. These analyses included the piping supports. The SV inlet piping and the SV and PORV outlet piping were all initially overstressed (Reference 12) and required some modifications. The acceptance criteria used in these analyses was not identified. The overstressed conditions were identified and modifications were planned and made. OPPD also stated that a reanalysis of all the modified piping was done that showed all stresses and loads were acceptable. However, no details of the acceptance criteria and analysis were provided. The licensee's response to a request for additional information (Reference 15) was insufficient to determine the adequacy of the SV and PORV piping and supports. A summary of the licensee's submittal follows.

In Reference 12, OPPD stated that they were considering reducing the SV loop seal volume and increasing the loop seal temperature to reduce the SV discharge loads. In Reference 13, the licensee stated that the PORV downstream piping was modified prior to startup from the 1983 refueling outage to reduce the possible stresses to acceptable levels, and that modifications of the SV downsteam piping were planned for completion before startup from the 1984 refueling outage. However, the licensee did not respond to question 10 (See Reference 15) requesting information regarding the modifications being considered. The licensee merely stated that the discharge piping was analysed in its post modification configuration to verify the piping adequacy and that those analyses were checked and verified by an outside consultant.

The thermal-hydraulic analyses of the SV/PORV piping systems refered to in Reference 15 was done using the RELAP5/MOD1 code. This code was verified for use on SV/PORV piping in Reference 19. 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). The SV and PORV rated flow rates were used rather than the actual test or predicted flow rates. This is acceptable since the measured flow during EPRI testing ranged from 99 to 104% of rated flow at 3% accumulation. Not performing a case where both safety valves lift simultaneously is not acceptable. During the loop seal tests, the opening pressures for the valves ranged from +1.4 to +5.5% of set pressure, and during the steam tests the opening pressures ranged from -2.6 to +1.4% of set pressure. 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. In addition, no details of the analysis was presented. Therefore, not only is the analysis considered inadequate, it is impossible to verify that the parameters used in the RELAP5 analysis were appropriate.

In Reference 15, the licensee stated that hydraulic loads were calculated using the FORCE code in conjunction with the RELAP5 output. TPIPE was then used to do the piping analysis where piping stresses and hanger loads were predicted. These analyses were done by OPPD and checked by Impell Corp. No other details were provided even though verification of the codes was requested in Reference 14.

The analyses discussed above, do not demonstrate that a bounding case has been chosen for the piping configuration, which would verify that Item 3 of Section 1.2 has been met. In addition, since the licensee did not describe the modified piping and support systems, the analyses performed, the load combinations, and a comparison of maximum to allowable stresses it can only be concluded that Item 8 of Section 1.2 has not been met.

5. EVALUATION SUMMARY

The Licensee for Fort Calhoun has not provided an acceptable response to the requirements of NUREG-0737, reconfirming that the General Design Criteria 14, 15, and 30 of Appendix A to 10 CFR 50 have been met with regard to the safety valves and PORVs. The rationale for this conclusion is given below.

The licensee participated in the development and execution of an acceptable relief and safety valve test program to qualify the operability of prototypical valves and to demonstrate that their operation would not invalidate the integrity of the associated equipment and piping. The subsequent tests were successfully completed under operating conditions which, by analysis, bound the most probable maximum forces expected from anticipated design basis events. The test results showed that the safety valves tested functioned correctly and safely for all steam discharge events specified in the test program that were applicable to Fort Calhoun. Analysis and review of both the test results and the licensee justifications indicated the performance of the safety valves can be directly extended to the in-plant valves.

However, the tests demonstrated the need for inspection and maintenance for reliable continued operability of the safety valve. Therefore, the licensee must inspect the safety valve after each lift and a formal procedure requiring inspection and refurbishment must be developed and incorporated into the plant operating procedures. In addition, the operability of the safety valves for liquid water fluid inlet conditions during a feedwater line break accident has not been addressed.

The test results demonstrated that the PORVs might not close properly following operation, and therefore can not be considered reliably operable. In addition, the requirement that the PORV control circuitry be qualified has not been met, and Fort Calhoun has PORV block valves that were not tested in the EPRI test program. Finally, the licensee has not demonstrated by analysis that the plant specific piping is acceptable.

Thus, the requirements of Item II.D.1 of NUREG0737 to ensure that the reactor primary coolant pressure boundary will have a low probability of abnormal leakage (General Design Criterion No. 14), that the reactor primary coolant pressure boundary and its associated components (piping, valves, and supports) have been designed with a sufficient margin so that design conditions are not exceeded during relief/safety valve events (General Design Criterion No. 15), and that this equipment was constructed in accordance with high quality standards (General Design Criterion No. 30) have not been met.

6. REFERENCES

1. TMI-Lessons Leaned 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. D. 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", LTC-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, Safety and Relief Valve Test Program," LIC-82-182, August 2, 1983.

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, NUREG-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.

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13. ABSTRACT (200 words or less)

Light water reactors have experienced a number of occurrences of improper performance of safety and relief valves installed in the primary coolant system. As a result, the authors of NUREG-0578 (TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations) and subsequently NUREG-0737 (Clarification of TMI Action Plan Requirements) recommended that programs be developed and completed which would reevaluate safety, relief, and block valves and which would verify the integrity of the piping systems for normal, transient, and accident conditions. This report documents the review of these programs by the Nuclear Regulatory Commission (NRC) and their consultant, EG&G Idaho, Inc. Specifically, this report documents the review of the Fort Calhoun Licensee response to the requirements of NUREG-0578 and NUREG-0737. This review found that the licensee, Omaho Public Power District has not provided an acceptable submittal reconfirming that the General Design Criteria 14, 15, and 30 of Apprndix A to 10 CFR 50 have been met.

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