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TECHNICAL EVALUATION REPORT
TMI ACTION--NUREG-0737 (II.D.1)
BYRON UNITS 1 and 2
DOCKET NO. 50-454, 50-455

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ABSTRACT

Light water reactors have experienced a number of occurrences of improper performance of safety and relief valves installed in the primary coolant system. As a result, the authors of NUREG-0578 (TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations) and subsequently NUREG-0737 (Clarification of TMI Action Plan Requirements) recommended that programs be developed and completed which would reevaluate the functional performance capabilities of Pressurized Water Reactor (PWR) safety, relief, and block valves and which would verify the integrity of the piping systems for normal, transient, and accident conditions. This report documents the review of these programs by the Nuclear Regulatory Commission (NRC) and their consultant, EG&G Idaho, Inc. Specifically, this review examined the response of the Licensee for the Byron Nuclear Power Station, Units 1 and 2, to the requirements of NUREG-0578 and NUREG-0737 and finds that the Licensee provided an acceptable response, reconfirming that the General Design Criteria 14, 15, and 30 of Appendix A to 10 CFR 50 were met.

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1. INTRODUCTION

1.1 Background

Light water reactor experience has included a number of instances of improper performance of relief and safety valves installed in the primary coolant system. There 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 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 Genera 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 Commonwealth Edison Co. (CECo), owner of the Byron Nuclear Power Station, Units 1 and 2, 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 Byron, Units 1 and 2, were 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.

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 Byron, Units 1 and 2, was submitted by CECo to the NRC on October 26, 1982 (Reference 11). This was followed by a submittal of additional information regarding piping and support adequacy on December 30, 1983 (Reference 12). A request for additional information was submitted to CECo by the NRC on March 26, 1987 to which CECo responded on December 2, 1987 (References 13 and 14).

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

Byron, Units 1 and 2, are four-loop PWRs designed by the Westinghouse Electric Co. Each unit 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 SS 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 PORV block valves are 3-in. Westinghouse Model 3000GM88 gate valves with Limitorque SB-00-15 motor operators. The inlet pipe to the safety valve includes a hot loop seal (282^{o}F); the inlet to the PORV has a cold loop seal (170^{o}F).

Safety valves, PORVs, and PORV block valves identical to those used at Byron, Units 1 and 2, 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 Byron, Units 1 and 2. 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, Byron, Units 1 and 2, are four-loop PWRs 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 cold 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 Byron PWRs, 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 533 psia (Reference 14). Hot loop seals are used at the safety valve inlet. The loop seal temperature at the inlet to the valve is 282^{o}F. The safety valves at Byron, Units 1 and 2, use manufacturer's recommended ring settings.

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 were considered. For steam flow conditions, four loop seal discharge tests (Test No. 929, 1406, 1415, 1419) were applicable to Byron, Units 1 and 2. These tests were performed with valve ring settings representative of the typical ring settings used in Westinghouse PWRs including Byron. 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. 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 two sets of ring settings are considered comparable to each other. The loop seal temperature measured in the tests ranged from 90 to 360^{o}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 show that the inlet fluid conditions and backpressures of these tests envelop the corresponding fluid data predicted for the Byron 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 Byron. A test simulating loop seal discharge was conducted at a pressure of 2725 psia with a water temperature of 134°F at the valve inlet. The backpressure developed at the outlet of the PORVs is not an important consideration because the air operated PORVs used at the Byron plant are not sensitive to backpressure (Reference 6). Therefore, the EPRI test inlet fluid conditions for the PORV with 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). The submittal did not address the transient conditions that involve liquid discharge through safety valves and PORVs. The Licensee stated that its decision not to evaluate the Byron 1 and 2 safety valves and PORVs for liquid discharge was based on a probabilistic risk study presented in Appendix A of Reference 11. This study concluded that liquid discharge in the feedline break, extended high pressure injection, or low temperature overpressurization event was very unlikely to occur. Therefore, liquid discharge through the safety valves and PORVs was not considered.

However, the Westinghouse Valve Inlet Fluid Condition Report (Reference 7) stated that the main feedline pipe rupture event was classified as a Class IV licensing event. That is, one which was not expected to take place but was postulated because its consequences include the potential for the release of a significant amount of radioactive material. Also, NUREG-0737 specifically requires the safety valves and PORVs be qualified for inlet fluid conditions resulting from transients and accidents referenced in Regulatory Guide 1.70, Rev. 2. The feedwater line break is specifically defined in Regulatory Guide 1.70, Rev. 2. From a review of the feedwater line break analysis for Byron (see below), 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 even though the probabilistic analysis showed the frequency of occurrence is extremely low.

The Byron feedline break analysis indicated that the safety valves and PORVs opened on saturated steam at about seven minutes into the transient and steam to saturated liquid transition would follow at thirteen minutes into the event (Appendix A, Reference 11). It is apparent that liquid discharge through the safety valves and PORVs cannot be ruled out. Therefore, valve operability will be reviewed using the feedline break data provided in Reference 7.

Reference 7 showed that, in a feedline break accident at Byron, the maximum pressure at the safety valve inlet during liquid discharge was calculated to be 2508 psia and the pressurization rate was 3.5 psi/s. Fluid temperatures at the valve inlet range from 615 to 635°F and the maximum liquid surge rate into the pressurizer is 569 gpm.

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 (Test 931b). Because the inlet temperature and pressure of the tests compare favorably with the predicted in-plant conditions, the results of these tests are applicable to the Byron safety valves.

The expected fluid conditions at the safety valve inlet 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 2508 psia and 635°F for the Byron plant. Therefore, these tests are 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. 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 low temperature overpressure protection (LTOP) during the low temperature stages of reactor start-up and shutdown operations. The expected valve inlet conditions in the low temperature overpressure protection mode were given in Reference 14 as pressures ranging from 350 to 2450 psig and water temperatures ranging from 70 to 450°F. It is also possible for the PORV to actuate under high and low pressure steam conditions.

For steam discharge through the PORV, the high pressure steam tests discussed above cover the low pressure steam conditions predicted for LTOP. 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 and 442°F, respectively. In addition there was a high pressure water test with a pressure of 2535 psia and a water temperature of 647°F. These conditions are representative of those at Byron. Therefore, the EPRI tests can be used to evaluate the performance of the Byron PORV during LTOP transients.

4.2.5 PORV Block Valve Fluid Conditions

The block valves at Byron are Westinghouse 3 in. gate valves, Model 3000GM88, with Limitorque SB-00-15 operators. The block valves are required to operate over a range of fluid conditions (steam, steam-to-water, water) similar to those of the PORVs. The 3-in. Westinghouse 3GM88 valve with Limitorque SB-00-15 operator was subjected to full pressure steam conditions (to 2485 psia) in the EPRI-Marshall tests. Later tests were performed by Westinghouse using subcooled water as a test fluid. Tests were conducted at differential pressures across the valve discharge ranging from 800 to 2600 psi and with flow rates ranging from 60 to 600 gpm. The Westinghouse tests showed that the torque required to operate the valve is almost entirely dependent on the differential pressure and is rather insensitive to momentum loading. Thus operability of the valve is nearly the same for steam and liquid discharge, and the results of the EPRI-Marshall tests can be used to assess operability of the valve.

4.2.6 Test Conditions Summary

The test sequences and analyses described above demonstrate that the test conditions bound the conditions for the plant valves. They also verify that Items 2 and 4 of Section 1.2 were met, in that conditions for the operational occurrences were 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, was also met.

4.3 Operability

4.3.1 Safety Valves

The EPRI tests representative of the steam discharge condition for the Byron 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 ±2% 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 Byron 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 each inspection and the damaged parts were refurbished or replaced before the next test started. The test results showed that the valve performed acceptably in the test following each repair, but that closing chatter recurred in a subsequent test. Test 1415 was performed immediately after valve maintenance and the valve performed stably. The next test (Test 1419) encountered chatter in closing though it was a repeat of Test 1415. These results suggest 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

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	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	0.75	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.

after each actuation and the procedure should be incorporated into the plant operating procedures or licensing documents such as the plant technical specifications.

Blowdown in these tests (5.1 to 9.4%) was 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 increased blowdown that 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 Byron 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 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 Byron, Units 1 and 2, during the feedline break transient is 569 gpm. The 6M6 valve tested by EPRI passed 235[illegible] gpm at 2415 psia and 641°F which is much higher than the predicted flow rate for Byron. The above results demonstrate that the Byron safety valves would be adequate to perform the required water relief function.

Bending moments as high as 298,750 in-lb (Test 908) were induced on the discharge flange of the Crosby 6M6 test valve, which had no adverse effect on valve performance. Because this applied moment exceeds the maximum estimated bending moment of 177,460 in-lb for the Byron valves (Reference 14), the performance of the plant valves is also expected to be unaffected by bending moments imposed during discharge transients.

For the tests to be an adequate demonstration of 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 253 psid on opening and 181 psid on closing. The values calculated for the Byron, Units 1 and 2, safety valves were 235 and 120 psid for opening and closing, respectively (see Reference 14). Therefore, the plant valves should be as stable as the test valves.

4.3.2 Power Operated Relief Valves

The EPRI tests on the Copes-Vulcan PORV with 316 SS stellited plug and 17-4 PH cage demonstrated that the valve opened and closed on demand in loop seal/steam, 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 Byron PORVs.

The predicted value of the maximum bending moment induced at the Byron PORV discharge flange was calculated to be 65,860 in-lb (Reference 14). This exceeds the maximum tested bending moment of 43,000 in-lb. Therefore, operability of the Byron PORVs with the maximum expected bending moment cannot be shown directly using the EPRI data. However, CECo stated in Reference 14 that Westinghouse qualified the Byron PORVs for these loads by analysis. The maximum loading of 65,860 in-lb results in stresses of 71% of the allowable value. Therefore, the Copes-Vulcan PORVs at Byron are expected to operate with the higher bending moments. This is reasonable because the EPRI test bending moments represent the maximum value tested, not the maximum permissible bending moment.

4.3.3 Electric Control Circuitry

NUREG-0737 Item II.D.1 states that the control circuitry associated with the PORVs shall also be qualified for design basis accidents and transients. The Nuclear Regulatory Commission staff agreed that meeting the licensing requirements of 10 CFR 50.49 for this electrical equipment is satisfactory and that specific testing per the NUREG-0737 requirements is not necessary. CECo included the PORV controls that would be located in a harsh environment in the Byron 10 CFR 50.49 environmental qualification program (Reference 14) thereby ensuring that the control circuitry will function properly. CECo stated the NRC reviewed and approved the Byron environmental qualification program (NUREG-0876, Supplement 5).

4.3.4 PORV Block Valves

The PORV block valve must be capable of closing over a range of steam and water conditions. As described in Section 4.2, results from the high pressure steam tests can be used to evaluate valve operation over the full range of inlet conditions. The tests on the Westinghouse 3GM88 valve with Limitorque SB-00-15 actuator showed that the valve successfully opened and closed on command once the torque switch was set to its maximum. The plant block valves were also modified by adjusting the torque switches to optimal values for opening and closing thrust, adjusting the pinion gear ratios, and changing the wiring from torque controlled to limit controlled stroking. With these changes, the plant valves are expected to operate acceptably.

4.3.5 Operability Summary

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, were met provided the Licensee documents a formal procedure for the inspection of the safety valves as discussed in Section 4.3.1. Also, the licensing action for 10 CFR 50.49 satisfies 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 nozzle to the pressurizer relief tank. The piping was designed for deadweight, 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 low temperature overpressurization events including the single failure that would maximize the forces on the valve.

The safety valve and PORV discharge transients were analyzed as two separate events. This approach is acceptable, since the safety valves and PORVs have different set points and will not lift simultaneously. Also the sequential discharge of the PORVs and safety valves (i.e., PORV discharge followed by safety valve discharge) would not generate higher piping loads than the separate PORV and safety valve discharge events. Therefore, the valve discharge conditions considered for the piping and support analyses were the steam discharge condition resulting from the simultaneous actuation of all safety valves and the steam discharge condition resulting from the simultaneous actuation of all PORVs. Because water seals are maintained upstream of the safety valves and PORVs, steam discharge conditions would generate the highest loads on the piping system when the water slug is expelled from the loop seal and forced down the discharge piping. Therefore, the selection of the steam discharge cases as the limiting conditions for the evaluation of the piping loads is considered adequate.

For this analysis, steam at a pressure of 2575 psia and enthalpy of 1130 Btu/lb was assumed to be discharged through the safety valves. A loop seal having an enthalpy distribution based upon a temperature profile consistent with EPRI hot loop seal Test 917 was assumed to be present upstream of the safety valves. That is, a temperature of approximately 300°F at the valve inlet and saturation temperature at the steam-water interface. The safety valves were assumed to open linearly in 0.040 s.

The initial conditions for steam discharge of the PORVs were: 2350 psia and enthalpy of 1162.4 Btu/lb. Cold water seals (170°F) were assumed upstream of the PORVs. The PORVs were assumed to open linearly in 1.0 s.

The valve opening times used for the safety valves and PORVs were greater than those measured in the EPRI tests. For the Crosby 6M6 valves used at Byron, measured pop times were on the order of 0.020 s. For the Copes-Vulcan PORVs, the measured main disk opening time was on the order of 0.40 s. However, with Byron using loop seals upstream of both the safety valves and PORVs, the opening times used in the analysis are considered adequate because the peak piping loads are due to the passage of the loop seals and valve characteristics such as opening time are less important in the calculation of these loads. This conclusion is supported by the analysis of EPRI Test 917 presented in Reference 15. When a valve opening time of 0.090 s was used, compared to a measured pop time of 0.015 s, the measured forces were still adequately calculated.

The thermal hydraulic analysis was performed using the Westinghouse computer code, ITCHVALVE. ITCHVALVE calculates the fluid parameters as a function of time. The unbalanced force or wave force in each piping segment is 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 were obtained from these calculations.

The adequacy of the ITCHVALVE/FORFUN programs for 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 were presented in Reference 14 and these comparisons are considered satisfactory.

The thermal hydraulic and stress analyses of the Byron 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 was fully reviewed in previous submittals for similar PWR plants such as Diablo Canyon Units 1 and 2 (Reference 16). The method of analysis used by Westinghouse including the analysis assumptions, the structural modeling as well as the key parameters used in computer inputs such as the node spacing, calculation time interval, valve opening time, etc. was examined and found to be acceptable. Because the Byron piping analysis followed the same method and procedure used in previous Westinghouse analyses, the analysis method is considered acceptable. The safety valve and PORV flow rates used in the analysis were greater than 120% of the rated flow. The conservative factor contained in these flow rates is more than sufficient to account for the 10% derating of the safety valve required by the ASME Code and includes allowance for uncertainties or errors.

4.4.2 Stress Analysis

The structural responses of the piping system due to safety valve/PORV discharge transients were calculated using the modal superposition method. The fluid force time histories generated from the FORFUN program were used as forcing functions on the structure. The Westinghouse series of structural analysis programs, WESTDYN, FIXFM3, and WESTDYN2, were used to calculate the piping natural frequencies, mode shapes, nodal displacements, and the internal forces and support reactions. The FIXFM3 code calculated the displacements at the structural node points using the forcing functions generated by FORFUN and the modal data from WESTDYN. 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 17). 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 18).

The important parameters in the structural analysis were reviewed and found acceptable. The time step size was 0.0005 s. Damping of 1% was used for the OBE and 2% for the SSE. Lumped mass spacing was determined to ensure all appropriate mode shapes were accurately represented. For the thermal hydraulic analysis, the cutoff frequency was greater than 333 Hz.

The piping upstream of the safety valves and PORVs was analyzed for the requirements of the ASME Code, Section III, 1977 Edition, Addenda through Summer 1979. The downstream piping was analyzed for the requirements of the ANSI B31.1 Code, 1973 Edition, Summer 1973 Addenda. The load combinations and stress limits used to evaluate the upstream and downstream piping are identical to those recommended by EPRI (Reference 19). The Licensee provided a comparison of the highest stresses in the piping against the applicable stress limits for the load combinations defined above. All stresses were within allowable stress limits (Reference 14).

The upstream pipe supports were designed in accordance with ASME Section III, Subsection NF, and the downstream supports were designed in accordance with ANSI B31.1 (Reference 14). The load combinations were consistent with the load combinations in the EPRI Submittal Guide (Reference 19), and all stresses were less than the code allowables.

In EPRI tests performed on the Crosby 6M6 safety valve, pressure oscillations of 170-260 Hz occurred in the piping upstream of the safety valve as the loop seal water was discharged. This phenomenon was not accounted for in the structural analysis of the system. The piping upstream of the safety valves in the EPRI tests was 8-in. Schedule 160 and 6-in. Schedule XX while at Byron, Units 1 and 2, the piping is 6-in. Schedule 160. The test piping did not sustain any discernible damage during the tests. Thus, the plant piping is also not expected to be damaged during similar oscillations, and an analysis for these pressure oscillations is not necessary for this plant.

4.4.3 Piping and Support Summary

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

5. EVALUATION SUMMARY

The Licensee for Byron, Units 1 and 2, 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 were 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. The Licensee must document a formal procedure to inspect the safety valves each time they discharge the loop seal or water. The plant specific piping was shown by analysis to meet code requirements.

Thus, the requirements of Item II.D.1 of NUREG-0737 were 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) were 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 was constructed in accordance with high quality standards (General Design Criterion No. 30).

6. REFERENCES

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