EGG-NTA-8480

TECHNICAL EVALUATION REPORT
TMI ACTION--NUREG-0737 (II.D.1)
RELIEF AND SAFETY VALVE TESTING
BEAVER VALLEY UNIT 2
DOCKET NO. 50-412

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April 1989

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Prepared for the
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555
Under DOE Contract No. DE-AC07-76ID01570
FIN No. D6005
TAC No. 62894

8908160335 XA -

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 that programs be developed and completed to: (a) reevaluate the functional performance capabilities of Pressurized Water Reactor (PWR) safety, relief, and block valves and (b) verify the integrity of the pressurizer safety and relief valve piping systems for normal, transient, and accident conditions. This report documents the review of these programs by the Nuclear Regulatory Commission (NRC) and their consultant, EG&G Idaho, Inc. Specifically, this report documents the review of the Beaver Valley Unit 2 Licensee response to the requirements of NUREG-0578 and NUREG-0737. This review found the Licensee has not provided an acceptable response; therefore, the Licensee has not reconfirmed that the General Design Criteria 14, 15, and 30 of 10 CFR 50, Appendix A, were met.

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

Summary

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. This failure, plus other previous instances of improver valve performance, led the task force which prepared NUREG-0578 and NUREG-0737 to recommend that programs be developed to reexamine the functional performance capabilities of Pressurized Water Reactor (PWR) safety, relief, and block valves. The task force also recommended the programs verify the integrity of the pressurizer safety and relief valve piping systems for normal, transient, and accident conditions. This was deemed necessary to reconfirm that the General Design Criteria 14, 15, and 30 of 10 CFR 50, Appendix A, have indeed been satisfied.

This report documents the review by EG&G Idaho, Inc., of the Beaver Valley Unit 2 Licensee response to the requirements of NUREG-0578 and NUREG-0737. The Licensee submittals were reviewed to determine the applicability of the test valves and test conditions to the plant valves and inlet conditions. The operability of the test valves was reviewed to determine the operability of the plant valves. The Licensee's analysis of the pressurizer discharge piping was reviewed to determine if acceptable stress limits were met for valve discharge transients.

The Licensee only partially met the requirements of NUREG-0578 and NUREG-0737. The test results showed that the PORV and the PORV block valve functioned satisfactorily for all steam and water discharge events. The test results also indicated the safety valve functioned satisfactorily for all steam discharge events that are applicable to the plant valves. This demonstrated the valves were constructed in accordance with high quality standards, meeting General Design Criterion No. 30. However, the information supplied by the Licensee was not sufficient to determine if:
(a) the safety valves will operate acceptably for a feedwater line break,
(b) the plant safety valves will operate as stably as the test valve,
(c) the plant PORVs will operate with the maximum expected bending moment,
and (d) the analysis of the safety valve discharge piping was adequate.
Because of these findings, it could not be concluded the Licensee had met General Design Criteria 14 and 15 of 10 CFR 50, Appendix A.

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1. INTRODUCTION

1.1 Background

In the past, relief and safety valves installed in the primary coolant systems 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 these past instances of improper valve performance, it is not known whether they occurred because of a limited qualification of the valve or because of basic unreliability of the valve design. It is known that the failure of a 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 to: (a) reexamine the functional performance capabilities of Pressurized Water Reactor (PWR) safety, relief, and block valves and (b) verify the integrity of the pressurizer safety and relief valve 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 10 CFR 50, Appendix A, were indeed satisfied.

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 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, which was issued for implementation on October 31, 1980. As stated in the NUREG reports, each pressurized water reactor Licensee or Applicant shall:

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

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

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

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

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

6. Provide test data for 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 OWNER'S GROUP RELIEF AND SAFETY VALVE PROGRAM

In response to the NUREG requirements previously listed, a group of utilities with PWRs requested the assistance of the Electric Power Research Institute (EPRI) in developing and implementing a generic test program for pressurizer safety valves, PORVs, block valves, and associated piping systems. Duquesne Light Co., the owner of Beaver Valley Unit 2, was one of the utilities sponsoring the EPRI Valve Test Program. The results of the program, which are contained in a series of reports, were transmitted to the NRC by Reference 3. The applicability of these reports is discussed below.

Electric Power Research Institute developed a plan (Reference 4) for testing PWR safety, relief, and block valves under conditions which bound actual plant operating conditions. Electric Power Research Institute, through the valve manufacturers, identified the valves used in the overpressure protection systems of the participating utilities and representative valves were selected for testing. These valves included a sufficient number of the variable characteristics so that their testing would adequately demonstrate the performance of the valves used by utilities (Reference 5). Electric Power Research Institute, 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).

Electric Power Research Institute contracted with Westinghouse Corporation to produce a report on the inlet fluid conditions for pressurizer safety and relief valves in Westinghouse designed plants (Reference 7). Since Beaver Valley Unit 2 was designed by Westinghouse, this report is relevant to this evaluation.

Several test series were sponsored by EPRI. Power operated relief valves 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 Combustion Engineering (CE) Company's Kressinger Development Laboratory located in Windsor, Connecticut. The results of the relief and safety valve tests are reported in Reference 8. The results of the block valve tests are reported in Reference 9.

The primary objective of the EPRI/CE 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 Item 6 (provide test data to the NRC) of Section 1.2 in this report.

3. PLANT SPECIFIC SUBMITTAL

An assessment of the adequacy of the overpressure protection system was submitted by Duquesne Light on April 10, 1987 (Reference 10). A request for additional information (Reference 11) was submitted to Duquesne Light Co. by the NRC on October 14, 1988. Duquesne Light responded to this request on January 27, 1989 (Reference 12).

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

Beaver Valley Unit 2 utilizes three safety valves and three PORVs. In addition, three PORV block valves are installed at the plant. The safety valves are 6 in. Crosby Model HB-BP-86, $6M_16$, spring loaded valves with loop seal internals. The PORVs are 3 in. by 6 in. Garrett solenoid operated pilot valves. Both the safety and relief valves have hot water seals upstream of the valves. The PORV block valves at Beaver Valley Unit 2 are Westinghouse Model 3GM88 gate valves with Limitorque SB-00-15 operators.

The Beaver Valley Unit 2 plant-specific safety valves are Crosby $6M_16$ valves with loop seal internals. This valve was not tested in the test program. However, a Crosby Model 6M6 valve with loop seal internals was tested in the EPRI test program. This valve is of the same basic design but differs in orifice size and flow capacity. The difference in orifice size only affects the valve capacity but not the valve behavior. Other differences, such as body construction (cast or forged), disk holder type, and disk materials, do not have a significant effect on valve operability. The valve was tested on a long inlet piping/loop seal configuration which represents the Beaver Valley Unit 2 plant installation. The results from the EPRI tests can, therefore, be used to demonstrate operability of the Beaver Valley Unit 2 safety valves.

The Garrett PORVs used at Beaver Valley Unit 2 closely resemble the valve tested by EPRI. Therefore, the test valve is considered representative of the plant valve.

The block valves used at Beaver Valley Unit 2 are identical to the valve tested by EPRI. The valve was tested by EPRI in a horizontal configuration. The valve is designed for use in either a horizontal or vertical orientation. The plant valves have Limitorque SB-00-15 motor operators which is the Limitorque operator used with the test valve.

When questioned on the torque switch setting of the plant operator and the torque produced at that setting, the Licensee stated the rated torque for a similar operator was 250 ft-lb. This response is not considered adequate to ensure the plant operators are set to produce a torque greater than 175 ft-lb, the minimum torque tested. Therefore, the response is not acceptable to meet the requirements of NUREG-0737, Item II.D.1. The plant specific block valve/operator combination must be shown to be adequate based on test data. To use the EPRI test data to support the operability of the plant specific block valve/operator combination, the plant operator must be set to produce a torque greater than the minimum torque tested. Until adequate information is supplied to verify this, the EPRI block valve test results cannot be used to demonstrate operability of the Beaver Valley Unit 2 valves.

Based on the discussion above, the safety valve and PORV tested are considered to be applicable to the plant valves at Beaver Valley Unit 2, and to have fulfilled Items 1 and 7 of Section 1.2 in this report regarding applicability of the test valves. The block valve at Beaver Valley Unit 2 is the same as the valve tested by EPRI. Therefore, Item 1 of Section 1.2 was also met for the block valve. However, the block valve/operator combination tested by EPRI is not considered to be applicable to the block valve/operator combination at Beaver Valley Unit 2; therefore, Item 7 of Section 1.2, regarding applicability of the test valves, was not met for the block valves.

4.2 Test Conditions

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 valve inlet conditions applicable to Beaver Valley Unit 2 are those identified for Westinghouse three-loop plants.

For FSAR transients resulting in steam discharge, the safety valves in three-loop plants experience a peak pressure of 2592 psia and a maximum pressurization rate of 216 psi/s. The limiting transient in both cases was the locked rotor transient. According to Reference 12, the discharge piping system at Beaver Valley Unit 2 will develop a 485 psia backpressure during a safety valve actuation. This backpressure was calculated assuming the PORVs remain closed. Therefore, the expected backpressure during the normal valve opening sequence of PORVs followed by the safety valves would be higher than 485 psia. However, the backpressure is not expected to exceed the highest backpressure (710 psia) in the EPRI tests. Based on Reference 12, the ring settings for all the safety valves at Beaver Valley Unit 2 are manufacturer recommended and set ring settings.

The Crosby HB-BP-86 6M6 safety valve was subjected to eleven loop seal-steam tests with a long inlet configuration in the EPRI testing program. Of these tests, two (1415 and 1419) are applicable to the Beaver Valley Unit 2 safety valves with hot loop seals (290 and 350°F in the EPRI tests and 400°F at Beaver Valley Unit 2). The ring settings in these two tests were manufacturer recommended settings of (-77, -18) relative to the bottom of the disc ring. In these tests, the peak pressure was greater than 2675 psia. The pressurization rate for both tests was 360 psi/s with backpressures of 250 and 255 psia. The inlet conditions for these tests bound those at the plant except for the backpressure condition. One test was run with a backpressure of 710 psia and a colder loop seal than the loop seal at Beaver Vally Unit 2 (929, 90°F loop seal water temperature). This test used manufacturer recommended ring settings of (-71, -18). Also, the peak pressure, 2726 psia, and pressurization rate, 319 psi/s, bound the conditions expected at Beaver Valley Unit 2. The use of the colder loop seal test to bound the Beaver Valley Unit 2 condition is acceptable because the maximum backpressure does not develop until after the loop seal is discharged and full steam flow is developed. By considering these three tests, all the conditions expected at Beaver Valley Unit 2 are bounded by the EPRI tests.

For FSAR transients resulting in steam discharge, the PORVs will open at a pressure somewhat above the opening setpoint of 2350 psia. A maximum pressurizer pressure of 2555 psia is expected based on the locked rotor accident (Reference 7). The peak backpressure seen by the PORVs was not provided by the Licensee. However, the PORV rated flow, 210,000 lb/h, is 61% of the safety valve rated flow, 345,000 lb/h. The three PORVs discharge into 6 in. piping as do the safety valves. Based on this data, the conclusion is reached that the expected backpressure for the PORVs is less than the 485 psia backpressure for the safety valves.

The Garrett test PORV was subjected to thirteen steam tests, one transition test, and two water seal simulation tests in the EPRI test program. In the steam tests, the maximum pressure at valve opening ranged from 2415 to 2760 psia. The valve opening pressure for the steam-water transition test was 2760 psia. The two water seal tests were conducted at initial pressures of 2755 and 2760 psia and inlet fluid temperatures of 130 and 293°F. The plant PORV water seal temperature is estimated to be 400°F (Reference 12) based on plant measurements. The maximum back pressure for these tests ranged from 25 to 875 psia. The valve opening time is relatively short (~0.5 s). The test fluid conditions in the steam and water seal tests on the PORVs are representative of FSAR transients.

The only FSAR transient resulting in liquid discharge through the safety valves is a main feedwater line break. The PORVs are assumed not to actuate during this transient. The maximum pressure during the liquid discharge case is 2503.7 psia with a maximum pressurization rate of 8.0 psi/s. Fluid temperatures at the valve inlet range from 553.8 to 572°F with a maximum pressurizer liquid surge rate of 225.4 gpm.

The Crosby 6M6 valve was subjected to one loop seal-transition test (931a) with a peak pressure of 2578 psia. Test 931a was a cold loop seal test. Because the EPRI tests showed better valve performance with hot loop seals, the valve behavior in this test conservatively bounds that for Beaver Valley Unit 2 with hot loop seals. The Crosby 6M6 valve was also subjected to one saturated water test (931b), with a peak pressure of 2475 psia and a valve inlet temperature of 635°F, and one cold water test (932), with a peak pressure of 2520 psia and a water temperature of 515°F. The pressurization rate for both the transition and water tests ranged from 2.5 to 3.0 psi/s with backpressures of 650 to 725 psia. The transition and water tests were conducted with ring settings of (-71, -18), which are typical factory set plant ring settings similar to the factory set ring settings at Beaver Valley Unit 2.

During the FSAR analysis of the feedwater line break (FWLB), it was assumed the PORVs were not operable. During a FWLB it is expected the PORVs would open and the inlet conditions would be similar to those experienced by the safety valves (discussed above). The Garrett PORV was subjected to one transition test and three high pressure water tests. In the transition test, the peak pressure was 2760 psia and the water temperature was 653°F. In the water tests, the pressure ranged from 2640 to 2760 psia and water temperatures ranged from 249 to 648°F. These conditions bound those expected for the Beaver Valley Unit 2 PORVs.

For Extended High Pressure Injection (HPI) events, the limiting transient was a spurious activation of the high pressure injection system at power. Only the PORVs are challenged for three loop plants (Reference 7). However, Reference 7 also notes that it takes at least 20 minutes to fill the pressurizer. This is more than enough time for the operator to terminate the transient before the valve must discharge liquid. Thus, the potential for liquid discharge in the extended HPI event can be disregarded.

Low temperature overpressurization events challenge only the PORVs since these are used to mitigate such transients. The fluid conditions for these events can vary from steam to subcooled water. The range of potential fluid conditions for low temperature overpressure events was provided in Reference 12. The fluid temperatures range from 70 to 350°F and the maximum pressures range from 464 to 730 psia.

In addition to the high pressure water, steam, and transition tests previously mentioned, the PORV was subjected to two low pressure water tests. The pressures in these tests were 683 and 686 psia while the valve inlet temperatures were 94 and 460°F. These test conditions, together with the test conditions in the high pressure tests, sufficiently encompass the range of expected fluid conditions for low temperature overpressure events at Beaver Valley Unit 2.

The block valves are required to operate over a range of fluid conditions (steam, steam-to-water, water) similar to those of the relief valves. The block valves, however, were only tested under full pressure steam conditions (to 2485 psia). With similar internal materials under full pressure steam conditions, the required torque to open or close the valve: (a) depends almost entirely on the differential pressure across the valve disk, (b) is rather insensitive to momentum loading, (c) is nearly the same for water or steam, and (d) is nearly independent of the flow. The full pressure steam tests, therefore, are adequate to show valve operability for steam and water conditions.

However, as noted in Section 4.1 in this report, sufficient information was not supplied by the Licensee to determine whether the plant specific block valve operator is set to produce sufficient torque to close the block valve. Therefore, although the EPRI test conditions bound those for the plant block valves, the test results are not applicable to the Beaver Valley Unit 2 block valve/operator combination.

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

4.3 Operability

As discussed in the previous section the safety and relief valves are required to operate over a range of full pressure steam, steam-to-water transition, and subcooled water fluid conditions. The valves were tested for the range of required conditions in the EPRI test program. The block valves are also required to operate for steam and liquid flow conditions. These valves were subjected to full pressure steam tests, the results of which apply also to liquid flow.

Because the Beaver Valley Unit 2 Crosby $6M_16$ safety valve was not tested, operability will be demonstrated based on testing of the Crosby 6M6 safety valve with typical plant ring settings and similar inlet fluid conditions.

Tests 1415 and 1419 were performed on the Crosby 6M6 valve with typical plant ring settings of (-77, -18). These correspond to the plant ring settings of the Beaver Valley Unit 2 safety valves which use factory set ring settings. During Test 1415, the Crosby 6M6 safety valve opened, fluttered or chattered in a partial lift position during loop seal discharge, then popped open, stabilized on steam, and closed with 6.2% blowdown. The valve opened at 2555 psia (+2.2% of the nominal set pressure). Though the test valve was in a lift position as low as 92% of rated lift, the valve flow rate reached 109% of rated steam flow at 3% accumulation.

Test 1419 was run with 350^{o}F loop seal water and was designed to be a repeat of Test 1415. The valve performance in Test 1419 raised a concern with respect to the Beaver Valley Unit 2 Crosby 6M$_1$6 safety valves. The concern is in regards to the reliability of the valve rather than stability in a specific EPRI test. In Test 1419, the valve reopened after closure, chattered, and the test was terminated by manually opening the valve. This result does not indicate a valve closing problem for the Beaver Valley Unit 2 safety valves because an identical test (1415) 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 the Crosby 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 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 subsequent tests. Test 1415 was performed immediately after valve maintenance and the valve performed stably. The next 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 this valve.

TABLE 4.3.1 EPRI TESTS ON CROSBY HB-BP-86 6M6 SAFETY 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 sets of ring settings were tested. Actual ring positions not shown.
Terminated--Test terminated after valve was manually opened to stop chatter.

Therefore, the Licensee should inspect the safety valves after each lift involving loop seal or water discharge, and a formal procedure requiring the inspection should be developed and incorporated into the plant operating procedures or licensing documents such as the plant technical specifications. The Licensee stated that they were willing to incorporate a requirement to inspect the safety valves after any lift involving liquid discharge, such as in a feedwater line break (Reference 12). However, based on the EPRI tests, the Licensee should also inspect the safety valves after any lift involving loop seal discharge.

Tests 931a, 931b, and 932 with typical plant ring settings of (-71, -18) simulate the expected Beaver Valley Unit 2 FWLB conditions. Test 931a was a loop seal/steam/water transition test. The Crosby 6M6 valve opened, fluttered or chattered in a partial lift position during loop seal discharge, then popped open, stabilized during steam and water discharge, and closed with a 12.7% blowdown. At 2415 psia with 641^{O}F water, the valve passed 2355 gpm of liquid with the valve at 56% of rated lift. In Test 931b, the valve opened on 635^{O}F water, chattered during opening, stabilized, and closed with 4.8% blowdown. The liquid flow rate in Test 931b was not recorded. For these tests the valve opened within -1% and +3% of the set pressure.

In Test 932, however, the valve opened and immediately began to chatter. The valve chattered for 6.5 s before the test was terminated by manually opening the valve. This test used 515^{O}F water. The temperatures expected in a FWLB at Beaver Valley Unit 2, 553.8 to 572^{O}F, fall between the available test data at 515 and 640^{O}F. In Reference 12, the Licensee stated that a Westinghouse report, <u>Pressurizer Safety Relief Valve Operation For Water Discharge During A Feedwater Line Break</u>, WCAP-11677, January 1988, concluded that all plants addressed in the evaluation had initial and final FWLB water relief temperatures above 600^{O}F. Although Beaver Valley Unit 2 was not specifically addressed in the Westinghouse report, the Licensee stated that Westinghouse confirmed the FSAR FWLB analysis water relief temperatures at Beaver Valley Unit 2 are above 600^{O}F. Therefore, the FWLB analysis water relief temperatures at Beaver Valley Unit 2 are enveloped by the report. Based on this, the Licensee concluded water relief temperatures below 600°F are not expected at Beaver Valley Unit 2. The report referred to by the Licensee, WCAP-11677, was not provided in the Licensee's submittal for EG&G Idaho, Inc., to review. Therefore, the results of the analysis cannot be verified with respect to the water temperatures expected during a FWLB. Based on the information supplied by the Licensee, it is not clear whether the valve will operate stably or chatter at the expected inlet temperatures. Therefore, the Licensee has not demonstrated the operability of the plant safety valves for conditions expected during a FWLB.

As discussed above, the observed blowdown in the applicable EPRI tests ranged from 4.8 to 12.7%, which exceeds the design value of 5%. Thus, it must be demonstrated these extended blowdowns will not impact plant safety and valve operability. From a valve operability standpoint, filling the pressurizer is not a concern. The Crosby 6M6 safety valves at Beaver Valley Unit 2 were shown in the EPRI tests to be operable with steam, steam/water transition, and saturated water inlet conditions. Blowdown of 12.7% from a valve setpoint of 2500 psia should not present a challenge to plant protection equipment; therefore, this was not considered a safety concern. A second concern with extended blowdown is the possibility of voiding in the primary coolant system causing a significant loss of decay heat removal capability. To resolve this concern, data from the Semiscale natural circulation (NC) test series was reviewed (Reference 13). This data applies to plants with U-tube steam generators. The NC test series showed the various modes of the NC (single-phase, two-phase, and reflux) were able to keep the core cool at decay heat levels of 1.5 to 5% and at primary system mass inventories of 55 to 100%. Thus, if any voiding of the primary system due to extended blowdown should occur, it would not endanger the core because forced circulation (early in the transient) and NC (late in the transient) would continue to remove the decay heat.

For the tests to adequately demonstrate safety valve stability, the test inlet piping pressure drop should exceed the plant pressure drop. The test inlet pressure drop for the Crosby 6M6 valve on the loop seal configuration was 263 psid on opening and 181 psid on closing. Calculated values for the Beaver Valley Unit 2 safety valves were not provided. In Reference 12, the Licensee stated it compared the inlet piping geometry and pipe velocities to those in the EPRI tests. The Licensee stated the inlet piping is shorter and the pipe velocities lower than those in the EPRI tests; therefore, the Beaver Valley Unit 2 configuration is bounded by the EPRI configuration. Based on this comparison, the Licensee concluded the plant valves should be as stable as the test valves. This response is considered inadequate. The request for information clearly asked the Licensee to compare the calculated pressure drop for the plant and the EPRI tests. This approach was recommended in EPRI material available to utilities participating in the EPRI Valve Test Program (Reference 14). Therefore, until the requested comparison is provided, the Licensee has not shown the plant valves will operate as stably as the test valves.

Bending moments as large as 268,875 in-lb were induced on the discharge flange of the Crosby 6M6 safety valve during the applicable tests; valve performance was unaffected. The largest predicted bending moment for the Beaver Valley Unit 2 safety valves is 259,068 in-lb (Reference 12). Therefore, bending loads on the safety valve bodies induced by actuation will not affect safety valve operability.

For all applicable tests on the Garrett PORV, the valve opened and closed on demand. Total valve opening times were less than 1.24 s and closing times were less than 2.35 s. Following test completion, the valve was inspected. No damage was observed that would affect future valve performance. Based on valve performance during the EPRI tests, the demonstration of PORV operability under expected inlet conditions was adequate.

With respect to the PORVs, NUREG-0737, Item II.D.1, was concerned with the operability of the PORVs; that is, will the valves open and close on demand for the expected inlet fluid conditions. The Garrett PORV tested was shown to adequately meet this requirement for the Beaver Valley Unit 2 PORVs. However, during the review it was noted that the plant PORVs may not pass the rated flow based on the EPRI test data. According to the plant FSAR, the PORV rated steam flow is 210,000 lb/h. The test valve passed

378,000 lb/h of steam at 2415 psia. The orifice in the Beaver Valley Unit 2 PORVs is, however, 44% the size of the orifice in the test PORV; thus, the plant PORVs would be expected to pass only 44% of the test valve flow. This results in a flow rate of 166,320 lb/h or 80% of rated flow. Based on this data, the Licensee may want to verify the flow rating of the plant PORVs.

A bending moment of 33,200 in-lb was induced on the discharge flange of the Garrett test PORV, which has nearly the same valve body as the plant PORV. This moment had no adverse effect on valve performance. The Licensee did not provide the maximum calculated bending moment for the plant PORVs but simply stated the PORVs at Beaver Valley Unit 2 do not have flanges (Reference 12). This response is considered inadequate. Although the Beaver Valley Unit 2 PORVs do not have flanges, the valves still experience bending moments as a result of various loads during valve actuation. An appropriate point in the piping system could have been chosen for comparison of the calculated bending moment to the EPRI data. Until the Licensee provides this comparison, the Licensee has not demonstrated the PORVs will operate under the maximum expected bending moment.

NUREG-0737, Item II.D.1, requires qualification of associated control circuitry as part of the safety/relief valve qualification. If the Licensee meets the requirements of 10 CFR 50.49 for this electrical equipment, then specific testing per the NUREG-0737 requirement is not required. The Licensee provided information to show the solenoid operated pilot valve for the PORV was included in an environmental qualification program designed to meet the requirements of Reg. Guide 1.89, which defines the type of components and testing required to meet 10 CFR 50.49 (Reference 10). This ensures that the control circuitry will function properly.

Because the information provided by the Licensee does not adequately demonstrate that the plant specific block valve operator torque switch is set so the operator can close the block valve, the EPRI tests, conducted on the Westinghouse 3GM88V block valve and Limitorque SB-00-15 operator, are not applicable to the Beaver Valley Unit 2 block valves. Thus, operability of the block valves was not demonstrated.

The above discussion verifies the part of Item 1 of Section 1.2 in this report which requires conducting tests to qualify the valves was met for the safety valves, PORVs, and block valves, except for the operation of the safety valves under FWLB conditions. The information supplied by the Licensee was not sufficient to conclude the plant safety valves will operate satisfactorily during a FWLB. However, because of differences between plant and test valve installations, the tests do not qualify the plant safety valves, PORVs, or block valves. This is because the Licensee did not meet Item 7 of Section 1.2 in this report. For the safety valves, this item was not met because the information provided by the Licensee did not directly compare the inlet piping pressure drop at Beaver Valley Unit 2 to the EPRI tests. Therefore, stable operation of the safety valves in the plant specific configuration could not be assured. Also, the Licensee must document a formal procedure for the inspection of the safety valves. The part of Item 7 which requires that the effect of discharge piping on operability be considered was met for the safety valves but not for the PORVs. This is because the Licensee did not compare the maximum calculated bending moment for the plant PORVs to that in the EPRI tests. For the block valves, the test results are not directly applicable to the plant valves. Meeting the requirements of 10 CFR 50.49 is adequate to satisfy Item 5 of Section 1.2 in this report regarding the PORV control circuitry.

4.4 Piping and Support Evaluation

The piping from the pressurizer nozzles to the safety valves and the PORVs was analyzed as ASME Section III, Class 1, piping. The piping from the valves to the pressurizer relief tank was analyzed as ASME Section III, Class 3, piping. The supports were analyzed for the requirements of the ASME Code, Section III, ANSI B31.1, AISC Manual of Steel Construction, and the AISC Specification. The piping was analyzed for thermal expansion, pressure, weight, earthquake, and safety valve and relief valve discharges. The load combinations and acceptance criteria were based on the Beaver Valley Unit 2 FSAR, which is acceptable

The safety valve and relief valve discharge loads were calculated for the fluid transient condition that will produce the most severe loading on the piping system. For the safety valve piping and the PORV discharge by approximately 62%. With respect to the use of the stated density and squared-edged water slugs in the WATSLUG analysis, the conservatism from these assumptions does not appear to be very large. Figures 3A.3.8-3/4 of the same FSAR amendment compare WATSLUG forcing functions to RELAP5/MOD1 (which was verified for use in calculating this type of load in Reference 16) forcing functions for an EPRI test. The comparisons show that the WATSLUG forcing functions, in a calculation that also used the measured valve flow rate, were approximately the same as those calculated by RELAP5/MOD1. Therefore, the density and square-edged slug assumptions do not add to the conservatism of the thermal-hydraulic analysis. With respect to the overall conservatism of the NUPIPE-SW results, this is clear from the comparison in Figure 3A.3.8-6. But considering the above statements in regards to the WATSLUG analysis, the conservatism would appear to be from the structural analysis assumptions not the WATSLUG results. Therefore, the safety valve thermal-hydraulic analysis completed by the Licensee is not considered adequate using the safety valve rated flow.

The structural analysis was performed using Stone & Webster Engineering Corporation's version of NUPIPE. This is a linear elastic piping structural analysis program widely used in industry which is fully verified for pipe stress analysis. The NUPIPE code was benchmarked by the NRC in 1979 as part of a five plant review conducted by SWEC.

The key structural analysis parameters of lumped mass spacing, integration time step, and cutoff frequency are adequate. Lumped mass spacing was selected so the model contained at least one mass point between restraints active in the same direction. Mass points were also input at locations with concentrated weights, intersections, elbows, reducers, and each terminus. Integration time steps were 0.0007 to 0.0013 s and the Licensee stated that a cutoff frequency of 300 Hz was used. Although the stated cutoff frequency was 300 Hz, the time steps used in the analysis, 0.0007 to 0.0013 s, would only allow frequencies from 100 to 180 Hz to be accurately calculated. However, this is still considered adequate. The damping factor used in the analysis was not provided by the Licensee. However, based on other SWEC analyses reviewed it was assumed that a damping factor of 1% was used, which is adequate.

The results of the piping analysis showed the pipe stresses in the safety valve and PORV inlet and outlet lines were less than their allowables. For the supports, three supports were added to the safety valve inlet lines during 1985 and 1986 to reduce the loads on the safety valve flanges and pressurizer nozzles below allowable limits.

Testing, by EPRI, of the Crosby 6M6 safety valve with loop seals and water resulted in high frequency pressure oscillations upstream of the valve. A concern was raised as to whether these pressure oscillations could excite vibration modes creating bending moments in the pipe which could damage the pipe. Beaver Valley Unit 2 has 6 in., Sch. 160, pipe upstream of the safety valves. This inlet piping is similar to that used in the EPRI tests at CE which had 8 in., Sch. 160, and 6 in., Sch. XX, piping upstream of the test valve. No damage to the inlet piping was noted in the EPRI tests. While the wall thickness of 6 in., Sch. 160, pipe is approximately 80% the wall thickness of 6 in., Sch. XX, pipe, they are sufficiently close in size to conclude the pressure oscillations will not cause damage to the plant inlet piping.

The discussion above, demonstrating that a bounding load case was chosen for the piping evaluation, verifies that Item 3 of Section 1.2 in this report was met. Based on the information provided by the Licensee, the thermal-hydraulic and structural analyses are considered adequate to meet the NUREG-0737 requirements for the PORV piping. Therefore, Item 8 of Section 1.2 in this report was met for this piping. However, based on the information provided by the Licensee it cannot be concluded that the analysis of the safety valve piping and support system was adequate to meet Item 8 of Section 1.2.

5. EVALUATION SUMMARY

The Licensee for Beaver Valley Unit 2 has not provided an acceptable response to the requirements of NUREG-0737. Therefore, they have not reconfirmed that the General Design Criteria 14, 15, and 30 of 10 CFR 50 Appendix A, were met. The rationale for this conclusion is given below.

5.1 NUREG-0737 Items Fully Resolved

Based on the following information provided by the Licensee, the requirements of Item II.D.1 of NUREG-0737 were partially met (part of Item 1, Items 2 to 6, and part of Item 7 of Section 1.2 in this report).

The Licensee participated in the development and execution of an acceptable test program. The program was 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 inlet conditions which by analysis bounded the most probable maximum forces expected from anticipated design basis events. The generic test results and piping analyses showed the test valves functioned correctly and safely for all relevant steam discharge events specified in the test program, and the pressure boundary component design criteria were not exceeded. Analysis and review of the test results and the Licensee's justifications indicated, except as discussed in Section 5.2 in this report, direct applicability of the prototypical valve and valve performance to the in-plant valves and systems intended to be covered by the generic test program.

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

5.2 NUREG-0737 Items Not Resolved

Based on the Licensee's submittal, the following requirements of NUREG-0737, Item II.D.1, as shown in Section 1.2 in this report, were not met:

1. Item 1: The part of Item 1, which requires tests to be conducted to qualify reactor coolant system safety valves, was not met for the FWLB. This is because the test results applicable to the FWLB indicate the safety valves may or may not operate acceptably. Based on the information provided by the Licensee, it cannot be concluded the safety valves will operate acceptably under FWLB conditions.

2. Item 7: The part of Item 7, which requires the consideration of the effect of as-built discharge piping on safety valve and PORV operability, was not met. This is because the calculated pressure drop when the plant safety valves open was not directly compared to the corresponding values for the test valves. Therefore, it could not be concluded the Beaver Valley Unit 2 safety valves will perform as stably as the test valve. Also, the maximum expected bending moment on the Beaver Valley Unit 2 PORVs was not supplied. Thus, operability of the PORVs with the maximum expected applied moment could not be assured.

3. Item 7: The part of Item 7, regarding the applicability of the test valves, was not met for the block valves. The test results on the valve/operator combination tested by EPRI are not applicable to the block valve/operator combination at Beaver Valley Unit 2. This is because, based on the information provided by the Licensee, it cannot be concluded the plant operators are set to produce a torque greater than the minimum torque used in the EPRI tests.

4. Item 8: Item 8, which requires qualification of the piping and supports, was not met for the safety valve discharge piping. This

is because, based on the information provided by the Licensee, it could not be concluded the thermal-hydraulic analysis portion of the safety valve piping analysis was adequate to meet the NUREG-0737, Item II.D.1, requirements.

The test results demonstrated the need for inspection and maintenance of the safety valves following each lift involving loop seal or water discharge. A requirement to inspect the valves needs to be incorporated into the plant operating procedures or technical specifications.

Therefore, the Licensee has not demonstrated 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) were designed with sufficient margin such that design conditions are not exceeded during relief/safety valve events (General Design Criterion No. 15).

6. REFERENCES

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