February 11, 1987

Docket No. 50-261

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Mr. E. E. Utley, Senior Executive
Vice President
Power Supply and Engineering
and construction
Carolina Power and Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Dear Mr. Utley,

Subject: NUREG-0737, Item II.D.1., Performance Testing of Relief and Safety Valves for H. B. Robinson, Steam Electric Plant, Unit No. 2

We have completed our review of your submittals in response to TMI Action Plan Requirements of NUREG-0737, Item II.D.1, Performance Testing of Relief and Safety Valves. A list of your submittals is included in the enclosure under References.

Based on the results of our review, we conclude that there is reasonable assurance that the relief and safety valves installed on the primary coolant system at H. B. Robinson-2 will perform their design function under accident conditions. However, test results show that the valves may not be operable for subsequent lifts. Valve disassembly, inspection and refurbishing subsequent to chattering and other problems encountered during the tests demonstrated that the valves could be operable after the inspection and refurbishing of worn or damaged parts.

Consequently, our conclusion for valve operability is contingent upon Carolina Power & Light committing to have a procedure in place to assure valve operability for subsequent lifts. This procedure should require that the safety valves be inspected internally and externally after each lift and that any necessary maintenance be performed.

Please provide us with your written commitment within 30 days from receipt of this letter. The procedures should be in place within 30 days after your commitment. Upon receipt of your commitment, we will then close this item.

A copy of our safety evaluation is enclosed for your information.

This information affects fewer than 10 respondents; therefore, OMB clearance is not required under P.L. 96-511.

Please contact your NRC Project Manager should you have any questions regarding this item.

Sincerely,

/s/

Glode Requa, Project Manager
PWR Project Directorate #2
Division of PWR Licensing-A

Enclosure:
As stated

cc w/enclosure:
See next page

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SAFETY EVALUATION REPORT

PERFORMANCE TESTING OF RELIEF AND SAFETY VALVES

(NUREG-0737, ITEM II.D.1)

H. B. ROBINSON, UNIT 2

50-261

PRINCIPAL CONTRIBUTORS:

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H.B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

DOCKET NO. 50-261

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

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

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

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

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

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

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

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

7. Submit a correlation or other evidence to substantiate that the valves tested in a generic test program demonstrate the functionability of as-installed primary relief and safety valves. This correlation must show that the test conditions 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, power operated relief valves, block valves, and associated piping systems. Carolina Power and Light (CP&L), the owner of H. B. Robinson 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.

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

EPRI contracted with Westinghouse Corporation to produce a report on the inlet fluid conditions for pressurizer safety and relief valves in Westinghouse designed plants (Reference 7). Since H. B. Robinson 2 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 (SRVs) were tested at the Kressinger Development Laboratory which is part of the Combustion Engineering Test Facility 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/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 (1) obtain valve capacity data, (2) assess hydraulic and structural effects of associated piping on valve operability, and (3) obtain piping response data that could ultimately be used for verifying analytical piping models.

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

3. PLANT SPECIFIC SUBMITTAL

A preliminary assessment of the adequacy of the overpressure protection system was submitted by CP&L on June 1, 1982 (Reference 10). Final evaluation reports covering the SRV, PORV, and PORV Block Valves followed on November 4, 1982 (References 11 and 12). An initial transmittal assessing the Pressurizer Safety and Relief Valve Piping was submitted December 28, 1982 (Reference 13). This report was superseded by one submitted August 30, 1983 which made use of more recent "as-built" information than the earlier report (Reference 14). A request for additional information (Reference 15) was submitted to CP&L by the NRC on July 5, 1984. CP&L responded to this request on November 20, 1984 (Reference 16). Additional questions regarding unresolved questions were submitted to CP&L by the NRC on May 9,1985 (Reference 17). CP&L responded to these questions on June 14, 1985 (Reference 18).

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

H. B. Robinson Unit 2 utilizes three safety valves and two PORVs. In addition, Robinson 2 employs two PORV block valves. The safety valves are Crosby Valve and Gauge model HB-BP-86 4K26 valves with loop seal internals. The PORVs are Copes-Vulcan model D-100-160. Both the safety and relief valves have loop seals upstream of the valves. The PORV block valves are Velan model B10-3054B-13MS gate valves with Limitorque SMB-000-5 operators.

The Crosby 4K26 safety valves installed at H. B. Robinson 2 were not specifically tested in the EPRI test program. However, all Crosby valves are of the same basic design except for the orifice size and the inlet and outlet flanges. Other differences between valves discussed in Reference 5 include seat materials, disc holder construction, and valve body construction (cast or forged). EPRI tested three Crosby valves: 3K6, 6M6, and 6N8. The Robinson safety valves, with the K2 orifice, fall between the 3K6 and the 6M6 test valves with respect to orifice size. The 3K6 and 6M6 valves were used by CP&L in evaluating the operability of the 4K26 valves installed at the plant. In Table 4.1.1 the Robinson valves are compared with the test valves in the areas discussed above. As shown in Table 4.1.1, the Robinson valves differ from the 3K6 test valve only in the orifice size and the inlet and and outlet size. These differences only affect the valve capacity, not valve operability, and therefore the 3K6 test results can be used to evaluate the Robinson 2 4K26 valves. In addition to inlet/outlet size and orifice size differences, the Robinson 4K26 valves and the 6M6 test valve also differ in disc holder and valve body construction. As before, the differences in orifice size and inlet/outlet size only affect valve capacity not operability and the differences in disc holder and valve body construction will not affect valve operability. Thus, the 6M6 test results can be used to demonstrate operability of the Robinson 4K26 safety valves.

TABLE 4.1.1 COMPARISON OF H. B. ROBINSON UNIT 2 SAFETY VALVES WITH EPRI TEST VALVES

	4K26[1]	3K6[2]	6M6[2]
Orifice size	K2/1.800 in.	K/1.531 in.	M/2.154
Seat material[3]	Stellite	Stellite	Stellite
Disc holder construction	Type 1[4]	Type 1[4]	Type 2[5]
Valve body construction	Cast	Cast	Forged
Inlet size/rating	4/1500#	3/1500#	6/1500#
Outlet size/rating	6/600#	6/600#	6/600#

1. Valve installed at H. B. Robinson 2.

2. EPRI test valve.

3. Stellite seats are used for all Crosby valves installed with loop seals. EPRI tests of the 3K6 and 6M6 valves with loop seals used the stellite seats.

4. Type 1 construction: 347 stainless steel disc holder with stellite lands and disc bushing.

5. Type 2 construction: ASME SA637 Grade 718 disc holder without disc bushing.

The Copes-Vulcan PORVs used at H. B. Robinson 2 are essentially the same as the Copes-Vulcan test valve. The Robinson 2 PORVs have a 2 in. NPS body with a 17-4PH plug and cage. The test valve used the 17-4PH plug and cage, but the valve body was 3 in. NPS. The difference is body size will not affect operability and, thus, the test valve adequately represents the plant valve.

The block valve used at Robinson 2, a Velan model B10-3054B-13MS, was one of the valves tested by EPRI. The valve was designed to be installed in a vertical or horizontal piping configuration. The valve is installed at the plant and was tested by ERPI in a horizontal configuration. The plant valves have a Limitorque SMB-000-5 motor operator while the test valve used a Limitorque SB-00-15. The CP&L submittal (Reference 12) stated that the SMB-000-5 operator is used with the plant valve because the required closing time for the valve is 40 s compared to the 10 s closing time required by the test program. According to the CP&L block valve submittal, the SMB-000-5 operator has sufficient thrust to close the valve. With a torque switch setting of 1.7, the SB-00-15 operator used in the EPRI tests has an output torque of 150.5 ft-lb and a thrust of 9678 lb. The SMB-000-5 operator used at the plant, with a torque switch setting of 3, has an output of torque of 109 ft-lb and a thrust of 11645 lb. This thrust exceeds that generated by the SB-00-15 operator in the EPRI tests and the EPRI tests, therefore, conservatively bound the Robinson 2 valve response.

Based on the above, the valves tested are considered to be applicable to the in-plant valves at H. B. Robinson 2 and to have fulfilled the criteria of Items 1 and 7 of Section 1.2 regarding applicability of the test 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 cold overpressurization events. The valve inlet conditions applicable to H. B. Robinson 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 (locked rotor transient) and a maximum pressurization rate of 216 psi/sec (locked rotor transient). CP&L has enclosed each loop seal in an insulated box to raise the loop seal temperature to 300°F. The maximum developed backpressure at the safety valve is 350 psia.

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, four are applicable to H. B. Robinson 2 because the valve ring setting in these tests were representative of the plant valve ring settings. In these tests the peak pressure ranged from 2520 to 2726 psia and the pressurization rate ranged from 3.2 to 360 psi/sec. Valve inlet temperatures range from 90 to 350°F. These test conditions bound the plant valve conditions identified above.

The Crosby HB-BP-86 3K6 safety valve was subject to eight tests with a long inlet configuration in the EPRI tests. These tests will be used to demonstrate operability of the Robinson 2 valves. The eight tests with the 3K6 valve include tests with drained loop seals, tests with cold loop seals and factory ring settings, and tests with cold loop seals and lowered ring settings. In these tests, the peak pressure ranged from 2436 to 2708 psia and the pressurization rate ranged from 3.4 to 222 psi/sec. These test conditions bound those identified above for the plant valves. Therefore, the test inlet fluid conditions for the loop-seal steam tests were representative of the expected conditions for FSAR transients resulting in steam discharge from the safety valves.

The relevant data for the tests with the 3K6 and 6M6 test valves and the expected conditions for the 4K26 plant valves are presented in Table 4.2.1.

TABLE 4.2.1 SUMMARY OF TEST DATA FOR CROSBY 3K6 AND 6M6 SAFETY VALVES AND COMPARISON WITH 4K26 REQUIREMENTS

Valve	Test Number	Test Type	Inlet Conditions	Initial Fluid Temperature at Valve Inlet (°F)	Safety Valve Ring Settings	Pressure[1] at Valve Opening (psia)	Peak Tank Pressure (psia)	Peak Back-pressure (psia)	Percent Blowdown	Pressurization Rate (psi/s)	Valve Stability	Inlet Pressure Drop Opening (psi)	Inlet Pressure Drop Closing (psi)	Comments
3K6--Steam internals	506	Steam	Drained loop seal	Sat	-55, -14	2708	2709	455	6.8	4.1	Flutter	391	194	See Note 2
	508	Steam	Drained loop seal	Sat	-55, -14	2507	2508	515	--	2.6	Chatter	391	194	See Notes 2,3
	516	Steam	Drained loop seal	Sat	-115, -14	2435	2436	507	15.9	222	Stable	391	194	See Note 4
	517	Steam	Drained loop seal	Sat	-115, -14	2465	2725	582	15.6	3.4	Stable	391	194	See Note 4
3K6--Loop seal internals	525	Steam	Cold loop seal	110	-115, -14	2536	2558	471	18.8	3.4	Stable	391	194	See Note 4
	526	Steam	Cold loop seal	94	-115, -14	2608	2708	513	18.9	220	Stable	391	194	See Note 4
	529	Steam	Cold loop seal	86	-115, -14	2602	2638	480	17.7	13.3	Stable	391	194	See Note 4
	536	Steam	Cold loop seal	98	-115, -14	2637	2650	507	20.1	43.6	Stable	391	194	See Note 4
4K26	FSAR steam tran-sient		Hot loop seal	300	See Note 2	2485	2692	350	Nominally 5%	216	--	362	180	
6M6--Loop seal internals	929	Steam	Cold loop seal	90	-71, -18	2600	2726	710	5.1	319	Stable	263	181	See Note 2
	1415	Steam	Hot loop seal	290	-77, -18	2555	2760	255	6.2	360	Stable	263	181	See Note 2
	1419	Steam	Hot loop seal	350	-77, -18	2464	2675	245	--	360	Chatter	263	181	See Notes 2,3
	931a	Steam/water transition	Cold loop seal	117	-71, -18	2570	2578	725	12.7	2.5	Stable	263	181	See Note 2

1. Set pressure = 2485 psig.
2. Factory recommended ring setting.
3. Test terminated because of valve chatter.
4. Ring settings adjusted to increase blowdown.

For FSAR transients resulting in steam discharge, the PORV will be subject to a peak pressure of 2555 psia (locked rotor transient) and a maximum pressurization rate of 200 psi/sec (locked rotor) when both the relief and safety valve actuate. The Copes-Vulcan D-100-160 PORV with 17-4PH plug and cage was subjected to thirteen steam tests and one loop seal simulation test. In the steam tests the pressure at valve opening ranged from 2430 to 2745 psia and backpressures reached 615 psia. The loop seal simulation test was conducted at a pressure of 2715 psia and a backpressure of 618 psia. The Copes-Vulcan PORV is an air operated valve which is not considered to be sensitive to backpressure (Reference 6). Thus plant specific and test backpressures need not be compared to justify applicability of the test results. These considerations indicate the test fluid inlet conditions in the steam and loop seal tests on the Copes-Vulcan PORV were representative of FSAR transients where the valve passes steam.

The main feedline break accident causes an overcooling transient, and does not challange the safety valves or PORVs. Therefore, The main steamline break accident does not have to be considered in this evaluation (See Reference 18 dated June 14, 1985).

The limiting Extended High Pressure Injection (HPI) event was a spurious activation of the safety injection system at power (Reference 7). The H. B. Robinson 2 HPI pumps, however, have a cutoff head of 1500 psi and therefore no liquid flow through the safety valves or PORVs is expected (References 11 and 16). With a HPI pump cutoff head of 1500 psi, this event would not challenge the PORVs or safety valves. Therefore, conditions for this event need not be considered in demonstrating the operability of the Robinson 2 valves.

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 ranges of potential fluid conditions for low temperature overpressure events are presented in References 11 and 16. In addition to the high pressure loop seal and steam tests previously mentioned, the relief valve was subjected to two low

pressure water tests and a high pressure transition test. The high pressure transition test was at a pressure of 2545 psia with fluid conditions varying from saturated steam to saturated water. The low pressure water conditions were 675 psia at temperatures of 441°F and 109°F. These test conditions, together with the test conditions from the high pressure steam and loop seal testing, sufficiently encompass the range of fluid conditions for cold overpressure events at H. B. Robinson 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 2490 psia). Friction testing done by Westinghouse on stellite coated specimens (the Velan valve has stellite seats) showed that in the span of 21 test cycles (as applied in tests at the Marshall steam station) the thrust required to cycle the valve during steam tests would be similar to that required if the test medium were water. In addition, the required torque to open or close the valve depends almost entirely on the differential pressure across the valve disk and is rather insensitive to the momentum loading and, therefore, is nearly the same for water or steam and nearly independent of the flow. The full pressure steam tests, therefore, are adequate to demonstrate operability of the valve for low pressure steam and the required water conditions.

The presentation above, demonstrating that the test conditions bounded the conditions for the plant valves verifies 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 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 have been met.

4.3 Operability

As discussed in the previous section the safety valves are required to operate with full pressure steam, while the PORVs 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.

Demonstration of the operability of the Crosby 4K26 safety valves installed at H. B. Robinson 2 was based on the performance of the Crosby 3K6 and 6M6 valves during the EPRI tests. The safety valves at Robinson 2 are required to operate on loop seal to steam inlet conditions since transition to liquid has been ruled out by References 11 and 16. H. B. Robinson 2 has installed an insulated box around each safety valve loop seal in the plant to raise the loop seal temperature to approximately 300°F. Therefore, tests with hot, cold, or drained loop seals are applicable to evaluating the Robinson 2 safety valves.

The results presented in Table 4.2.1 show that when the Crosby 3K6 safety valve is mounted on the long inlet piping, it is only marginally stable with the factory recommended ring settings. During test 506, the valve fluttered; while during test 508 the valve chattered when closing, and the test had to be terminated. It is very important to note that the computed inlet piping pressure drop when the Robinson 2 4K26 valve opens (362 psi) is less than the pressure drop for the 3K6 test valve (391 psi), and the pressure rise when closing (180 psi) is less than the pressure rise for the 3K6 test valve (194 psi). The lower pressure changes for the plant valves than for the test valve indicate more stable behavior.

Loop seal-steam tests on the 3K6 valve were also performed with cold loop seals (initial water temperature approximately 100°F), lower ring settings, and backpressures which bound the maximum predicted values for the plant. These tests are applicable to the Robinson 2 valves because they give a conservative estimate of the Robinson 2 valve performance. EPRI

testing found colder loop seal temperatures usually resulted in less stable valve performance. With the cold loop seals the 3K6 valve opened at a pressure ranging from 2536 to 2637 psia (+1.4% to +5.5% of the set pressure). During loop seal discharge the valve fluttered or chattered before stabilizing on steam. Overall, the 3K6 valve gave acceptable performance during these tests although, with the colder loop seal, it tended to open at pressures outside the $\leq$3% range. One of these tests was run with a pressurization rate of 220 psi/sec, which indicates the plant valve will operate acceptably at the expected pressurization rate (216 psi/sec). Rated lift was achieved at 3% accumulation in all tests; however, rated flow was achieved at 3% accumulation in only one test. In the two tests where the appropriate measurement conditions were met at 6% accumulation, rated flow was achieved.

During testing the 3K6 valve was subject to bending moments on the valve discharge flange as great as 147,500 in.-lb without impairing valve operability. This conservatively bounds the maximum expected bending moment of 97,755 in.-lb at H. B. Robinson 2.

Valve blowdown during the testing of the 3K6 valve always exceeded the 5% design blowdown. Blowdowns ranging from 6.8% to 20.1% were measured. A Westinghouse Owners Group analysis of blowdowns in excess of 10% has been completed which showed no adverse effects on plant safety due to increased blowdown (Reference 19). In addition, the pressurizer did not fill during these analyses so that liquid discharge was not calculated. Thus, the increased blowdown observed in the tests is not a problem with respect to safety.

EPRI testing of the Crosby 6M6 valve was also used to qualify the Robinson 2 safety valves. During test 1415 (which had representative ring settings and a hot loop seal) the valve opened within $\pm$3% of the set pressure, fluttered or chattered in a partial lift position during loop seal discharge and then stabilized on steam. The valve closed with 6.2% blowdown in one test. In the second test (Test 1419), the valve chattered on closure and it was manually opened to terminate the test. The results of tests 1415 and 1419 indicate that inspection and maintenance are important to the continued operability of this valve. The computed inlet pressure drop

(263 psi) for the 6M6 mounted on the long inlet pipe was lower than for the 3K6 test valve or for the Robinson 2 4K26 valve.

During the applicable 6M6 tests, pressurization rates of 360 psi/sec were used which bound the expected plant conditions; however, the maximum backpressure was 255 psia which is less than the expected plant backpressure of 350 psia. Since the maximum backpressure is not developed until after the loop seal has been discharged and full steam flow conditions achieved, a cold water loop seal test (Test 929), with a peak backpressure of 710 psia, will be used to demonstrate valve operability with respect to backpressure. The pressurization rate during Test 929 was 319 psi/sec which also bounds the plant response. In this test the valve had stable performance on steam and closed with 5.1% blowdown adequately demonstrating valve operability at high backpressures.

Bending moments as high as 268,875 in.-lb were applied to the valve discharge flange without impairing valve operability. This conservatively bounds the predicted plant value of 97,755 in.-lb. During the two applicable tests, the 6M6 valve did not achieve rated lift at 3% or 6% accumulation. Flows in excess of rated flow, however, were achieved at 3% accumulation.

Valve blowdown of 6.2% was measured in one of the applicable 6M6 tests. Blowdown in the second test was not measured because the test was manually terminated by opening the valve to stop it from chattering. Blowdown of 6.2% is only slightly greater than the 5% design blowdown. As discussed for the 3K6 valve, blowdowns in excess of 10% have been shown to have no adverse affect on plant safety and do not result in the pressurizer filling with water.

To summarize , the results of the tests on the 3K6 and 6M6 safety valves demonstrate operability of the 3K26 safety valve. The valves opened within ±3% of the set pressure, except for the 3K6 valves with cold loop seals. Valve performance was stable on steam at pressurization rates and backpressures representative of plant conditions. During loop-seal discharge the valve fluttered or chattered before popping open on steam which is typical of valves with loop seals. Valve blowdowns were generally

in excess of the design value of 5%, however this has not been shown to be a problem.

For all tests on the PORV, the valve fully opened and closed on demand. Disassembly after testing showed the cage to body gasket had partially washed out but no damage was found that would affect valve operability. During one steam test the valve was subject to a bending moment of 43,000 in.-lb and valve performance was not affected. The largest predicted bending moment on the Robinson 2 relief valves is 35,703 in.-lb. The test data, therefore, bound the expected plant response.

Since the PORV performed satisfactorily during all tests and since the full range of expected inlet fluid conditions in the plant was represented by the test conditions, the demonstration of relief valve operability is considered adequate.

The PORV block valve must be capable of closing over a range of steam and water conditions. As described in Section 4.2, high pressure steam tests are adequate to bound operation over the full range of inlet conditions and as described in Section 4.1, the tests with the Velan valve and the SMB-000-15 operator conservatively demonstrate the operability of the plant valve. The test valve was cycled successfully at full steam pressure with full flow. It was shown to open and close successfully with torques as low as 82 ft-lbs (Reference 9). The plant valve operator is set to produce a torque of 109 ft-lbs (Reference 12) and, therefore, the tests are considered to adequately demonstrate acceptable block valve operation.

NUREG-0737, Item II.D.1 requires qualification of associated control circuitry as part of the safety/relief valve qualification. For Robinson 2, the licensee has determined that there are no accidents which expose the PORVs to a harsh environment during which they are required to operate, nor will any failure prevent the satisfactory accomplishment of required safety functions. Therefore the qualification requirements in NUREG-0737, Item II.D.1 are considered to have been met.

The presentation above, demonstrating that the valves operated satisfactorily, verifies that the portion of Item 1 of Section 1.2 that requires conducting tests to qualify the valves and that part of Item 7 requiring that the effect of discharge piping on operability be considered have been met.

4.4 Piping and Support Evaluation

In the piping and support evaluation, the safety/relief valve piping between the pressurizer nozzles and pressurizer relief tank was analyzed for requirements of the ANSI B31.1-1967 Code. 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 EPRI piping subcommittee recommendations as presented in Reference 20.

The safety and relief valve discharge loads were calculated for the fluid transient condition that will produce the most severe loading on the piping system. This occurs during a high pressure steam transient where steam from the pressurizer forces the water in the water seal through the safety or relief valve down the piping system to the relief tank. Cold loop seals were assumed upstream of the PORVs and hot loop seals were assumed upstream of the SRVs. The temperature distribution for the hot loop seal case was assumed to be consistent with EPRI Test 917. The temperature of the SRV loop seals was measured subsequent to the piping analysis and found to be colder than assumed. To ensure the loop seal temperatures are consistent with those assumed in the analysis each SRV loop seal was enclosed in an insulated box.

The computer code ITCHVALVE was used to perform the thermal hydraulic analysis for this transient. The unbalanced fluid forces for each straight segment of piping were calculated using the program FORFUN. The code ITCHVALVE uses the Method of Characteristics approach to generate fluid parameters as a function of time. The code FORFUN calculates fluid forces from the fluid parameters using the momentum balance equation. The capability of these two programs to calculate accurate fluid force time

histories for loop seal discharges was demonstrated by comparing calculated force histories using ITCHVALVE and FORFUN with force histories generated from tests conducted by EPRI at the Combustion Engineering facility (Reference 14). Reasonable comparisons were obtained.

To perform the plant specific analysis, thermal hydraulic models of the safety and relief valve piping from the pressurizer to the relief tank were developed. Based on the response from other Westinghouse plants (Reference 22) who also had Westinghouse perform the piping evaluation, it was concluded the plant specific analysis was conducted using the same approach to selecting node spacing, time step size and modeling the water slug as was used in the comparison analysis discussed above. As noted above, with this approach a good comparison between calculated and test data was achieved.

Reference 22 stated that in the case of choking at the valve, the velocity at the valve orifice area was set at the sonic velocity. Upstream and downstream boundary conditions were iteratively set to conserve mass and energy. Choked flow was internally checked to assure that the proper formulation was applied.

To account for uncertainties in valve flow rates, the flow rate in the piping analysis was conservatively adjusted. A conservative factor of 1.20 was included in the maximum rated valve mass flow rate for the PORVs and SRVs. The conservative valve flow rates used in the analysis acceptably account for 10% ASME derating and potential error in the flow rate.

Two valve opening cases were addressed in the analysis. One was a simultaneous opening of the three safety valves, the other a simultaneous opening of the two relief valves. This approach is reasonable since the three safety valves are identical and have the same set pressure. Likewise, the relief valves are identical and have an equal set pressure. Maximum forces in the common header region of the piping system could theoretically be expected when the opening sequencing is such that the initial pressure

waves from valve opening reach the common junction downstream simultaneously. This event is unlikely, however, because the valves would be required to open at times perfectly spaced to compensate for differing piping lengths leading to the common junction. More importantly, based on the response in Reference 22, the common region is sufficiently isolated from the valves and pressurizer by a significant amount of piping and dynamic supports between the valve outlets and common junction that valve operability and nozzle loading would not be significantly affected. Thus, the assumption of simultaneous valve openings is acceptable.

The structural analysis was performed using the WESTDYN computer program. The piping deflection solution for static loads was obtained by the transfer matrix method. In this method, transfer matrices for all pipe sections were defined, then an overall stiffness matrix was developed. The dynamic analysis was performed by modifying the model used in the static analysis to include mass characteristics of the piping and equipment. The analysis for safety/relief valve discharge loading was performed using the model for the dynamic analysis with some modifications. Time-history forces determined by FORFUN were applied at piping system lumped mass points. The time-history piping displacement response was determined with the FIXFM3 program. Input to this program consisted of natural frequencies and normal modes that were determined with the WESTDYN program and applied forces from FORFUN. Results from the FIXFM3 analysis were then used as input to WESTDYN2 to determine internal forces and deflections at the ends of each pipe element. These results were in turn used as input to POSDYN2 to determine maximum forces, moments, and displacements at the ends of pipe elements and maximum loads for pipe supports. It should be noted the programs used in the structural analysis have previously been reviewed and approved by the NRC (Reference 23).

The Robinson 2 structural analysis was performed by Westinghouse with the same programs used in the Westinghouse structural analysis of Diablo Canyon Units 1 and 2. Based on the response from Pacific Gas and Electric to questions on the Diablo Canyon 1 and 2 structural analysis (Reference 21) it was concluded the Robinson 2 structural analysis was performed using the following methods and considerations. A system damping of 2% of critical was assumed, which is in accord with the damping specified by U.S.

Regulatory Guide 1.61. Lumped masses were spaced to ensure that all appropriate mode shapes were accurately represented. Supports were modeled with linear stiffness elements. The integration time-step was internally determined within the structural program based on convergence criteria for stable solutions. The largest time-step that can be so selected is 0.0001 seconds, which is sufficiently small for safety/relief valve discharges. The safety valve bonnet assemblies and relief valve actuators were modeled as extended masses, displaced from the pipe centerline. The valve stem diameter and thickness were represented in the model to obtain the appropriate frequencies. Potential axial extension caused by balancing forces at the ends of pipe segments were evaluated in the analysis. The effect of these forces was determined to be negligible relative to the unbalanced forces for this application.

Utilization of these methods in analyses of EPRI/CE tests resulted in good comparisons between calculated and measured test results. Maximum support and pipe loads compared well between analysis and test results.

Results of the structural analysis for the piping system indicate the calculated stresses for all applicable loads and load combinations meet acceptance criteria. Calculated stresses come nearest to allowable stresses in the piping upstream of the relief valve under emergency conditions (normal sustained loads plus safety valve discharge). The maximum calculated stress was 28.9 ksi compared to an allowable stress of 29.6 ksi. Reference 16 states the design load capacities of the system pipe supports were checked against the loads predicted by Westinghouse stress calculations. Some supports were found which could not accommodate the Westinghouse calculated loads. These supports were redesigned to accommodate the calculated loads and all required support modifications have been completed. The above analysis is considered to adequately qualify the plant specific piping of the support system.

The discussion above, demonstrating that a bounding loading case has been chosen for the piping evaluation verifies that Item 3 of Section 1.2 has been met. The analysis of the piping and support system verifies that Item 8 has also been met.

5. EVALUATION SUMMARY

The licensee for H. B. Robinson 2 has provided an acceptable response to the requirements of NUREG-0737, Item II.D.1. However, in order to reconfirm that General Design Criteria 14, 15, and 30 of Appendix A to 10 CFR 50 have been met with regard to the safety valves and PORVs, the licensee must adopt inspection and maintenance procedures following each lift of the safety valves involving discharge of water. The rationale for this conclusion is given below.

The licensee participated in the development and execution of an acceptable relief and safety valve test program to qualify the operability of prototypical valves and to demonstrate that their operation would not invalidate the integrity of the associated equipment and piping. The subsequent tests were successfully completed under operating conditions which, by analysis, bound the most probable maximum forces expected from anticipated design basis events. The test results showed that the valves tested functioned correctly and safely for all steam and water discharge events specified in the test program that were applicable to H. B. Robinson Unit 2 and that the pressure boundary component design criteria were not exceeded. Analysis and review of both the test results and the licensee justifications indicated the performance of the prototypical valves and piping can be extended to the in-plant valves and piping. The plant specific piping also has been shown by analysis to be acceptable. However, the results of the safety valve tests demonstrate the need for inspection and maintenance for reliable continued operability of the safety valve. The licensee must inspect the safety valve after each lift involving loop seal or water discharge and a formal procedure requiring the inspection must be developed and incorporated into the plant operating procedures.

The requirements of Item II.D.1 of NUREG-0737 (Items 1-8 in Paragraph 1.2) which ensure that the reactor primary coolant pressure boundary will have a low probability of abnormal leakage (General Design Criterion No. 14) will be considered met when the licensee for H. B. Robinson, Unit 2 formally adopts procedures requiring inspection and maintenance of the safety valves following each lift of the valves involving

discharge of water. In addition, the reactor primary coolant pressure boundary and its associated components (piping, valves, and supports) have been designed with a sufficient margin so that design conditions are not exceeded during relief/safety valve events (General Design Criterion No. 15).

Further, 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, meeting General Design Criterion No. 30.

6. REFERENCES

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3. Letter, D. P. Hoffman, Consumers Power Co. 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, December 1982.

6. EPRI PWR Safety and Relief Valve Test Program Test Condition Justification Report, EPRI NP-2460, December 1982.

7. Valve Inlet Fluid Conditions for Pressurizer Safety and Relief Valves in Westinghouse Plants, EPRI NP-2296, December 1982.

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

9. EPRI/Marshall Electric Motor Operated Block Valve, NP-2514-LD, July 1982.

10. D. G. Eisenhut, NRC, from P. W. Howe, "Relief and Safety Valve Test Requirements," June 1, 1982.

11. S. A. Varga, NRC, from S. R. Zimmerman, "Safety and Relief Valve Adequacy Report," November 4, 1982.

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18. S. A Varga, NRC, from S. R. Zimmerman, "NUREG 0737, Item II.D.1 - Performance Testing of Relief and Safety Valves," June 14, 1985

19. Safety Valve Contingency Analysis in Support of the EPRI S/RV Test Program--Volume 3: Westinghouse Systems, EPRI NP-2047-LD, October 1981

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21. Application of RELAP5/MOD1 for Calculation of Safety and Relief Valve Discharge Piping Hydrodynamic Loads, EPRI-2479, December 1982.

22. G. W. Knighton, NRC, from J. O. Schuyler, PG&E, "Response to Request for Additional Information--NUREG-0737, Item II.D.1, Diablo Canyon Units 1 and 2," January 23, 1984.

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ENCLOSURE 2

SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE

(SALP INPUT)

Licensee: Carolina Power and Light Company
Plant Name: H. B. Robinson, Unit 2
Reviewer Name: A. S. Masciantonio
Reviewer Branch: Engineering Branch
Functional Activity: Review of Licensee's Response to NUREG-0737, Item II.D.1, Performance Testing of Relief and Safety Valves

(1) Management Involvement In Assuring Quality

Licensee's management involvement appeared to be adequate.

Rating Category: 2

(2) Approach to Resolution of Technical Issues from a Safety Standpoint

Licensee's management appeared to have an adequate understanding of the technical issues.

Rating Category: 2

(3) Responsiveness to NRC Initiatives

Licensee's management provided timely responses and met deadlines.

Rating Category: 2

(4) Staffing

Meetings and other contacts with licensee were staffed with competent personnel.

Rating Category: 2

(5) Reporting and Analysis of Reportable Events

N/A

(6) Training and Qualification Effectiveness

N/A

(7) Enforcement History

N/A