Idaho National Engineering Laboratory

Managed by the U.S. Department of Energy

EG&G Idaho

Work performed under DOE Contract No. DE-AC07-76ID01570

EGG-NTA-7701
May 1987

INFORMAL REPORT

TECHNICAL EVALUATION REPORT

TMI ACTION--NUREG-0737 (II.D.1)
COMANCHE PEAK STEAM ELECTRIC STATION, UNIT 1

N. E. Pace
C. Y. Yuan
C. L. Nalezny

Prepared for the

U.S. NUCLEAR REGULATORY COMMISSION

EGG-NTA-7701

TECHNICAL EVALUATION REPORT
TMI ACTION--NUREG-0737 (II.D.1)
COMANCHE PEAK STEAM ELECTRIC STATION, UNIT 1
DOCKET NO. 50-445

N. E. Pace
C. Y. Yuan
C. L. Nalezny

May 1987

Idaho National Engineering Laboratory
EG&G Idaho, Inc.
Idaho Falls, Idaho 83415

Prepared for the
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
Under DOE Contract No. DE-AC07-76ID01570
FIN No. D6005

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 review examined the response of the Licensee for the Comanche Peak Steam Electric Station, Unit 1, to the requirements of NUREG-0578 and NUREG-0737 and finds that the Licensee has provided an acceptable response, reconfirming that the General Design Criteria 14, 15, and 30 of Appendix A to 10 CFR 50 have been met.

FIN No. D6005--Evaluation of CW Licensing Actions-NUREG-0737, II.D.1

CONTENTS

TABLES

TECHNICAL EVALUATION REPORT

TMI ACTION--NUREG-0737 (II.D.1)

COMANCHE PEAK STEAM ELECTRIC STATION, UNIT 1

DOCKET NO. 50-445

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 a 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 (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 transient events, and (c) 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 (Reference 2), 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 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 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 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 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 OWNERS' 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 power operated relief valves, safety valves, block valves, and associated piping systems. The Texas Utilities Generating Co. (TUGC), owner of the Comanche Peak Steam Electric Station (CPSES), Unit 1, was one of the utilities sponsoring the EPRI Valve Test Program. The results of the program are contained in a group of reports which 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. Representative valves were selected for testing with a sufficient number of the variable characteristics 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 overpressure protection would be required (Reference 6).

EPRI contracted with Westinghouse Electric Corp. to produce a report on the inlet fluid conditions for pressurizer safety and relief valves in Westinghouse designed plants (Reference 7). Since CPSES, Unit 1, was designed by Westinghouse 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 were tested at the Combustion Engineering Company, Kressinger Development Laboratory, located in Windsor, Connecticut. The results for the relief and safety valve tests are reported in Reference 8. The results for the block valves tests are reported in Reference 9. A guide for applying these test results to the specific plants is presented in Reference 10.

The primary objective of the EPRI/C-E Valve Test Program was to test each of the various types of primary system safety valves used in PWRs for the full range of fluid conditions under which they may be required to operate. The conditions selected for test (based on analysis) were limited 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 that could ultimately be used for verifying analytical piping models.

Transmittal of the test results meets the requirement of Item 6 of Section 1.2 to provide test data to the NRC.

3. PLANT SPECIFIC SUBMITTAL

A preliminary plant specific evaluation of the adequacy of the overpressure protection system for CPSES, Unit 1, was submitted by TUGC to the NRC on July 8, 1981 (Reference 11) and March 31, 1982 (Reference 12). The information provided was reviewed and a request for additional information was sent on July 5, 1985 (Reference 13), to which the licensee responded on June 13, 1986 (Reference 14). On March 27, 1987 (Reference 15) the NRC transmitted a second request for information to TUGC, and the licensee responded on April 15, 1987 (Reference 16).

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

CPSES, Unit 1, is a four-loop PWR designed by the Westinghouse Electric Co. It is equipped with three (3) safety valves, two (2) PORVs, and two (2) PORV block valves in its overpressure protection system. The safety valves are 6-in. Crosby Model HB-BP-86, 6M6, spring loaded valves with loop seal internals. The design set pressure is 2485 psig and the rated steam flow capacity is 420,000 lbm/h. The PORVs are 3-in. Copes-Vulcan Model D-100-160 globe valves with 316 S.S. stellited plugs and 17-4 PH cages. The PORV opening set pressure is 2335 psig and the rated steam flow capacity is 210,000 lbm/h. The inlet pipe to the safety valve and PORVs include loop seals. The PORV block valves are 3-in. Westinghouse Model 3GM88 gate valves with Limitorque SB-00-15 motor operators; the operator gear ratio has been modified and the wiring has been changed for limit control to assure valve operability as recommended by Westinghouse.

Safety valves, PORVs, and PORV block valves identical to those used at CPSES, Unit 1, were included in the EPRI tests. Since there is no difference between the valves tested and the valves installed at the plant, the test results for these valves are directly applicable to CPSES, Unit 1. Therefore, those parts of the criteria of Items 1 and 7 as identified in Section 1.2 of this report regarding applicability of the test valves are fulfilled.

4.2 Test Conditions

As stated above, CPSES, Unit 1, is a four-loop PWR designed by the Westinghouse Electric Corp. The valve inlet fluid conditions that bound the overpressure transients for Westinghouse designed PWR Plants are identified in Reference 7. The transients considered in this report include FSAR, extended high pressure injection, and low temperature overpressurization events. The expected fluid conditions for each of these events and the applicable EPRI tests are discussed in this section.

4.2.1 FSAR Steam Transients

For the CPSES, Unit 1, PWR, the limiting FSAR transients resulting in steam discharge through the safety valves alone and in steam discharge through both the safety and relief valves are the loss of load event (for maximum pressurizer pressure) and the locked rotor event (for the maximum pressurization rate).

In the case when the safety valves actuate alone, the maximum pressurizer pressure and maximum pressurization rate are predicted to be 2555 psia and 144 psi/s, respectively. The maximum developed backpressure in the outlet piping is 500 psig (Reference 15). The loop seal is insulated such that the valve inlet temperature is 300°F (Reference 15).

EPRI tests representative of the valve inlet fluid conditions for the limiting transient were selected for the plant specific evaluation. In selecting the EPRI tests, the safety valve ring settings and the pressure drop through the inlet pipe were also considered. For steam flow conditions, four loop seal discharge tests (Test No. 929, 1406, 1415, 1419) were applicable to CPSES, Unit 1. These tests were performed with valve ring settings representative of the typical ring settings used in Westinghouse PWRs including CPSES. 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. The relative ring settings used at CPSES Unit 1 are -82 to -103, and -18 relative to the level position. Since both the test ring settings and the in-plant ring settings were determined by the valve manufacturer, the Crosby Valve and Gage Co., using the same methods and the same standard of performance, these ring settings are considered comparable.

Both the inlet pipe length and the water seal volume in the EPRI tests were greater than those employed at the plant. Therefore, the pressure drop through the inlet pipe would be higher in the tests than that at CPSES, Unit 1. The Licensee has provided calculated values for the inlet pressure drop on valve opening and closing which are comparable to the test values. The loop seal temperature measured in the tests ranged from 90 to 360°F at the valve inlet. The maximum pressurizer (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, which is summarized in Table 4.2.1, show that the inlet fluid conditions and backpressures of these tests envelop the corresponding fluid data predicted for the CPSES, Unit 1, safety valves.

When both the safety valves and PORVs are actuated, the maximum pressurizer pressure is predicted to be 2532 psia and the maximum pressurization rate is 130 psi/s. In the EPRI tests on the Copes-Vulcan PORV, the maximum steam pressure at valve opening was 2715 psia, which bounds the predicted pressure at CPSES, Unit 1. The backpressure developed at the outlet of the PORVs is not an important consideration, since the air operated PORVs used at the CPSES, Unit 1, plant are not sensitive to backpressure (Reference 6). Therefore the EPRI test inlet fluid conditions for the PORV in steam discharge are representative of 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 7). From a review of feedwater line break analysis for CPSES, Unit 1, it is clear that the feedwater line break is most likely to be the limiting transient for providing high pressure liquid to the safety valves, a fluid for which they were not originally designed. Therefore, in accordance with the NUREG requirements, the safety valves and PORVs should be qualified for inlet conditions typical of the feedline break event. Therefore, the valve operability will be reviewed using the feedline break data provided in Reference 7.

Reference 7 showed that, in a feedline break accident at CPSES, Unit 1, the maximum pressure at the safety valve inlet during liquid discharge was

TABLE 4.2.1 SUMMARY OF TEST DATA FOR CROSBY 6M6 SAFETY VALVE AND COMPARISON WITH COMANCHE PEAK 1 REQUIREMENTS

Valve	Test Number	Test Type	Inlet Conditions	Initial Fluid Temperature at Valve Inlet (°F)	Safety Valve Ring Settings[1]	Pressure at Valve Opening[2] (psia)	Peak Tank Pressure (psia)	Peak Back-pressure (psia)	Percent Blowdown	Pres-surization Rate (psi/s)	Valve Stability	Inlet Pressure Drop (psi)
6M6-Plant Valve	FSAR Steam Trans-ient	---	Hot loop seal	300	- 82, -18 -102, -18	2500	2555	500	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 -102, -18	2500	2503	500	Nominally 5.0	5.0	---	269.0
6M6-Loop seal internals	931a[5]	LS/ Trans	Cold loop	117	-77, -18	2570	2578	725	12.7	2.5	Stable	263.0
	931b	Water	Hot loop	635	-77, -18	2475	2475	700	4.8	2.5	Chatter[6]	263.0

1. The plant ring settings are relative to the highest locked position, while the test ring settings are relative to the level position. The test and plant ring settings are equivalent.

2. The set pressure of the test valves was 2485 psia.

3. This test was terminated because of valve chatter.

4. The maximum liquid surge rate during a feedwater line break is 2989 gpm.

5. The maximum liquid flow rate during test 931a was 2355 gpm.

6. The valve chattered during opening but then stablized.

calculated to be 2503 psia, the pressurization rate was 5 psi/s and the maximum pressurizer surge rate was 1109.5 gpm (~369,000 lbm/hr) liquid at 608-615°F.

In a feedline break accident resulting in safety valve actuation, water discharge is always preceded by steam and steam to water transition flows. Among the EPRI tests performed on the 6M6 valve, Tests 931a and 931b were performed for loop seal steam, steam to water transition, and water discharge conditions. The valve ring settings and inlet pipe configuration used in these tests were comparable to those 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 fluid temperature was 117°F and the tank fluid temperature was 635°F. After the valve closed in Test 931a, the system was allowed to repressurize and the valve cycled on approximately 640°F water for Test 931b. Since the inlet temperature and pressure of the tests bounded the predicted in-plant condition, the results of these tests are considered representative of the CPSES, Unit 1, safety valves. The inlet fluid conditions and corresponding test data for liquid discharge are also summarized in Table 4.2.1.

The expected fluid conditions at the inlet of the safety valve were based on a Westinghouse analysis that assumed the PORVs were not operable during the feedline break transient. If the PORVs are operable, the same fluid conditions postulated for the safety valve inlet can also be expected at the PORV inlet (Reference 6). In the EPRI tests, high temperature water discharge and steam to water transition tests were performed with the Copes-Vulcan PORV. In the water discharge test, Test No. 76-CV-316-2W, the maximum pressure at the valve inlet 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 compare well with the predicted maximum pressure and temperature of 2505 psia and 615°F for the CPSES, Unit 1, plant. Therefore this test is 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 7). For a four-loop plant, both the safety valves and PORVs will be challenged. Both steam and water discharge are expected. For safety valve actuation, the maximum pressure at valve inlet is predicted to be 2507 psia and the pressurization rate is within 4 psi/s. The inlet temperature ranges from 567 to 572°F. For the PORV, the maximum pressure is predicted to be 2353 psia and the pressurization rate is within 4 psi/s and the inlet temperature ranges from 565 to 569°F. In this event, however, the safety valves or PORVs open on steam and liquid discharge would not be observed until the pressurizer becomes water solid. According to Reference 7, this would not occur until at least 20 minutes into the event which allows ample time for operator action. Thus the potential for liquid discharge in extended HPI events can be disregarded.

4.2.4 Low Temperature Overpressurization Transient

The PORV is used for overpressure protection during the low temperature stages in reactor startup and shutdown operations. The low pressure set point of the PORVs vary with valve and temperature and range from 445 psig to 2350 psig (Reference 14). The expected inlet fluid conditions for low temperature overpressurization transients are identified in Reference 7 and 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 low temperature overpressurization transients. For water discharge conditions, there were two low pressure and low temperature water tests performed on the Copes-Vulcan PORV with stellited plug and 17-4 PH cage. The tests were conducted at an inlet pressure of 675 psia and water temperatures of 105°F and 442°F, respectively. These conditions are representative of those at CPSES, Unit 1. Therefore, the EPRI tests can be used to evaluate the performance of the CPSES, Unit 1, PORV for low temperature overpressurization transients.

4.2.5 PORV Block Valve Fluid Conditions

The block valves used are 3-inch Westinghouse Model 3GM88 gate valves with Limitorque SB-00-15 operators. These valves were tested for full flow and pressure conditions in the EPRI tests and, as modified, qualify for operation at these conditions.

The test sequences and analyses described above demonstrate that the test conditions bound the conditions for the plant values. They also verify that Items 2 and 4 of Section 1.2 have been met, in 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.

4.3 Operability

4.3.1 Safety Valves

The EPRI tests representative of the steam discharge condition for the CPSES, Unit 1, safety valves are the loop seal tests on the Crosby 6M6 valve, Test No. 929, 1406, 1415, 1419. In all 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. Up to 111% of rated flow was achieved at 3% accumulation with valve lift positions at 92 to 94% of rated lift. These tests demonstrated that the valve performed its function in spite of the initial chatter during loop seal discharge.

In Test 1419, the valve chattered on closing and the test was terminated after the valve was manually opened to stop the chatter. This result does not indicate a valve closing problem for the CPSES, Unit 1, safety valve since an identical test (Test 1415) had already demonstrated that the valve performed satisfactorily and exhibited no sign of instability. The closing chatter in Test 1419 may possibly be a result of the repeated actuation of the valve in loop seal and water discharge tests. As shown in Table 4.3.1 the 6M6 test valve was subjected to seventeen steam, water, and transition tests. 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 test was terminated due to chattering on closing. Galled guiding surfaces and damaged internal parts were found during inspection and the damaged parts were refurbished or replaced before the next test started. The test results showed that the valve performed well after each repair, but the closing chatter recurred in the subsequent test. Test 1415 was performed immediately after valve maintenance and the valve performed stably. The next test (Test 1419) encountered chatter in 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 Licensee should develop a formal procedure requiring that the safety valves be inspected after each actuation and the procedure should be incorporated into the plant operating procedures or licensing documents such as the plant technical specifications.

The blowdown in these tests (5.1 to 9.4%) were in excess of the 5% value specified by the valve manufacturer and the ASME Code. Westinghouse performed an analysis, "Safety Valve Contingency Analysis in Support of the EPRI Safety/Relief Valve Testing Program--Volume 3: Westinghouse Systems," EPRI NP-2047-LD, October 1981, on the effects of increased blowdown and concluded that no adverse effects on plant safety occurred in that the reactor core remained covered. Therefore, the amount of increased blowdown which occurred in the Crosby 6M6 steam tests is considered acceptable.

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 with typical plant ring settings of (-71, -18) simulate the expected CPSES, Unit 1, feedwater line break conditions. Test 931a was a loop seal/steam/water transition test. The 6M6 valve initially opened, fluttered or chattered in a partial lift position during loop seal discharge, then

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

Seqn No.	Test No.	Ring Setting	Test Type		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		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	
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	1416	4	L.S.		Terminated	0	1.5
				Inspection/Repair			

c--chatter

f/c--flutter/chatter

L.S.--loop seal

Ring setting--four different sets of ring settings were tested. Actual ring positions not shown.

Terminated--Test terminated after valve was manually opened to stop chatter.

popped open, stabilized on steam, and closed with a 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 CPSES, Unit 1, during the feedline break transient is 1109.5 gpm. The 6M6 valve tested by EPRI passed 2355 gpm at 2415 psia and 641°F which is much higher than the predicted flow rate for CPSES, Unit 1. The above results demonstrate that the Crosby 6M6 safety valves would be adequate to perform the required water relief function.

The loads induced on the safety valves tested by EPRI exceed the loads for the CPSES, Unit 1, safety valves. The maximum moment tested for the 6M6 valve was during test 908 and was 298,750 in.-lb. The largest moment predicted for the safety valve inlet at CPSES, Unit 1, is 178,175 in.-lb. This demonstrates functionability for the CPSES, Unit 1 safety valves.

4.3.2 Power Operated Relief Valves

The EPRI tests on the Copes-Vulcan PORV with 316 S.S. stellited plug and 17-4 PH cage demonstrated that the valve opened and closed on demand in steam, water, and steam to water transition conditions. The opening and closing time were within the 2.0 second opening and closing time normally required for Westinghouse PWRs. The lowest steam flow rate observed in the tests was 232,000 lb/h which exceeded the rated flow of 210,000 lb/h for the CPSES, Unit 1, PORVs.

A bending moment of 43,000 in.-lb was induced in the inlet of the Copes-Vulcan PORV test valve per EPRI 64-CV-174-2S. The largest moment predicted for the PORV inlet in the CPSES, Unit 1, valves is 28,708 in.-lb. This demonstrates functionability of the CPSES, Unit 1, relief valves.

4.3.3 Electric Control Circuitry

NUREG-0737 II.D.1 requires qualification of associated control circuitry as part of the safety/relief valve qualification. The Licensee addressed (Reference 14) the qualification of the PORV control circuitry required by NUREG-0737, Item II.D.1. The information contained a list of equipment and the abnormal events for which they have been qualified. This submittal included the tests required by NUREG-0737, Item II.D.1, thereby assuring the functionality of the control circuitry.

4.3.4 PORV Block Valves

The Westinghouse 3-inch Model 3GM88 block valves that are used in CPSES, Unit 1, are the same as those finally tested by EPRI (modified the same as the test valves) and the test valves opened and closed fully under the full range of operating conditions. Therefore, the plant valves are expected to function acceptably when required to do so.

The above discussion, demonstrating that the valves operated satisfactorily, verifies that the part of Item 1 of Section 1.2 which requires conducting tests to qualify the valves and that part of Item 7 which requires the effect of discharge piping on operability be considered have been met provided that the Licensee documents formal procedure for the inspection of safety valves as discussed in Section 4.3.1. Also, the qualification testing of the control circuitry satisfies the requirements of Item 5 of Section 1.2.

4.4 Piping and Support Evaluation

This evaluation covers the piping and supports upstream and downstream of the safety valves and PORVs extending from the pressurizer nozzles to the pressurizer relief tank. The piping was designed for dead weight, internal pressure, thermal expansion, earthquake, and safety and relief valve discharge conditions. The calculation of the time histories of hydraulic forces due to valve discharge, the method of structural analysis, and the load combinations and stress evaluation are discussed below.

4.4.1 Thermal Hydraulic Analysis

Pressurizer fluid conditions were selected for use in the thermal hydraulic analysis such that the calculated pipe discharge forces would bound the forces for any of the FSAR, HPI, and cold pressurization events, including the single failure that would maximize the forces on the valve.

In the analysis, the safety valve and PORV discharge transients were treated as two separate events, that is, the safety valves were assumed to actuate simultaneously with the PORVs closed, and the PORVs were assumed to actuate simultaneously with the safety valves closed. This approach is acceptable, since the safety valves and PORVs have different set points; they will not lift at the same time. A valve operating condition which is more likely to occur would be a discharge of the PORVs followed by discharge of the safety valves at their respective set points. Since the PORVs have a lower set point, they will open ahead of the safety valves. In this case, the PORV piping loads would be the same as those calculated from the simultaneous PORV actuation case above, but the safety valve discharge forces would be reduced due to the build-up of backpressure in the discharge piping resulting from the preceding PORV actuation. Therefore, this condition needs not to be analyzed.

The steam discharge transients are potentially the worst loading conditions for the safety valve and PORV piping. Both the safety valves and PORVs at Comanche Peak 1 have loop seals upstream of the valve inlets. When the safety valve or PORV actuates, the water slug in the loop seal driven by the high steam pressure imposes the highest hydrodynamic loads on the piping and supports. The piping loads due to water discharge are not expected to exceed the steam discharge conditions. The loop seal temperature is 300°F at the safety valve inlet and 650°F at the steam/water interface at the upstream end of the loop seal (Reference 17). The average temperature of the loop seal water would be in the range of 350°F to 400°F. The limiting event for water discharge through the safety valves is the feedline break accident. The water temperature was predicted to be 608°F to 615°F (Reference 7). This temperature is considerably higher than the loop seal temperature. Therefore, more flashing is expected in the high temperature water discharge case and the hydrodynamic forces on the piping would be less severe than the water slug discharge condition.

For the PORVs, the 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 14). The thrust of the cold water slug under high steam pressure generates the highest piping loads of all steam and water discharge transients including the cold overpressurization events. Therefore, the valve discharge conditions selected for the piping and support stress evaluations were the steam discharge conditions resulting from the simultaneous actuation of the safety valves and the simultaneous actuation of the PORVs respectively.

In the thermal hydraulic analysis, fluid conditions were assumed 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 assumed to be 2575 psia and the initial downstream pressure was assumed to be 18 psia. The pressurizer conditions were held constant for the entire transient at 2575 psia and 1100 Btu/lb. The loop seal water temperature was assumed to be 300°F at the safety valve inlet. For the PORV analysis, the initial upstream pressure of the saturated steam was assumed to be 2350 psia and the downstream pressure was assumed to be 18 psia. The pressurizer conditions were held constant for the entire transient at 2350 psia and 1162 Btu/lb. The temperature of the liquid upstream of the PORV was assumed to be a constant 150°F. The pressurizer pressure used in the analysis is lower than the maximum pressure of 2532 psia predicted by Westinghouse. This slight nonconservatism is more than compensated by assuming a constant loop seal temperature of 150°F, since the loop seal water at the steam-water interface would be at saturation and the temperature along the loop seal would be hotter than at the PORV inlet. The effect on calculated discharge loads of slightly lower upstream pressure is less than the the effect of lower loop seal temperature.

The thermal hydraulic analysis was performed using the Westinghouse computer code, ITCHVALVE. ITCHVALVE calculates the fluid parameters as a function of time. The unbalanced forces or wave forces in the piping segments are calculated from the fluid properties obtained from the ITCHVALVE analysis using another Westinghouse program, FORFUN. The forcing functions on the piping system resulting from the fluid transients are obtained from these calculations.

The adequacy of the ITCHVALVE/FORFUN programs for the thermal-hydraulic analyses was verified by comparing the analytical and test results for thermal hydraulic loadings in safety valve discharge piping for two EPRI tests (Test Nos. 908 and 917). The detailed comparisons of the ITCHVALVE predicted force time-histories and the EPRI test results are presented in the submittal (Reference 3) and results of these comparisons are considered satisfactory.

The thermal hydraulic and stress analysis of the Comanche Peak 1 safety valve and PORV piping and supports were performed by the Westinghouse Electric Co. as a consultant to the Licensee. The typical Westinghouse analysis for such piping systems has been fully reviewed in previous submittals for similar PWR plants such as the Diablo Canyon Units 1 and 2 (References 18 and 19). The method of analysis used by Westinghouse including the analysis assumptions, the structural modeling as well as the key parameters used in the computer inputs such as the node spacing, calculation time interval, valve opening time, etc. has been examined and found to be acceptable. The Comanche Peak 1 piping analysis followed the same method and procedure used in previous Westinghouse analyses. Therefore the Comanche Peak 1 analysis method is considered acceptable. The flow rate of the safety valve used in the analysis was 120% of the rated flow for the Crosby, 6M6, safety valves. The conservative factor contained in this flow rate is greater than needed to account for the 10% derating for the safety valve required by ASME Code and the allowance for uncertainties or errors. The PORV flow rate used in the analysis was 139% of the rated flow for the Copes-Vulcan valve, which again is amply conservative.

4.4.2 Stress Analysis

The structural responses of the piping system due to the safety valve/PORV discharge transients were calculated using the modal superposition method. The fluid force time histories generated from the FORFUN program in the thermal hydraulic analysis were used as forcing functions on the structural model. The Westinghouse series of structural analysis programs, namely WESTDYN7, FIXFM3 and WESTDYN2 were used to calculate the piping natural frequencies and mode shapes, the nodal displacements and the internal forces and support reactions. The FIXFM3 code calculates the displacements at the structural node points, using the forcing functions generated by FORFUN and the modal data from WESTDYN7. The structural displacements were then used by WESTDYN2 to compute the piping internal loads and support loads.

The WESTDYN series of structural programs mentioned above was previously reviewed and approved by the NRC (Reference 20). The adequacy of these programs for piping discharge analysis was further verified by comparing the solutions generated by these programs with the EPRI safety valve test results (Reference 21).

The piping stress analysis was performed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, 1977 Edition, with addenda to and including Summer 1979. The piping supports were analyzed in accordance with the ASME Boiler and Pressure Vessel Code Section III, Subsection NF, 1974 Edition, with addenda to and including Winter 1979. The load combination equations and stress limits used for the evaluation of the piping and support stresses are identical to those recommended by the piping subcommittee of the PWR Pressurizer safety and relief valve test program (Reference 10). The piping stress summaries presented by the Licensee (Reference 16) contain a comparison of the highest stresses in the piping with the applicable stress limits for the load combinations defined above. The piping stress results were reviewed and all the stresses were found to be within the applicable stress limits.

According to results of EPRI tests performed on the Crosby 6M6 safety valve, high frequency pressure oscillations of 170-260 Hz occurred in the piping upstream of the safety valve as a loop seal water slug passed through the valve. This raises 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. This phenomenon was not accounted for in the structural analysis of the system. The piping between the pressurizer and safety valves in the EPRI tests, however, was composed of 8-in. Schedule 160 and 6-in. Schedule XX while that at Comanche Peak 1, is 6-in. Schedule 160. Since the test piping did not sustain any discernible damage during pressure oscillations occurring in the tests, it is expected that 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 16 presented the worst case load/stress versus the allowables for representative piping supports. The results all showed that the load/stresses were within their respective allowables.

The selection of a bounding case of the piping evaluation and the piping and support stress analysis demonstrate that the requirements of Item 3 and Item 8 of Section 1.2 outlined in this report have been met.

5. EVALUATION SUMMARY

The Licensee for CPSES, Unit 1, has provided an acceptable response to the requirements of NUREG-0737, and thereby reconfirmed that the General Design Criteria 14, 15, and 30 of Appendix A to 10 CFR 50 have been met. 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 designed 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 bounded the most probable maximum forces expected from anticipated design basis events. The generic test results and piping analyses showed that the valves tested functioned correctly and safely for all relevant steam discharge events specified in the test program and that the pressure boundary component design criteria were not exceeded. Analysis and review of the test results and the Licensee's justifications indicated direct applicability of the prototypical valve and valve performances to the in-plant valves and systems intended to be covered by the generic test program.

Thus, the requirements of Item II.D.1 of NUREG-0737 have been met (Items 1-8 in Paragraph 1.2) and, thereby demonstrate by testing and analysis, that the reactor primary coolant pressure boundary will have a low probability of abnormal leakage (General Design Criterion No. 14) and that the reactor primary coolant pressure boundary and its associated components (piping, valves, and supports) have been designed with sufficient margin such that design conditions are not exceeded during relief/safety valve events (General Design Criterion No. 15).

Furthermore, the prototypical tests and the successful performance of the valves and associated components demonstrated that this equipment has been constructed in accordance with high quality standards (General Design Criterion No. 30).

REFERENCES

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

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

3. D. P. Hoffman, Consumers Power Co., letter to H. Denton, NRC, "Transmittal of PWR Safety and Relief Valve Test Program Reports," September 30, 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, January 1983.

6. EPRI PWR Safety and Relief Valve Test Program Test Condition Justification Report, EPRI NP-2460, January 1983.

7. Valve Inlet Fluid Conditions for Pressurizer Safety and Relief Valves in Westinghouse-Designed Plants, EPRI NP-2296, January 1983.

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

9. R. C. Youngdahl, Consumers Power Co., letter to H. Denton, NRC, "Submittal of PWR Valve Data Package," June 1, 1982.

10. 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.

11. P. Knight, Texas Utilities Services, Inc., memorandum for R. L. Tedesco, Assistant Director of Licensing, NRC, "Input to Comanche Peak SER," July 8, 1981.

12. H. C. Schmidt, Texas Utilities Services, Inc., letter to S. B. Burwell, Licensing Project Manager, NRC, "Comanche Peak Steam Electric Station Draft Response to Generic Letter No. 81-36," Log # TXX--3503, File # 10035, March 31, 1982.

13. V. S. Noonan, NRC, letter to M. D. Spence, Texas Utilities Generating Co., "Performance Testing of Relief and Safety Valves," July 5, 1985.

14. W. G. Counsil, Texas Utilities Generating Co., letter to V. S. Noonan, Director, Comanche Peak Project, DL, NRC, "Comanche Peak Steam Electric Station (CPSES), Docket Nos. 50-445 and 50-446, NUREG-0737, Item II.D.1--Performance Testing of Relief and Safety Valves," Log # TXX-4849, File # 10010, June 13, 1986.

15. C. Grimes, NRC, letter to W. G. Counsil, Texas Utilities Generating Co., "Performance Testing of Relief and Safety Valves," March 27, 1987.

16. W. G. Counsil, Texas Utilities Generating Co., letter to U. S. Nuclear Regulatory Commission, "COMANCHE PEAK STEAM ELECTRIC STATION (CPSES), DOCKET NOS. 50-445 and 50-446, REQUEST FOR ADDITIONAL INFORMATION - NUREG-0737, ITEM II.D.1--PERFROMANCE TESTING OF RELIEF AND SAFETY VALVES," Log # TXX-6398, File # 10010, April 15, 1987.

17. Amendment 40 to Comanche Peak FSAR, "Response to NRC Action Plan II.D.1," May 10, 1983.

18. L. C. Smith and T. L. Nazlett, Pressurizer Safety and Relief Line Evaluation Summary Report, Pacific Gas and Electric Corp., Diablo Canyon Unit 1 and Unit 2, Westinghouse AM-SSA-2534, November 1982.

19. J. O. Schuyler, Pacific Gas and Electric, letter to G. W. Knighton, NRC, "Response to NRC Request for Additional Information, NUREG-0737, Item II.D.1, Performance Testing of Relief and Safety Valves," January 23, 1984.

20. R. L. Tedesco, NRC letter to T. M. Anderson, Westinghouse Electric Co., "Acceptance for Referencing of Licensing Topical Report WCAP-8252, Revision 1," April 7, 1981.

21. L. C. Smith and T. M. Adams, "Comparison of Analytically Determined Structural Solutions with EPRI Safety Valve Test Results," 4th National Congress on Pressure Vessel and Piping Technology, Portland, Oregon, June 19-24, 1983, PVP-Volume 74, pp. 193-199.

NRC FORM 335
(2-84)
NRCM 1102,
3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE

1. REPORT NUMBER (Assigned by TIDC, add Vol. No., if any)
EGG-NTA-7701

2. TITLE AND SUBTITLE
TMI Action-NUREG-0737 (II.D.1) Comanche Peak Steam Electric Station, Unit 1

3. LEAVE BLANK

4. DATE REPORT COMPLETED
MONTH: May YEAR: 1987

5. AUTHOR(S)
N. E. Pace, C. Y. Yuan, C. L. Nalezny

6. DATE REPORT ISSUED
MONTH: May YEAR: 1987

7. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
INEL-EG&G Idaho, Inc.
Idaho Falls, ID 83415

8. PROJECT/TASK/WORK UNIT NUMBER

9. FIN OR GRANT NUMBER
D6005

10. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Mechanical Engineering Branch
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

11a. TYPE OF REPORT
Informal

b. PERIOD COVERED (Inclusive dates)

12. SUPPLEMENTARY NOTES

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 the functional performance capability 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 review examined the response of the Licensee for the Comanche Peak Steam Electric Station, Unit 1, to the requirements of NUREG-0578 and NUREG-0737 and finds that the Licensee has provided an acceptable response, reconfirming that the General Design Criteria 14, 15, and 30 of Appendix A to 10 CFR 50 have been met.

14. DOCUMENT ANALYSIS - a. KEYWORDS/DESCRIPTORS

b. IDENTIFIERS/OPEN-ENDED TERMS

15. AVAILABILITY STATEMENT
Unlimited

16. SECURITY CLASSIFICATION
(This page) Unclassified
(This report) Unclassified

17. NUMBER OF PAGES

18. PRICE