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TECHNICAL EVALUATION REPORT
TMI ACTION--NUREG-0737 (II.D.1)
RELIEF AND SAFETY VALVE TESTING
COMANCHE PEAK - UNIT 2
DOCKET NO. 50-446

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ABSTRACT

In the past, safety and relief valves installed in the primary coolant system of light water reactors have performed improperly. 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 development and completion of programs to do two things. First, the programs should reevaluate the functional performance capabilities of pressurized water reactor safety, relief, and block valves. Second, they should verify the integrity of the pressurizer safety and relief valve piping systems for normal, transient, and accident conditions. This report documents the review of those programs by EG&G Idaho, Inc. Specifically, this report documents the review of the Comanche Peak, Unit 2, Applicant response to the requirements of NUREG-0578 and NUREG-0737. This review found the Applicant provided an acceptable response reconfirming they met General Design Criteria 14, 15, and 30 of Appendix A to 10 CFR 50 for the subject equipment.

FIN L1354 - Technical Assistance for TVA Reviews

Summary

The failure of a power-operated relief valve (PORV) to reseat was a significant contributor to the Three Mile Island sequence of events. This failure, plus other previous instances of improver valve performance, led the task force which prepared NUREG-0578 and NUREG-0737 to recommend development of programs to do two things. First, the programs should reexamine the functional performance capabilities of pressurized water reactor safety, relief, and block valves. Second, they should verify the integrity of the pressurizer safety and relief valve piping systems for normal, transient, and accident conditions. The task force deemed this necessary to reconfirm that Licensees and Applicants satisfied General Design Criteria (GDC) 14, 15, and 30 of 10 CFR 50, Appendix A, for the subject equipment.

This report documents the EG&G Idaho, Inc., review of the Comanche Peak, Unit 2, Applicant response to the above NUREG requirements. EG&G Idaho reviewed: (a) the Applicant's submittals to determine the applicability of the test valves and conditions to the plant valves and inlet conditions, (b) the operability of the test valves to determine the operability of the plant valves, and (c) the Applicant's analysis of the pressurizer discharge piping to determine if they met acceptable stress limits for valve discharge transients.

The Applicant met the requirements of NUREG-0578 and NUREG-0737. They participated in an acceptable test program. The tests were successful and completed under operating conditions which bounded the most probable maximum forces expected from anticipated design basis events. The test results showed the valves tested functioned correctly and safely for all steam and water discharge tests applicable to Comanche Peak, Unit 2. Also, the pressure boundary component design criteria were not exceeded. Review of the Applicant's justifications indicated direct applicability of the test valve performance to the in-plant valves and systems represented by the test program. Texas Utilities Electric Company's analysis of the plant specific piping showed it met code allowables. Thus, the Applicant reconfirmed they met GDC 14, 15, and 30 of Appendix A to 10 CFR 50 for the subject equipment.

PREFACE

EG&G Idaho, Inc., Regulatory and Technical Assistance Programs Unit, prepared this report for the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Reactor Regulation.

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1. INTRODUCTION

1.1 Background

In the past, safety and relief valves installed in the primary coolant system of light water reactors have performed improperly. There were instances of valves opening below set pressure, valves opening above set pressure, and valves failing to open or reseat. From the past instances of improper valve performance, it is not known whether they occurred because of limited valve qualification or because of a basic unreliability in 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 sequence of events. These facts led the task force which prepared NUREG-0578 (Reference 1) and, subsequently, NUREG-0737 (Reference 2) to recommend development and completion of programs to do two things. First, the programs should reexamine the functional performance capabilities of pressurized water reactor (PWR) safety, relief, and block valves. Second, they should verify the integrity of the pressurizer safety and relief valve piping systems for normal, transient, and accident conditions. The task force deemed this necessary to reconfirm that Licensees and Applicants satisfied General Design Criteria 14, 15, and 30 of 10 CFR 50, Appendix A, for the subject equipment.

1.2 General Design Criteria and NUREG Requirements

General Design Criteria 14, 15, and 30 require: (a) the reactor primary coolant pressure boundary be designed, fabricated, and tested so as to have an extremely low probability of abnormal leakage; (b) 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 operational occurrences; and (c) the components, which are part of the reactor coolant pressure boundary, be constructed to the highest quality standards practical.

To reconfirm the integrity of overpressure protection systems and thereby assure compliance to the General Design Criteria, the Division of Licensing, Office of Nuclear Reactor Regulation, issued the NUREG-0578 position as a requirement in a letter dated September 13, 1979, to all operating nuclear power plants. The NRC incorporated this requirement as Item II.D.1 of NUREG-0737, Clarification of TMI Action Plan Requirements, which they issued for implementation on October 31, 1980. As stated in the NUREG reports, each PWR 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 pressures 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 NRC staff review and evaluation, including criteria for success or failure of valves tested.

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 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.

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

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. The test program covered pressurizer PORV block valves, PORVs, safety valves, and associated piping systems. Texas Utilities Electric Company (TUEC), the owner of Comanche Peak, Unit 2, was one of the utilities sponsoring the EPRI Safety and Relief Valve Test Program. In Reference 3, the participating utilities transmitted to the NRC the series of reports containing the results of the program. This section discusses the applicability of those reports below.

In Reference 4, EPRI developed a plan for testing PWR safety and relief valves under conditions which bound actual plant operating conditions. Through the valve manufacturers, EPRI identified the valves used in the overpressure protection systems of the participating utilities. They then selected representative valves for testing. The valves selected included enough of the variable characteristics so that their testing would adequately demonstrate the performance of the valves used by utilities (Reference 5). Through the nuclear steam supply system vendors, EPRI evaluated the FSARs of the participating utilities. They then developed a test matrix which bounded the inlet conditions for the plant transients that require overpressure protection (Reference 6).

The utilities participating in the EPRI Safety and Relief Valve Test Program also tested PORV block valves (Reference 7). The Electric Power Research Institute developed a list of valves used or intended for use in participating PWR plants. They selected seven block valves to represent the block valves used in PWR plants. Westinghouse Electro-Mechanical Division (WEMD) performed additional tests on valve models they manufacture (Reference 8).

Westinghouse, under contract to EPRI, produced a report on pressurizer safety and relief valve inlet conditions in Westinghouse designed plants (Reference 9). Because Comanche Peak, Unit 2, is a Westinghouse designed plant, that report is applicable to this evaluation.

The Electric Power Research Institute sponsored several test series. They tested PORVs and block valves at the Duke Power Company Marshall Steam Station located in Terrell, North Carolina. Only steam tests were conducted at the Marshall Station. Therefore, EPRI tested block valves at Marshall only for full flow, full pressure steam conditions. Westinghouse (WEMD) performed water flow tests on four valve models they manufacture. The Electric Power Research Institute conducted additional PORV tests at the Wyle Laboratories Test Facility located in Norco, California. They tested safety valves at the Combustion Engineering Company Kressinger Development Laboratory located in Windsor, Connecticut. In Reference 10, EPRI reported the results of the relief and safety valve tests. They reported the results of the block valve tests in References 7 and 8.

The EPRI test program's primary objective was to test each of the various types of primary system safety valves used in PWRs for the full range of expected inlet fluid conditions. The test program limited the conditions selected for test (based on analyses) to steam, subcooled water, and steam to water transition. Additional objectives were to: (a) obtain valve capacity data, (b) assess hydraulic and structural effects of associated piping on valve operability, and (c) obtain piping response data for verifying analytical piping models.

The EPRI test program did not provide information on valve reliability. The EPRI program plan (Reference 4) states, "During the course of the specified tests, each valve will be subjected to a number of operational cycles. However, it should be noted that the test program, to be completed by July, 1981, is not intended to provide valve lifetime, cyclic fatigue or statistical reliability data."

Reference 11 contains NRC staff approval of the EPRI test program. The staff concluded the EPRI program produced enough generic valve performance information for utilities to meet the plant specific information requirements in NUREG-0737, Item II.D.1. Transmittal of the test results meets Item 6 (provide test data to the NRC) of Section 1.2 in this report.

3. PLANT SPECIFIC SUBMITTAL

Texas Utilities Electric Company submitted their Comanche Peak, Unit 2, evaluation report for NUREG-0737, Item II.D.1, in four parts. The submittal dates were, March 31, 1982 (Reference 12), May 18, 1992 (Reference 13), November 13, 1992 (Reference 14), and December 18, 1992 (Reference 15).

4. REVIEW AND EVALUATION

4.1 Valves Tested

Comanche Peak, Unit 2, uses three safety valves, two PORVs, and two PORV block valves in the overpressure protection system. The safety valves are Crosby Model HB-BP-86 6M6 valves with loop seal internals. The PORVs are 3-inch Copes-Vulcan Model D-100-160 air-operated globe valves with 316 SS stellited plugs and 17-4 PH cages. The safety valves have hot loop seals and the PORVs have cold water seals upstream of the valves. The block valves are Westinghouse Model 3GM88 motor operated gate valves with Limitorque SB-00-15 motor operators.

The Electric Power Research Institute tested the safety valve model used at Comanche Peak, Unit 2, the Crosby Model HB-BP-86 6M6 valve. At Comanche Peak, Unit 2, TUEC mounted the safety valves on loop seal piping with a hot loop seal upstream of the valve. The valve internals are for loop seal service. The test valve also had loop seal internals, and EPRI tested it on loop seal piping with a hot loop seal. In Reference 13, TUEC stated the Crosby 6M6 valves at Comanche Peak, Unit 2, use factory set ring settings. Therefore, TUEC can use the results from the EPRI tests with factory ring settings to demonstrate operability of the plant safety valves.

The PORVs at Comanche Peak, Unit 2, are the same design as one tested by EPRI. They tested the valve with a cold loop seal. Because there is no difference between the test and plant valves, the test results are directly applicable to Comanche Peak, Unit 2.

The block valves used at Comanche Peak, Unit 2, are the same design as one of the EPRI test valves, the Westinghouse 3GM88 block valve. The Electric Power Research Institute tested the valve in a horizontal configuration. Texas Utilities Electric Company installed the plant valve in the same configuration (Reference 13). The test valve had a Limitorque SB-00-15 motor operator, and the plant valves use the same Limitorque operator. During EPRI testing, the 3GM88 block valve fully closed only when the operator produced a torque of 182 ft-lb. Based on Reference 14, TUEC modified Unit 2 block valve operators to close on limit rather than torque to ensure complete valve closure. In this mode of operation, the operator torque output is greater than 182 ft-lb. The test valve is, therefore, representative of the plant valves.

Based on the above, the test valves represent the Comanche Peak, Unit 2, valves and fulfill the requirements of Items 1 and 7 of Section 1.2 in this report regarding applicability of the test valves.

4.2 Test Conditions

As stated earlier, Westinghouse Electric Corp. designed Comanche Peak, Unit 2. Reference 9 lists the valve inlet fluid conditions that bound the inlet conditions for overpressure transients in Westinghouse plants. In Reference 14, TUEC stated they verified the inlet conditions in the Westinghouse report are still applicable to Comanche Peak, Unit 2. The applicable inlet conditions in Reference 9 are those identified for four-loop plants. The transients considered in this report include FSAR, extended high pressure injection, and low temperature overpressurization events. This section discusses the expected inlet conditions for each of these events and the applicable EPRI tests.

4.2.1 FSAR Steam Transients

For Comanche Peak, Unit 2, the limiting FSAR steam discharge transients when only the safety valves open are the loss of load event and the locked rotor event. These same events are limiting for steam discharge when both the safety valves and PORVs open. The loss of load event gave the maximum pressurizer pressure and the locked rotor event gave the maximum pressurization rate.

When the safety valves open alone, the predicted maximum pressurizer pressure and maximum pressurization rate are 2555 psia and 144 psi/s, respectively. The maximum developed backpressure in the outlet piping is less than 515 psia (Reference 14). Texas Utilities Electric Company insulated the loop seal so the valve inlet temperature is 300°F (Reference 13).

The insulation used to maintain the loop seal temperature in Unit 2 is the same as that in Unit 1 (Reference 14). In Reference 14, TUEC stated they field measured the Unit 1 loop seal temperature as 314°F. Because both units have the same loop seal insulation, EG&G Idaho, Inc., concluded the Unit 2 loop seal temperature should also exceed 300°F.

For steam flow conditions, four loop seal discharge tests on the Crosby 6M6 valve (Test Nos. 929, 1406, 1415, 1419) are applicable to Comanche Peak, Unit 2. These tests used valve ring settings representative of those used in Westinghouse PWRs including Comanche Peak, Unit 2. The ring settings used in these tests were (-71, -18) or (-77, -18). These represent the upper and lower ring positions measured from the level position referenced to the bottom of the disc ring. In Reference 13, TUEC stated the ring settings used at Comanche Peak, Unit 2 are -82 to -103 (upper ring), and -18 (lower ring) relative to the level position. Also in Reference 13, TUEC stated Crosby Valve and Gage Co. determined both the test and in-plant ring settings using similar methods and standard of performance. Therefore, EG&G Idaho considers these ring settings comparable.

The loop seal temperature measured in the tests ranged from 90 to 350°F at the valve inlet. The maximum test (tank 1) pressures were in the range of 2675 to 2760 psia and the pressurization rate was 90 to 360 psi/s. The backpressures developed in the tests were 245 to 710 psia. The above data show that the test conditions envelope the corresponding data for the Comanche Peak, Unit 2, safety valves. Table 4.2.1 summarizes this comparison.

When both the safety valves and PORVs open, the maximum predicted pressurizer pressure is 2532 psia and the maximum pressurization rate is 130 psi/s. The loop seal temperature is 150°F at the PORV inlet.

In the EPRI tests on the Copes-Vulcan PORV, the maximum steam pressure at valve opening was 2715 psia. This bounds the predicted pressure at Comanche Peak, Unit 2. In the loop seal test, the temperature at the valve inlet was 134°F. The backpressure developed at the outlet of the PORVs is not an important consideration for Comanche Peak, Unit 2. This is because the air operated PORVs used at Comanche Peak, Unit 2, are not sensitive to

TABLE 4.2.1 SUMMARY OF TEST DATA FOR CROSBY 6M6 SAFETY VALVE AND COMPARISON WITH COMANCHE PEAK, UNIT 2, REQUIREMENTS

Valve	Test Number	Test Type	Test Conditions	Inlet Conditions	Initial Fluid Temperature at Valve Inlet (^oF)	Safety Valve Ring Settings[1]	Pressure at Valve Opening[2] (psia)	Peak Tank Pressure (psia)	Peak Back-pressure (psia)	Percent Blowdown	Peak Pressurization Rate (psi/s)	Valve Stability	Inlet Press. Drop (psi)
6M6-Plant Valve		FSAR Steam Trans-ient	---	Hot loop seal	300	-82, -18 -103, -18	2500	2555	515	Nominally 5.0	144.0	---	269.0
6M6-Loop seal internals	929		Steam	Cold loop	90	-71, -18	2600	2726	710	5.1	319.0	Stable	263.0
	1406		Steam	Cold loop	147	-77, -18	2530	2703	250	9.4	325.0	Stable	263.0
	1415		Steam	Hot loop	290	-77, -18	2555	2760	255	6.2	360.0	Stable	263.0
	1419		Steam	Hot loop	350	-77, -18	2464	2675	245	---	360.0	Chatter[3]	263.0
6M6-Plant Valve[4]		FSAR Liquid Trans-ient	---		Sat	-82, -18 -103, -18	2500	2503	515	Nominally 5.0	5.0	---	269.0
6M6-Loop seal internals	931a[5]		LS/ Trans Water	Cold loop	117	-77, -18	2570	2578	725	12.7	2.5	Stable	263.0
	931b			Hot loop	635	-77, -18	2475	2475	700	4.8	2.5	Chatter[6]	263.0

1. The plant and test valve ring settings are relative to the level position. The plant guide ring settings, -82 and -103, represent the range of guide ring settings for the plant valves.
2. The set pressure of the test valves was 2485 psig.
3. This test was terminated because of valve chatter.
4. The maximum liquid surge rate during a feedwater line break is 1109.5 gpm.
5. The maximum liquid flow rate during test 931a was 2355 gpm.
6. The valve chattered during opening but then stablized.

backpressure (Reference 6). Therefore, the EPRI test inlet fluid conditions for the PORV with steam discharge represent the plant specific transient conditions.

4.2.2 FSAR Liquid Transients

The limiting FSAR transient resulting in liquid discharge through the PORVs and safety valves is the main feedline break accident (Reference 9). In a feedline break accident at Comanche Peak, Unit 2, the calculated safety valve inlet conditions during water discharge are maximum pressure, 2503 psia, pressurization rate, 5 psi/s, and maximum pressurizer surge rate, 1109.5 gpm (~369,000 lbm/hr) liquid at 608-615°F. In a feedline break accident resulting in safety valve actuation, steam and steam to water transition flows always precede water discharge.

Tests 931a and 931b on the 6M6 valve included loop seal/steam, steam to water transition, and water discharge conditions. The valve ring settings and inlet pipe configuration used in these tests are representative of the in-plant safety valves. In Test No. 931a, the maximum inlet pressure was 2578 psia. The pressurization rate was 2.5 psi/s, the inlet loop seal fluid temperature was 117°F and the tank fluid temperature was 635°F. After the valve closed in Test 931a, the system repressurized and the valve cycled on approximately 635°F water for Test 931b. The inlet temperature and pressure of these tests bound the predicted in-plant condition. Therefore, EG&G Idaho considers these tests representative of the Comanche Peak, Unit 2, safety valve inlet conditions. Table 4.2.1 also summarizes the inlet fluid conditions and corresponding test data for liquid discharge.

Westinghouse based the expected safety valve inlet fluid conditions on an analysis that assumed the PORVs did not open during the feedline break transient. If the PORVs open, however, the same fluid conditions postulated for the safety valve inlet will occur at the PORV inlet (Reference 6). In the tests, EPRI performed high temperature water discharge and steam to water transition tests with the Copes-Vulcan PORV. In the water discharge test, Test No. 76-CV-316-2W, the maximum valve inlet pressure was 2535 psia and the temperature was 647°F. In the transition test, Test No. 77-CV-316-7S/W, the maximum inlet pressure was 2532 psia and the water temperature was 657°F. The inlet fluid conditions for these tests bound the expected inlet conditions for Comanche Peak, Unit 2. Therefore, EG&G Idaho considers these tests adequate to represent the in-plant PORV performance in the feedline break event.

4.2.3 Extended High Pressure Injection Event

The limiting extended high pressure injection event is the spurious actuation of the safety injection system at power (Reference 9). For a four-loop plant, an extended high pressure injection event challenges both the safety valves and PORVs. Valve inlet conditions include both steam and water discharge. In this event, however, the safety valves or PORVs open on steam, and liquid discharge would not occur until the pressurizer becomes water solid. According to Reference 9, this would not occur until at least 20 minutes into the event which allows ample time for operator action. Thus, EG&G Idaho disregarded the potential for liquid discharge in extended HPI events.

4.2.4 Low Temperature Overpressurization (LTOP) Transient

Texas Utilities Electric Company uses the PORV for overpressure protection during low temperature reactor startup and shutdown operations. The PORV low pressure setpoint varies with valve inlet temperature. The setpoint ranges from 445 to 2350 psig for inlet temperatures of 70 to 450°F (Reference 13). Reference 9 identified the expected inlet fluid conditions for LTOP transients, and they range from cold water to steam.

For steam discharge through the PORV, the high pressure steam tests discussed in Section 4.2.1 would cover the low pressure steam conditions predicted for LTOP transients. For water discharge conditions, EPRI performed two low pressure and low temperature water tests on the Copes-Vulcan PORV with stellited plug and 17-4 PH cage. The tests had an inlet pressure of 675 psia and water temperatures of 105°F and 442°F, respectively. EG&G Idaho considers these conditions representative of those at Comanche Peak, Unit 2. Therefore, EG&G Idaho will use the EPRI tests to evaluate the performance of the Comanche Peak, Unit 2, PORV for LTOP transients.

4.2.5 Block Valve Inlet Conditions

The block valves operate over a range of fluid conditions (steam, steam-to-water, water) similar to those of the relief valves. However, EPRI tested the block valves only under full pressure steam conditions (to 2420 psia). For Westinghouse manufactured valves, WEMD performed additional water flow tests. The WEMD test conditions ranged from 60 to 600 gpm and 1500 to 2600 psi differential pressure. Based on Reference 8, Westinghouse found four things concerning valves with similar internal materials. Westinghouse found that under full pressure steam conditions the required torque to open or close the valve:

(1) Depends almost entirely on the differential pressure across the valve disk.
(2) Is rather insensitive to momentum loading.
(3) Is nearly the same for water or steam.
(4) Is nearly independent of the flow.

Thus, full pressure steam tests are adequate to show valve operability for steam and water conditions.

4.2.6 Other Transients

Two transient conditions not part of the design basis are anticipated transients without scram (ATWS) and feed and bleed decay heat removal. This review did not consider the response of the overpressure protection system to these two transient conditions. Neither the Applicant nor the NRC have evaluated the performance of the system for these events.

4.2.7 Inlet Conditions Summary

The presentation above demonstrates that the test conditions bounded the conditions for the plant valves. It verifies TUEC met Items 2 and 4 of Section 1.2 in this report. That is, TUEC determined conditions for operational occurrences and chose the highest predicted pressures for the tests. The presentation also verifies that TUEC met the portion of Item 7 that requires showing test conditions are equivalent to those prescribed in the FSAR.

4.3 Operability

4.3.1 Safety Valves

The steam discharge tests representative of the Comanche Peak, Unit 2, conditions are loop seal tests, Test Nos. 929, 1406, 1415, 1419, on the Crosby 6M6 valve. In these tests (except Test No. 1415), the valve fluttered or chattered during loop seal discharge and stabilized when steam flow started. The valve opened within $\pm$4% of the design set pressure and closed with 5.1 to 9.4% blowdown. The valve achieved up to 111% of rated flow at 3% accumulation with valve lift positions at 92 to 94% of rated lift. These tests demonstrated that the valve performed adequately in spite of the initial chatter during loop seal discharge.

In Test 1419, the valve chattered on closing and the operators ended the test by manually opening the valve to stop the chatter. This result does not indicate a valve closing problem for the Comanche Peak, Unit 2, safety valve. This is because a similar test (Test 1415) had already demonstrated that the valve performed satisfactorily and exhibited no sign of instability. The closing chatter in Test 1419 may be a result of the repeated actuation of the valve in loop seal and water discharge tests. As shown in Table 4.3.1, EPRI performed seventeen steam, water, and transition tests on the 6M6 valve. In the first four or five tests, the valve fluttered and chattered during loop seal discharge but stabilized and closed successfully. After Test 913, there were four instances in which the operators stopped the test due to chatter on closing. The Electric Power Research Institute found galled guiding surfaces and damaged internal parts during inspection. They refurbished or replaced the damaged parts before the next test started. After each repair, the valve performed well, but the closing chatter recurred in the subsequent test. The Electric Power Research Institute performed Test 1415 immediately after valve maintenance and the valve performed stably. The next test (Test 1419) chattered on closing even though it was a repeat of Test 1415 at similar fluid conditions. This suggests that inspection and maintenance are important to the continued operability of the valves. The Applicant should develop a

TABLE 4.3.1 EPRI TESTS ON CROSBY HB-BP-86 6M6 SAFTEY VALVE

Seqn. No.	Test No.	Ring Setting	Test Type	Actions Taken Between Tests	Stability	Leakage Pre (gpm)	Leakage Post (gpm)
1	903	1	Steam		Stable	0	0
				Inspection/repair			
2	906a,b,c	1	L.S.		Stable	0	0
3	908	1	L.S.		f/c	0	0
4	910	1	L.S.		f/c	0	0
				Inspection/repair			
5	913	2	L.S.		f/c	0	1.0
6	914a,b,c	2	L.S. Transition		Terminated	0	Large
				Inspection/repair			
7	917	3	L.S.		f/c	0	0
8	920	3	L.S.		Terminated	0	0
				Inspection/repair			
9	923	3	L.S.		f/c	0	0
10	926a,b,c,d	3	Transition		Stable	0.36	0.08
				Inspection/repair			
11	929	4	L.S.		f/c	0	0
12	931a,b	4	L.S. Transition		c	0	0
13	932	4	Water		Terminated	0	--
				Inspection/repair			
14	1406	4	L.S.		f/c	0	0.63
				Inspection/repair			
15	1411	4	Steam		Stable	0.76	0.37
				Inspection/repair			
16	1415	4	L.S.		Stable	0	0
17	1419	4	L.S.		Terminated	0	1.5
				Inspection/repair			

c--chatter
f/c--flutter/chatter
L.S.--loop seal
Ring setting--four different ring settings were tested. Actual ring settings not shown.
Terminated--Test terminated after valve manually opened to stop chatter.

formal procedure to require inspection of the safety valves after each actuation. They should include the procedure in the plant operating procedures or licensing documents.

Texas Utilities Electric Company provided calculated values for the inlet pressure drop on valve opening and closing. They compared the plant specific values to the test values in Reference 13. The plant opening and closing pressure differences were 255-269 psi and 152-158 psi, respectively. The corresponding test pressure differences were 263 psi (valve opening) and 181 psi (valve closing). Based on this information, the plant valves should be as stable as the test valves.

As discussed in Section 4.2.2, the limiting FSAR transient resulting in liquid discharge is the main feedline break accident. Tests 931a and 931b represent Comanche Peak, Unit 2, feedwater line break conditions. Test 931a was a loop seal/steam/water transition test. The test valve opened, fluttered or chattered with partial lift during loop seal discharge, then popped open and stabilized on steam. The valve closed with 12.7% blowdown. Test 931b was a saturated water test. The 6M6 valve opened on 640°F water, chattered, and then stabilized. The valve closed with 4.8% blowdown. For these tests, the valve opened within -1% and +3% of the set pressure. The maximum calculated surge rate at Comanche Peak, Unit 2, during the feedline break transient is 1109.5 gpm. The EPRI 6M6 test valve passed 2355 gpm at 2415 psia and 641°F. This flow is much higher than the predicted liquid surge rate for Comanche Peak, Unit 2. The above results demonstrate that the Crosby 6M6 safety valves would adequately perform the required water relief function.

From the above steam and water results, the maximum observed blowdown in the applicable EPRI tests was 12.7%. This exceeds the design value of 5%. Thus, TUEC must demonstrate that extended blowdown will not impact plant safety and valve operability. They provided this information in Reference 15. Texas Utilities Electric Company stated they evaluated the impact of 13% blowdowns on the Comanche Peak, Unit 2, licensing basis safety analyses. They noted:

1. Extended safety valve blowdown of up to 13% will not cause the pressurizer to fill in any licensing basis event where the pressurizer does not already become water solid.

2. Extended safety valve blowdown of up to 13% will not challenge any safety systems which were not previously challenged in the licensing basis safety analyses.

3. Extended blowdown of up to 13% will not cause voiding of the primary system in any licensing basis event.

Therefore, the extended blowdown observed in the EPRI tests does not impact plant safety or valve operability.

The loads induced on the safety valve tested by EPRI exceed the loads for Comanche Peak, Unit 2. The maximum bending moment on the 6M6 test valve discharge flange was 298,750 in-lb during Test 908. Application of this bending moment did not affect test valve performance. The largest moment predicted for the safety valve inlet or outlet at Comanche Peak, Unit 2, is 172,428 in-lb. All valve nozzle loads are evaluated for the combined effects of deadweight, thermal expansion, safe shutdown earthquake (SSE), and valve actuation loads. Based on this, EG&G Idaho expects the plant valve to operate satisfactorily with the maximum expected plant bending moment.

4.3.2 <u>Power Operated Relief Valves</u>

The EPRI tests on the Copes-Vulcan PORV with 316 SS stellited plug and 17-4 PH cage demonstrated the valve opened and closed on demand under the full range of inlet conditions. The opening and closing times were within the 2.0 second opening and closing times normally required for Westinghouse PWRs. The lowest steam flow rate observed in the tests was 232,000 lb/hr. This flow exceeds the rated flow of 210,000 lb/hr for the Comanche Peak, Unit 2, PORVs.

During testing, EPRI induced a bending moment of 43,000 in-lb on the Copes-Vulcan PORV test valve in Test 64-CV-174-2S. Application of this bending moment did not affect test valve performance. The largest moment predicted for the PORV inlet or outlet at Comanche Peak, Unit 2, is

21,625 in-lb. All valve nozzle loads are evaluated for the combined effects of deadweight, thermal expansion, safe shutdown earthquake (SSE), and valve actuation loads. Therefore, the bending moment imposed during valve discharge transients will not affect plant valve performance.

4.3.3 PORV Control Circuit Qualification

NUREG-0737, Item II.D.1, requires the qualification of the PORVs and their associated control circuitry for design basis accidents and transients. The EPRI test program included the PORV control circuitry attached directly to the valve (Reference 16). It did not include the circuits away from the valve such as pressure sensing devices, cables, transmitters, etc. The individual utilities still need to meet the NUREG-0737, Item II.D.1, requirements for the circuits away from the valve. Based on Reference 11, the NRC concluded Applicants could meet the NUREG requirement for environmental qualification of those circuits by including them in their 10 CFR 50.49 program. If an Applicant includes the PORV control circuits in the 10 CFR 50.49 program, specific testing to meet the NUREG-0737 requirements is not necessary. Texas Utilities Electric Company included the PORV controls in the Comanche Peak, Unit 2, environmental qualification program (References 13 and 14). This meets the environmental qualification requirements for the control circuitry. Regarding control circuit qualification for normal operation, TUEC (Reference 14) included the PORV control circuits in its Generic Letter 90-06 (Reference 17) program. The generic letter required Applicants to include the PORVs in the inservice test program. This meets the requirement to qualify the PORV control circuitry during normal operation.

4.3.4 PORV Block Valves

The Westinghouse 3-inch Model 3GM88 block valves used in Comanche Peak, Unit 2, are the same design as one tested by EPRI. Texas Utilities Electric Company modified the block valves/operators as recommended by Westinghouse. The valve/operators now close on limit rather than torque (Reference 14). The plant valve operator will supply greater than 182 ft-lb of torque in this mode of operation. The test valve opened and closed fully under the full range of operating conditions with the operator set to produce 182 ft-lb of torque. Therefore, the tests demonstrated acceptable valve operation.

4.3.5 Operability Summary

The facts presented above demonstrate that TUEC met Item 1 (conducting tests for valve qualification) and Item 7 (considering the affects of discharge piping on operability) of Section 1.2 in this report. Meeting the requirements of 10 CFR 50.49 and including the PORV in the GL 90-06 program satisfy Item 5 of Section 1.2 in this report regarding the PORV control circuitry. However, TUEC should document a formal procedure for the inspection of the safety valves as discussed in Section 4.3.1.

4.4 Piping and Support Evaluation

This evaluation covers the piping and supports from the pressurizer nozzles to the pressurizer relief tank. The Applicant designed the piping for dead weight, internal pressure, thermal expansion, earthquake, and safety and relief valve discharge conditions. This section discusses the calculation of the hydraulic force time histories due to valve discharge, structural analysis methods, and the load combinations and stress evaluation.

4.4.1 Thermal Hydraulic Analysis

Texas Utilities Electric Company used pressurizer fluid conditions in the thermal hydraulic analysis such that the calculated pipe discharge forces bounded the forces for the FSAR, HPI, and cold pressurization events, including the single failure that would maximize the forces on the valve.

The forcing functions from the Comanche Peak, Unit 1, thermal hydraulic analysis were used for Unit 2. Texas Utilities Electric Company justified this approach in References 14 and 15. They stated (Reference 14) the Unit 1 and Unit 2 discharge piping layouts are mirror images of each other within the tolerances allowed by NCIG-05 (Reference 18). In Reference 15, TUEC stated these differences are approximately 6 inches or less. These differences, TUEC stated, are small enough not to affect the hydraulic forcing functions calculated for Unit 1 as applied to Unit 2. Based on this information, EG&G Idaho concluded TUEC's approach is adequate.

In the analysis, TUEC treated the safety valve and PORV discharge transients as two separate events (Reference 13). That is, the safety valves opened simultaneously with the PORVs closed, and the PORVs opened simultaneously with the safety valves closed. This approach is acceptable, because the safety valves and PORVs have different setpoints.

A valve operating condition more likely to occur would be a PORV discharge followed by a safety valve discharge. Because the PORVs have a lower setpoint, they will open first. In this case, the PORV piping loads would be the same as those calculated from the PORV actuation case above. However, this sequence reduces the safety valve discharge forces due to the build-up of backpressure in the discharge piping from the preceding PORV actuation. Therefore, TUEC need not analyze this condition.

Steam discharge transients have the potential to develop the worst loads on the safety valve and PORV piping. Both the safety valves and PORVs at Comanche Peak, Unit 2, have loop seals upstream of the valve inlets. When the safety valve or PORV opens, the loop seal water slug driven by the high steam pressure and flow imposes the highest hydrodynamic loads on the piping and supports.

For the safety valve loop seal, TUEC assumed a temperature of 300°F at the valve inlet. As discussed in Section 4.2.1, TUEC has not measured the Comanche Peak, Unit 2, loop seal temperature to verify the assumed temperature. However, TUEC provided information on the Comanche Peak, Unit 1, loop seal temperature taken by field measurements. The Unit 1 measured loop seal temperature was 314°F. Both Comanche Peak units use the same type of loop seal insulation. Because the Unit 1 measured temperature is greater than 300°F, this verifies the appropriateness of the loop seal temperature used in the Unit 2 thermal hydraulic analysis.

For the PORVs steam discharge also represents the limiting condition for the pipe loads. The PORV inlet piping has a cold loop seal with 150°F water (Reference 13). The thrust of the cold water slug under high steam pressure and flow generates the highest piping loads of all steam and water discharge transients including cold overpressurization events.

In the thermal hydraulic analysis, TUEC selected fluid conditions to bound all limiting transients discussed in Section 4.2. For the safety valve analysis, the initial pressure of the saturated steam upstream of the loop seals was 2575 psia and the initial downstream pressure was 18 psia. Texas Utilities Electric Company held the pressurizer conditions constant for the entire transient at 2575 psia and 1110 Btu/lb. They assumed the loop seal water temperature was 300°F at the safety valve inlet. For the PORV analysis, the initial upstream pressure of the saturated steam was 2350 psia and the downstream pressure was 18 psia. Texas Utilities Electric Company held the pressurizer conditions constant for the entire transient at 2350 psia and 1162 Btu/lb. They assumed the temperature of the liquid upstream of the PORV to be a constant 150°F.

The pressurizer pressure used in the PORV analysis is lower than the maximum pressure of 2532 psia predicted by Westinghouse for a loss of load event. The pressure used in the PORV piping analysis is the valve opening setpoint. They justified the pressure used in References 14 and 15. Texas Utilities Electric Company noted in some cases the pressurizer pressure will continue to rise above the valve setpoint. In the loss of load accident the pressure rises to 2532 psia at a rate of 130 psi/s. Texas Utilities Electric Company also noted that, although the water slug passes through the discharge piping quickly (less than 1.7 s), it does experience some increase in the driving force of the peak pressure. However, TUEC noted the peak pressure, 2532 psia, is less than 10% above the opening pressure. They also noted the loads on the critically loaded portions of the system (valves and pressurizer nozzle) peak within 0.5 s. For piping in the common header region, TUEC stated the forces on the header piping decrease rapidly because the water slug breaks up in the large pipe (inside diameter 12 inches). In the common header region, the stresses due to relief valve discharge are small (less than 1000 psi bending, for example). Therefore, TUEC concluded the pressure increase was not significant and did not include it in the analysis. EG&G Idaho agrees with this conclusion because, during the portion of the valve discharge transient when the critical loadings occur (that is, the first 0.5 s), the pressure would increase by approximately 65 psi. This pressure increase is not considered significant because the loads are dominated by the water slug discharge. After 0.5 s, the forces in the common header region are low enough the pressure increase would have negligible effect.

Texas Utilities Electric Company does not expect the safety valve and PORV piping loads from water discharge to exceed those from steam discharge (Reference 14). This is because of the water slug discharge involved. Based on discussions with Westinghouse, TUEC noted Westinghouse had previously performed analyses of scenarios other than the loop seal/steam discharge case analyzed for Comanche Peak, Unit 2. The results of Westinghouse's analyses indicated the other scenarios were less severe than the loop seal slug discharge. This is consistent with EG&G Idaho's understanding.

The thermal hydraulic analysis used the Westinghouse computer code ITCHVALVE. ITCHVALVE calculates the fluid parameters as a function of time. Another Westinghouse program, FORFUN calculates the unbalanced forces or wave forces in the piping segments from the fluid properties obtained from the ITCHVALVE analysis. These calculations provide the forcing functions on the piping system resulting from the fluid transients.

Westinghouse verified the ITCHVALVE/FORFUN programs for use in valve discharge piping analyses by comparing the analytical and test results for two EPRI tests (Test Nos. 908 and 917). In Reference 13, TUEC presented comparisons of the ITCHVALVE predicted force time histories and the EPRI test results. EG&G Idaho considers these comparisons satisfactory.

Westinghouse, TUEC's consultant, performed the thermal hydraulic analysis of the Comanche Peak, Unit 2, safety valve and PORV piping and supports. EG&G Idaho reviewed a typical Westinghouse analysis for such piping systems in previous submittals for a similar PWR plant (Reference 19). EG&G Idaho reviewed Westinghouse's methods including analysis assumptions and key computer input parameters (node spacing, time steps, valve opening time, etc.). We found these inputs adequate. In addition, TUEC stated in References 13 and 14, the Comanche Peak, Unit 2, piping analysis followed the same approach used in the Westinghouse verification analyses of the EPRI tests for time step, nodalization, and valve opening time. Therefore, EG&G Idaho considers the Comanche Peak, Unit 2, analysis adequate.

The valve opening times TUEC used were 0.040 s for the safety valves and 1.0 s for the PORVs. During testing, EPRI measured opening times for the safety valve and PORV that were faster than the valve opening times TUEC used.

The opening times measured by EPRI were less than 0.019 s for the safety valve and 0.66 s for the PORV. EG&G Idaho does not consider this difference significant because of the good comparisons to test data in the Westinghouse benchmark analyses. Also, Comanche Peak, Unit 2, uses loop seals upstream of both the safety valves and PORVs. Therefore, the valve opening time is not as important in determining peak loads as for plants without loop seals.

Texas Utilities Electric Company provided the safety valve and PORV flow rates used in the analysis in Reference 13. For the safety valve analysis, the flow rate was 120% of the rated flow for the Crosby 6M6 safety valves. The conservatism in this flow rate accounts for the 10% derating of the safety valve flow rate required by the ASME Code. This flow rate is also greater than the 111% of rated flow at 3% accumulation measured in the EPRI tests. The PORV flow rate used in the analysis was 139% of the rated flow for the Copes-Vulcan valve. This accounts for 10% ASME derating of the valve flow rate. It also exceeds the maximum flow observed in the EPRI tests, 122% of rated flow.

4.4.2 Stress Analysis

Westinghouse, for TUEC, calculated the structural response of the piping system to the safety valve/PORV discharge transients using normal mode theory. They used the FORFUN calculated fluid force time histories from the thermal hydraulic analysis as the forcing functions on the structural model. Westinghouse used the structural analysis program, WESTDYN, and its subroutines FIXFM3, WESTDYN2, and POSDYN2. Westinghouse used WESTDYN to calculate the piping natural frequencies and normal modes. FIXFM3 calculated the nodal time history displacements, and WESTDYN2 the internal forces and deflections. Westinghouse used POSDYN2 to calculate the maximum forces, moments, and displacements on the piping elements and maximum piping support loads.

The NRC previously reviewed and approved the WESTDYN series of structural programs (Reference 20). Westinghouse further verified these programs for valve discharge piping analysis by comparing calculated results from these programs with EPRI test results (Reference 13).

EG&G Idaho reviewed the important structural analysis parameters of time step size, lumped mass spacing, cutoff frequency, and damping. The step size was 0.001 s. This time step size will adequately analyze frequencies up to 100 Hz. Although TUEC's cutoff frequency was 1000 Hz, the time step size used will limit the analysis to approximately 100 Hz. Damping of 2% was used for the WESTDYN analysis of the PORV and safety valve discharge piping. EG&G Idaho considers this damping factor adequate based on Reference 21. Reference 21 indicated damping factors of 2% are more realistic. It also indicated using realistic damping factors, rather than small, overly conservative damping factors, could improve overall piping/support system performance. Texas Utilities Electric Company used the PAGES computer program to develop the mass point spacing. This program bases the mass point spacing on the support locations and the pipe size at Comanche Peak, Unit 2. In Reference 15, TUEC stated this program was also used to develop the mass point spacing in the benchmarks of the EPRI tests. These benchmark results were adequate when compared to the test data. Based on the above, EG&G Idaho considers the structural analysis parameters adequate for use in the Comanche Peak, Unit 2, analysis.

The governing code for the piping stress analysis was the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, 1977 Edition, with addenda to and including Summer 1979. For the piping supports, the governing code was the ASME Boiler and Pressure Vessel Code Section III, Subsection NF, 1974 Edition, with addenda to and including Winter 1979. The load combinations and stress limits used to evaluate the piping and support stresses are the same as those recommended by EPRI (Reference 22).

The piping stress summaries presented by the Applicant (Reference 13) compare the highest stresses in the piping with the applicable stress limits. EG&G Idaho reviewed the piping stress results and found all the stresses within the applicable stress limits.

During EPRI tests on the Crosby 6M6 safety valve, high frequency pressure oscillations of 170-260 Hz occurred in the piping upstream of the safety valve as the loop seal water slug passed through the valve. This raised a concern that these oscillations could potentially excite high frequency vibration modes in the inlet piping that could contribute to higher bending moments in the piping. The Applicant did not account for this phenomenon in the structural analysis of the system. However, the piping between the pressurizer and safety valves in the EPRI tests was 8-in. Schedule 160 and 6-in. Schedule XX. The same piping at Comanche Peak, Unit 2, is 6-in. Schedule 160. Because the test piping did not sustain any discernible damage during pressure oscillations occurring in the tests, EG&G Idaho concluded the plant piping also would not incur damage during similar oscillations. Thus, a specific analysis for these pressure oscillations is not necessary for this plant.

Reference 13 presented the worst case load/stress versus the allowables for representative piping supports. The results showed that the load/stresses were within their respective allowables.

In References 14 and 15, TUEC provided information on the pressurizer nozzle loads. They reviewed the nozzle loads due to valve discharge and found they were acceptable for all load conditions identified in the Comanche Peak, Unit 2, Class 1 stress analysis summary report.

4.4.3 Structural Analysis Summary

The selection of a bounding case for the piping evaluation and the piping and support stress analysis demonstrate TUEC met the requirements of Items 3 and 8 of Section 1.2 in this report.

5. EVALUATION SUMMARY

The Applicant for Comanche Peak, Unit 2, provided an acceptable response to the requirements of NUREG-0737, Item II.D.1. Therefore, the Applicant reconfirmed Comanche Peak, Unit 2, met General Design Criteria 14, 15, and 30 of Appendix A to 10 CFR 50 with regard to the safety valves, PORVs, and block valves. The discussion below provides the rationale for this conclusion.

The Applicant participated in the development and execution of an acceptable test program. The program would qualify the operability of prototypical valves and demonstrate that their operation would not invalidate the integrity of the associated equipment and piping. The Electric Power Research Institute successfully completed the subsequent tests under operating conditions which by analysis bounded the most probable maximum forces expected from anticipated operational occurrences and design basis events. The generic test results and piping analyses showed the valves tested functioned correctly and safely for all steam and water discharge events in the test program applicable to Comanche Peak, Unit 2. Also, the pressure boundary component design criteria were not exceeded. Analysis and review of the test results and the Applicant's justifications indicated direct applicability of the prototypical valve and valve performance to the in-plant valves and systems covered by the generic test program. The Applicant's analysis of the plant specific piping showed it was acceptable.

As discussed in Section 4.3.1 of this report, inspection and maintenance are important to the continued operability of the plant safety valves. The Applicant should develop a formal procedure to require safety valve inspection after each actuation and include it in the plant operating procedures or licensing documents.

Thus, TUEC met the requirements of Item II.D.1 of NUREG-0737 (Items 1-8 of Section 1.2 in this report). Therefore, the Applicant demonstrated by testing and analysis for the subject equipment that: (a) the reactor primary coolant pressure boundary will have a low probability of abnormal leakage (General Design Criterion No. 14), (b) the reactor primary coolant pressure boundary and its associated components (piping, valves, and supports) were designed with sufficient margin such that design conditions are not exceeded during relief/safety valve events (General Design Criterion No. 15), and (c) the valves and associated components were constructed in accordance with high quality standards (General Design Criterion No. 30).

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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

In the past, safety and relief valves installed in the primary coolant system of light water reactors have performed improperly. 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 development and completion of programs to do two things. First, the programs should reevaluate the functional performance capabilities of pressurized water reactor safety, relief, and block valves. Second, they should verify the integrity of the pressurizer safety and relief valve piping systems for normal, transient, and accident conditions. This report documents the review of those programs by EG&G Idaho, Inc. Specifically, this report documents the review of the Comanche Peak, Unit 2, Applicant response to the requirements of NUREG-0578 and NUREG-0737. This review found the Applicant provided an acceptable response reconfirming they met General Design Criteria 14, 15, and 30 of Appendix A to 10 CFR 50 for the subject equipment.

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