May 31, 1989

Docket No. 50-400

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Mr. Lynn W. Eury
Executive Vice President
Power Supply
Carolina Power & Light Company
P. O. Box 1551
Raleigh, N. C. 27602

Dear Mr. Eury:

SUBJECT: EVALUATION OF CAROLINA POWER & LIGHT COMPANY'S SHEARON HARRIS, UNIT 1, PLANT SPECIFIC SUBMITTALS IN RESPONSE TO NUREG-0737, TMI ACTION PLAN REQUIREMENT, ITEM II.D.1 (TAC NO. 63565)

Your initial submission for the Harris plant specific response to NUREG-0737, "TMI Action Plan Requirement," Item II.D.1 was dated June 28, 1984. The NRC requested additional information from you in a letter dated May 12, 1986 and subsequent telephone conferences between Carolina Power & Light Company (CP&L) and the NRC's contractor, EG&G, Idaho. CP&L submitted information in letters dated June 6 , July 3, 1986, and September 2, 1987 and subsequent telephone conferences between yourselves and the NRC's contractor, EG&G, Idaho.

The NRC staff, with the assistance of EG&G, Idaho, has completed the review of your plant specific response to TMI Item II.D.1. The results of that review are included with this letter in the enclosed Technical Evaluation Report from our contractor, EG&G, which the NRC staff endorses. Based on these results, the NRC staff has concluded that CP&L has provided an acceptable response with the exception of a current program to ensure operability of safety valves following any lift of the valves involving loop seal water or water (see Section 5 of enclosure 1). If, at present, you do not have such a program in place, we are requesting that you formulate and implement an acceptable program.

No further submissions are required on your part. Program acceptability will be determined as a normal part of our regional inspection program. Review activity is considered complete for this item.

Original Signed By:

Richard A. Becker, Project Manager
Project Directorate II-1
Division of Reactor Projects -I/II
Office of Nuclear Reactor Regulation

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As Stated

cc w/encl:
See next page

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Mr. L. W. Eury
Carolina Power & Light Company

Shearon Harris

cc:

Mr. R. E. Jones, General Counsel
Carolina Power & Light Company
P. O. Box 1551
Raleigh, North Carolina 27602

Mr. D. E. Hollar
Associate General Counsel
Carolina Power & Light Company
P. O. Box 1551
Raleigh, North Carolina 27602

Resident Inspector/Harris NPS
c/o U. S. Nuclear Regulatory Commission
Route 1, Box 315B
New Hill, North Carolina 27562

Mr. R. A. Watson
Vice President
Harris Nuclear Plant
P. O. Box 165
New Hill, North Carolina 27562

Mr. H. A. Cole
Special Deputy Attorney General
State of North Carolina
P. O. Box 629
Raleigh, North Carolina 27602

Nuclear Energy Public Information Committee
c/o Triangle J Council of Governments
P. O. Box 12276
Research Triangle Park, NC 27709

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta Street
Suite 2900
Atlanta, Georgia 30323

Mr. C. S. Hinnant
Plant General Manager
Harris Nuclear Plant
P. O. Box 165
New Hill, North Carolina 27562

Mr. Dayne H. Brown, Chief
Radiation Protection Section
Division of Facility Services
N. C. Department of Human Resources
701 Barbour Drive
Raleigh, North Carolina 27603-2008

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Enclosure 1

EGG-NTA-8029
Rev. 1

TECHNICAL EVALUATION REPORT
TMI ACTION--NUREG-0737 (II.D.1)
SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400

C. Y. Yuan
C. L. Nalezny
C. P. Fineman

June 1988

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

Prepared for the
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
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ABSTRACT

Light water reactor operators have experienced a number of occurrences of improper performance of safety and relief valves installed in their primary coolant systems. 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 provides the results of 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 Shearon Harris, Unit 1, Licensee response 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.

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

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TECHNICAL EVALUATION REPORT

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

SHEARON HARRIS NUCLEAR POWER PLANT

DOCKET NO. 50-400

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 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 General Design Criteria and NUREG Requirements

General Design Criteria 14, 15, and 30 require that (a) the reactor primary coolant pressure boundary be designed, fabricated, and tested so as to have an extremely low probability of abnormal leakage, (b) the reactor coolant system and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions are not exceeded during normal operation or anticipated transient events and (c) the components which are part of the reactor coolant pressure boundary shall be constructed to the highest quality standards practical.

To reconfirm the integrity of overpressure protection systems and thereby assure that the General Design Criteria are met, the NUREG-0578 position was issued as a requirement in a letter dated September 13, 1979 by the Division of Licensing (DL), Office of Nuclear Reactor Regulation (NRR), to ALL OPERATING NUCLEAR POWER PLANTS. This requirement has since been incorporated as Item II.D.1 of NUREG-0737, Clarification of TMI Action Plan Requirements (Reference 2), which was issued for implementation on October 31, 1980. As stated in the NUREG reports, each pressurized water reactor Licensee or Applicant shall:

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

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

3. Choose the single failures such that the dynamic forces on the safety 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. Carolina Power and Light Company (CP&L), the owner of Shearon Harris Plant, 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 are 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 the 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 Shearon Harris was designed by Westinghouse this report is relevant to this evaluation.

Several test series were sponsored by EPRI. PORVs and block valves were tested at the Duke Power Company Marshall Steam Station located in Terrell, North Carolina. Additional PORV tests were conducted at the Wyle Laboratories Test Facility located in Norco, California. Safety valves were tested at the Combustion Engineering Company, Kressinger Development

Laboratory located in Windsor, Connecticut. The results for the relief and safety valve tests are reported in Reference 8. The results for the block valves tests are reported in Reference 9.

The primary objective of the EPRI/C-E Valve Test Program was to test each of the various types of primary system valves in pressurized water reactor plant service 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

The plant specific evaluation of the adequacy of the overpressure protection system for the Shearon Harris Plant was submitted by the Carolina Power and Light Co. (CP&L) to the NRC on June 28, 1984 (Reference 11). Request for additional information was sent to CP&L by the NRC on May 12, 1986 (Reference 12), to which CP&L responded in two letters: the first dated June 6, 1986 (Reference 13), and the second dated July 3, 1986 (Reference 14). A submittal providing information on the analysis of the piping supports was made September 2, 1987 (Reference 15). Additional information on the peak pressure and flow rate used in the thermal hydraulic analysis of the PORV piping was provided in Reference 18.

The response of the overpressure protection system to Anticipated Transient 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

The Shearon Harris Nuclear Power Plant Unit 1 overpressure protection system is equipped with three (3) safety valves, three (3) PORVs, and three (3) block valves. The safety valves are 6-in. Crosby Model HB-BP-86, 6M6, spring loaded valves with loop seal internals. The Crosby safety valves have insulated loop seals upstream and a slug trap downstream to divert water when the loop seal clears. The design set pressure is 2485 psig and the rated steam capacity is 420,000 lbm/h. The PORVs are 3-in. Copes-Vulcan Model D-100 globe valves with 316 SS Stellited plug and 17-4PH cage and Model D-100-160 operator. The PORV design set pressure is 2,335 psig and the design flow capacity is 210,000 lbm/h. The PORV block valves are Westinghouse Model 3GM88 with Limitorque SB-00-15 operators. Insulated loop seals are used in the piping upstream of the PORVs.

A 6M6 safety valve identical to the model installed at Shearon Harris was tested in the EPRI safety valve and PORV testing program. Some of the EPRI tests were performed with typical PWR plant ring setting which are equivalent to the Shearon Harris safety valve ring settings. The tests from the EPRI test program can, therefore, be used to demonstrate operability of the Shearon Harris safety valve. PORV and PORV block valves identical to the in-plant valves were also tested in the EPRI program. Therefore, the valves tested by EPRI are representative of the Shearon Harris valves. Thus, the part of criteria of Item 1 and 7 as identified in Section 1.2 regarding applicability of the test valves are considered fulfilled.

4.2 Test Conditions

The Shearon Harris Unit 1 is a 3-loop pressurizer water reactor designed by the Westinghouse Electric Corp. The valve inlet fluid conditions that bound the overpressure transients for Westinghouse designed PWR plants are identified in Reference 7. The transients considered in this report include FSAR, extended high pressure injection, and low temperature overpressurization events. The expected fluid conditions for each of these events and the applicable EPRI tests are discussed in this section.

4.2.1 FSAR Steam Transients

The limiting event for the FSAR transients resulting in steam discharge through the safety valves and the simultaneous discharge through both the safety valves and PORVs is the locked rotor accident.

The safety valves are predicted to experience a peak pressure of 2592 psia and a maximum pressurization rate of 216 psia/s. The maximum developed back pressure in the outlet piping is 500 psig (Reference 11). The average loop seal temperature is 209.1°F.

In the EPRI tests, the Crosby 6M6 safety valve with ring settings equivalent to those of the in-plant safety valves was subjected to four loop seal steam tests (Test No. 929, 1406, 1415, 1419). The test valve was mounted on a long inlet pipe containing a loop seal. The valve ring settings were (-77, -18). The loop seal temperature ranged from 90 to 360°F. The valve initially opened at pressures ranging from 2464 to 2600 psia to clear loop seal water and then popped open on steam at pressures from 2674 to 2755 psia. The pressurization rates for the tests were 90 to 360 psi/s, and the back pressures were 245 to 710 psia. The above data, summarized in Table 4.2.1, show that the valve inlet fluid conditions and back pressures in these tests envelop the corresponding fluid data predicted for the Shearon Harris safety valves.

The PORVs are expected to open on steam at a pressure of 2350 psia. The maximum inlet pressure was predicted to be 2555 psia and the maximum pressurization rate was 200 psi/s. The Shearon Harris PORVs are installed with loop seals in the inlet piping.

Most of the EPRI tests on the Copes-Vulcan relief valve were performed without loop seals. One water seal simulation test, Test No. 70-CV-174-8W/W, was run with the Copes-Vulcan PORV and is included in

TABLE 4.2.1. SUMMARY OF TEST DATA FOR THE CROSBY 6M6 SAFETY VALVE AND COMPARISON WITH SHEARON HARRIS REQUIREMENTS

Valve	Test Number	Temperature at Valve Inlet (°F)	Pressure at Valve Inlet (psia)	Peak Pressure (psia)	Pressurization Rate (psi/s)	Percent Blowdown	Peak Back-pressure (psia)	Valve Stablilty
6M6	929	90	2600	2726	319	5.1	710	Stable
	1406	147	2530	2703	325	9.4	250	Stable
	1415	290	2555	2760	360	6.2	255	Stable
	1419	350	2464	2678	360	---	245	Chatter
Harris 1	----	-209	2485	2592	216	Nominally 5.0	500	-------

the evaluation. In the EPRI tests on the Copes-Vulcan PORV, the maximum steam pressure at valve opening ranged from 2450 to 2505 psia. The back pressure developed at the outlet of the PORVs is not an important consideration for this type of relief valve because the operation of air operated PORVs is not sensitive to back pressure (Reference 6). Therefore the EPRI test inlet fluid conditions for the PORV steam discharge events are representative of the plant specific transient conditions.

4.2.2 FSAR Liquid Transients

The FSAR transients resulting in liquid discharge through the safety valves are bounded by the main feedline line break accident. The maximum pressure at the safety valve inlet during liquid discharge is expected to be 2504 psia and the pressurization rate is 4.0 psi/s. Fluid temperature at the valve inlet ranges from 620.1 to 623.4^{o}F and the maximum liquid surge rate into the pressurizer is 313.7 gpm.

Two water discharge tests were performed with the 6M6 safety valve in the long inlet configuration and with typical PWR plant ring settings of (-77, -18). One of the water tests, Test No. 932a, was performed at an inlet temperature approximately 160^{o}F below that predicted for the in-plant valves. Thus, it is not directly applicable to the Shearon Harris safety valves. In the other test (Test No. 931b), the maximum inlet water pressure was 2475 psia. This is within 1% of the maximum predicted pressure at Shearon Harris. The pressurization rate was 2.5 psi/s and the inlet fluid temperature was 635^{o}F. These conditions are representative of the inlet fluid condition of the Shearon Harris safety valves.

The expected fluid conditions at the inlet of the safety valves, given above, was based on a Westinghouse analysis which assumed that the PORVs were not operable during the feedline break transient (Reference 6). 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, one test (Test No. 76-CV-316-2W) was performed with the Copes-Vulcan PORV (316 with stellite plug and 17-4 PH cage) for water discharge at high temperature. The maximum pressure and temperature at valve inlet was 2535 psia and 647°F. These compare favorably with the predicted maximum pressure and temperature of 2504 psia and 623.4°F for Shearon Harris. Therefore, this test is adequate to represent the in-plant PORV performance in the feedline break event.

4.2.3. Extended High Pressure Injection Event

The limiting extended high pressure injection event is the spurious activation of the safety injection system at power. The Westinghouse analysis (Reference 7) showed the safety valves were not challenged for this (3-loop) plant. The PORVs open with steam discharge followed by water flow. The maximum pressure predicted at the PORV inlet is 2352 psia with liquid temperatures ranging from 498 to 502°F. The pressurization rate ranges from 0 to 12 psi/s. Two water discharge tests (Test Nos. 73-CV-316-4W and 76-CV-316-2W) and one steam/water transition test (77-CV-316-7S/W) were performed with the Copes-Vulcan valve. These tests were performed at inlet pressures of 2532 to 2545 psia and temperatures ranging from 446 to 670°F. These tests are considered adequate to represent the PORV inlet conditions for the extended high pressure injection event.

4.2.4 Low Temperature Overpressurization Events

The potential fluid conditions for low temperature overpressurization events cover a wide range of pressure and temperature conditions and fluid states such as steam, water, and steam/water transition. Low temperature overpressurization transients do not challenge the safety valves at the Shearon Harris Plant (Reference 6), therefore only the operation of the PORVs and the block valves need to be considered. The tests representative of high pressure water, steam, and transition flow were previously discussed for the FSAR transients. For the low pressure water discharge condition, the inlet fluid conditions for Shearon Harris were provided in the submittal (Reference 11). The expected peak pressure at the PORV inlets is 505 psig, and the range of fluid temperatures is 85 to 450°F.

There were two low pressure water tests performed on the Copes-Vulcan PORV with Stellite plug and 17-4 PH cage. The tests were conducted at an inlet pressure of 675 psia and at temperatures of 105 and 442°F. These low pressure conditions together with the high pressure tests discussed in the preceding sections adequately envelop the expected inlet fluid conditions of the low temperature overpressurization events.

4.2.5 Block Valve Test Conditions

The block valves are required to operate over the same range of inlet fluid conditions as the PORVs. The Westinghouse 3MG88 block valve was only tested with full pressure steam (up to 2500 psia) in the EPRI test series. Since there were no tests performed for water discharge conditions, the operability of the block valve in water flow condition was not directly demonstrated by the tests. However, Westinghouse conducted an investigation on the opening and closing performance of a Westinghouse block valve of a similar type (Reference 16). Their tests showed that the required torque to open or close the gate valve depended almost entirely on the differential pressure across the valve disk and was rather insensitive to the momentum load. Thus, the required force for opening and closing the valve is nearly independent of the type of flow (i.e., water or steam). Further, according to the friction tests performed by Westinghouse on stellite coated specimen, the friction coefficient between the stellite surfaces is approximately the same under steam and water tests. In some instances, the friction force in water is lower than that in steam. Therefore, the full pressure steam tests are adequate to demonstrate the operability of the valve for the expected water conditions.

4.2.6 Test Conditions Summary

The test sequences and analyses described above, demonstrating that the test conditions bounded the conditions for the plant valves, 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

As discussed in Section 4.2, and summarized in Table 4.2.1, the representative EPRI steam discharge tests for the Shearon Harris safety valves were the loop seal tests on the Crosby 6M6 valve, Test No. 929, 1406, 1415, 1419. In all of these tests (except Test No. 1415), the valve fluttered or chattered during loop seal discharge and stabilized when steam flow started. The valve opened within $\pm$2% of the design set pressure and closed with 5.1 to 9.4% blowdown. Rated flow was achieved at 3% accumulation with valve lift positions at 92 to 94% of rated lift. These tests demonstrated that the initial chattering had no adverse effect on the effectiveness of the valve. The computed inlet pressure drop and rise when the Shearon Harris safety valve opens and closes are 260.4 and 154.3 psi respectively. The corresponding values for the test valve are 263 and 181 psi. Therefore, the plant valve should be as stable as the test valve.

The valve in Test 1415 performed stably, but in Test 1419 it did not perform very well. 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 Shearon Harris safety valves since an identical test (Test 1415) already demonstrated that the valve performed satisfactorily and exhibited no sign of instability. The closing chatter in Test 1419 may be a result of the repeated actuation of the valve in loop seal and water discharge tests. As shown in Table 4.3.1 on the next page, the 6M6 test valve was subjected to seventeen steam, water, and transition tests. In the first four or five tests, the valve fluttered and chattered during loop seal discharge but stabilized and closed successfully. After Test 913, there were four instances in which the test was terminated due to chattering on closing. Galled guiding surfaces and damaged internal parts were found during inspection and the damaged parts were refurbished or replaced before the next test started. The test results showed that the valve performed well after each repair, but the closing chatter recurred in a subsequent test. Test 1415 was performed immediately after valve maintenance and the valve

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	0.76	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.

performed stably. The next test (Test 1419) encountered chatter in closing even though it was a repeat of Test 1415 at similar fluid conditions. This suggests that inspection and maintenance are important to the continued operability of this valve. The Licensee should develop a formal procedure requiring that the safety valves be inspected after each actuation and the procedure should be incorporated into the plant operating procedures or licensing documents such as the plant technical specifications.

The blowdowns in the applicable tests (5.1 to 9.4%) were in excess of the 5% value specified by the valve manufacturer and the ASME Code. Westinghouse performed an analysis (Reference 17) on the effect of increased blowdown and concluded that blowdown of up to 10% had no significant effect on the plant safety. Therefore, the increased blowdown reported in the EPRI tests is acceptable for this plant.

The water discharge condition is represented by Test No. 931b. Test 931b followed a water to steam transition test (Test 931a). After the valve closed at the end of Test 931a, the system repressurized and the valve reopened on saturated water at 2475 psia. The valve chattered during opening but subsequently attained steady flow and closed at a pressure of 2380 psia (4.8% blowdown). Since the valve achieved steady flow and proper closure, the performance of the safety valve with water discharge is considered acceptable.

The maximum bending moment induced at the inlet flange of the safety valve during the EPRI tests was 286,800 in-lb and the valve performance was not affected. This bending moment is higher than the maximum bending moment of 248,000 in-lb calculated for the Shearon Harris safety valves (Reference 14). This indicates the predicted moment loading on the plant safety valves will not affect the valve operability.

4.3.2 Power Operated Relief Valve

In the EPRI tests applicable to the Shearon Harris PORVs, the valve opened and closed on demand. The opening and closing time were within the required opening and closing time of 2.0 s for Shearon Harris PORVs.

According to Reference 8 the slowest opening time was 1.85s and the slowest closing time was 1.65s. 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 Shearon Harris PORVs.

The actuation gas pressure for the PORV operator used in the EPRI tests at Wyle Laboratories was 86 ± 1 psig according to the test report (Reference 8). The normal and backup gas actuation pressures are 90 and 85 ± 5 psig, respectively, for the Shearon Harris PORVs (Reference 11). The small difference between the test and in-plant PORV actuation pressures will not affect valve performance.

The maximum bending moment induced on the discharge flange of the PORV during the EPRI tests was 43,000 in-lb. The operability of the valve was not affected by the applied load. The predicted maximum bending moment on the Shearon Harris PORVs is 30,970 in-lb which is lower than the bending moment tested. Thus, the predicted bending moment on the Shearon Harris PORVs will not affect valve operability.

NUREG-0737 II.D.1 requires qualifications of the associated circuitry as part of the safety and relief valve qualification task. The specific electric circuits under consideration are the control circuits of the PORVs. In Reference 14, CP&L stated that the PORVs are not assumed to open to mitigate the consequences of an accident and will perform their intended function of remaining shut during an accident due to their design of failing shut upon loss of power. The potential for spurious actuation was analyzed and control system failures were analyzed in response to IE Notice 79-22. This analysis was reviewed and approved by the NRC. In addition, the PORVs were qualified under the pump and valve operability program (PVORT), the actuation pressure transmitters are environmentally qualified, the cable is qualified (though not run as 1E), and the PIC cabinets are essentially the same hardware as the Class 1E cabinets. Therefore, it can be concluded that the PORV circuitry meets the qualification requirements of NUREG-0737, Item II.D.1.

4.3.3 Block Valves

The Westinghouse 3GM88 block valve was cycled 21 times against full steam flow at pressures of 2280 to 2420 psig. These pressures bound the predicted opening pressure of the Shearon Harris PORV of 2350 psia. The test valve fully opened and closed on demand. The stroke time ranged from 6.2 to 12.9 s.

During the initial testing of the Westinghouse 3GM88 block valve, valve leakage resulting from the incomplete closure of the block valve was observed. The torque switch setting of the actuator was increased and the valve closed fully in subsequent tests. Later study and valve tests conducted by Westinghouse (Reference 16) concluded that the closure problem was a result of under estimating the stem thrust required for full valve closure. Subsequent to the EPRI tests, the Licensee received a change request from Westinghouse requesting that the Shearon Harris block valve operator be rewired to close on position (Reference 11). With the modification completed, the Shearon Harris block valves are expected to close satisfactorily.

Tests for water flow with the Westinghouse block valves were not performed in the EPRI test program. As explained in Section 4.2, the valve behavior under the water flow condition is expected to be similar to that of the full pressure steam tests. Therefore, the operability of the valves for liquid flow condition was indirectly demonstrated.

4.3.4 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. In addition, the requirements of Item 5 regarding qualification of the PORV control circuits were satisfied.

4.4 Piping and Support Evaluations

This evaluation covers the stress analysis of the safety valve and PORV piping system extending from the pressurizer nozzle to the pressurizer relief tank. The piping was designed for deadweight, internal pressure, thermal expansion, earthquake, and safety/relief valve discharge conditions. The calculation of the hydraulic forces due to valve discharge, the method of structural analysis, and the load combinations and stress evaluations are discussed.

4.4.1 Thermal Hydraulic Analysis

The Shearon Harris safety valve and PORV piping was initially analyzed by Ebasco for the piping configuration without the slug diversion devices (Reference 11). A cold loop seal was assumed in the analysis and the analysis results indicated that unacceptably high piping stresses were developed in the downstream piping. In an effort to solve the overstress problem, Ebasco performed a number of additional analyses to study the effects of higher loop seal temperatures and modifications of the loop seal piping. According to the calculations made by Ebasco, if the loop seal pipes were installed with six inch insulation, the loop seal temperature could be raised to approximately 370°F. At this temperature, the piping stresses would be within the allowable limits. The insulation was installed accordingly. However, subsequent field measurements revealed that the average temperature of the loop seal with the six inch insulation was actually 209°F, not the 370°F expected by Ebasco. This indicated that the desired stress reduction was not achieved. Thus, an additional modification was implemented to eliminate the high fluid forces by adding a slug diversion device (slug trap) immediately downstream of the safety valve outlet. The slug trap diverts and holds the loop seal water as the safety valve simmers (before it pops open) and keeps it from traveling through the discharge piping. The loop seal water drains to the quench tank through a small diameter line from the slug trap. To ensure the adequacy of this modification, the Licensee engaged Westinghouse to perform a reevaluation of the safety valve piping. This section presents a review of the thermal hydraulic analysis performed by Ebasco and Westinghouse. Since the PORV piping analysis performed by Ebasco remains unchanged and Westinghouse utilized the appropriate portions of the forcing functions calculated by Ebasco, the Ebasco thermal hydraulic analysis is discussed first.

In the thermal hydraulic analysis, the pressurizer fluid conditions were selected such that the calculated piping 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, that is, all safety valves were assumed to actuate simultaneously with the PORVs closed and all PORVs were assumed to actuate simultaneously with the safety valves closed. Another case which simulated a more realistic operating condition in which the PORVs and safety valves actuated successively at their respective setpoints was also analyzed. The piping loads generated by this case were lower than the separate discharge conditions for the PORV and safety valves.

Both steam and water discharge conditions were considered. The piping loads resulting from water discharge were all lower than those associated with the water seal expulsion followed by steam discharge. In the final steam discharge case analyzed by Ebasco, the safety valve loop seal was assumed to have an average temperature of 367^{o}F. For water discharge, the water temperature at the valve inlet was predicted to be 620^{o}F. This is the water temperature calculated for the feedline break accident which is the limiting event for this plant (Reference 7). This temperature is considerably higher than the loop seal temperature. Therefore, more flashing is expected during water discharge and the resulting piping loads would be much lower. For the PORV piping, a high pressure low temperature overpressure event was analyzed. The piping loads for this case were found to be lower than the steam discharge loads also. Therefore the valve discharge conditions used to determine the piping and support stresses were the steam discharge conditions resulting from the simultaneous actuation of all safety valves and the simultaneous actuation of the PORVs respectively.

The limiting event resulting in steam discharge through the safety valves is the locked rotor accident. For the analysis of the safety valve transient, the pressurizer pressure was assumed to be 2559 psig with zero pressurization rate and saturated steam was assumed to be discharge through the valves. The above transient condition was analyzed for average loop seal temperatures of 194, 310, and 367°F and a drained loop seal, respectively. The limiting event resulting in steam discharge through the PORVs is the locked rotor accident also. The PORV analysis was performed for PORV actuation on saturated steam with a peak pressurizer pressure of 2592 psig (Reference 18).

The thermal hydraulic analysis was performed using the RELAP5 computer program and an Ebasco in house postprocessor CALPLOTFIII. RELAP5 calculates the thermal hydraulic properties of the fluid, such as pressure, temperature, and density at each control volume and at each junction, as a function of time. The CALPLOTFIII program uses the fluid parameters from the RELAP5 analysis to generate the force time histories for each piping segment. RELAP5 was shown to be a suitable tool for the prediction of piping discharge loads (Reference 19). The use of CALPLOTFIII in conjunction with RELAP5 to predict piping forces was verified by Ebasco using EPRI/CE safety valve discharge test results. A RELAP5 analysis was performed by Ebasco on EPRI Test 1411 using the same input data employed by EPRI in its application of RELAP5 for safety and relief valve discharge calculations (Reference 20). The calculated hydrodynamic conditions from RELAP5 were then used as input to CALPLOTFIII to calculate the fluid forces on the piping. A comparison of the CALPLOTFIII results with EPRI test results showed the calculated forces duplicated the measured forces. Plots of this comparison were provided in the Waterford, Unit 3, submittal (Reference 21).

The assumptions made and the key input parameters used in the thermal hydraulic analysis such as the valve opening time, valve flow area, the node spacing in the analysis model, and the calculation time step were reviewed and found to be acceptable. (The loop seal temperature distribution will be discussed later.) The analysis was based on a safety valve area of 0.0204 ft^2 which generated a flow rate of 499,320 lbm/h or 119% of the rated flow of the Crosby 6M6 valve. This flow rate is sufficient to account for the 10% safety valve derating required by the ASME Code plus an adequate margin for error. The maximum PORV flow rate calculated in the thermal hydraulic analysis was 262,500 lbm/h (Reference 18). This flow rate is representative of the flows measured in the EPRI tests for the Copes-Vulcan PORV used at Shearon Harris.

The safety valve piping was later reanalyzed by Westinghouse for the piping configuration incorporating the slug traps. For the modified safety valve piping, two valve discharge conditions need to be considered. One is the clearing of a cold water seal (209^{o}F) during steam discharge which generates the highest loads in the piping between the pressurizer nozzle and the slug trap. The other is water discharge through the safety valves which dominates the design of the rest of the downstream piping. Westinghouse treated the total load on the piping as a combination of the safety valve discharge load without slug traps and the thrust on the slug traps and its effect on adjacent piping during loop seal discharge. To avoid additional RELAP5 computer runs, Westinghouse utilized the hydrodynamic forcing functions previously generated by Ebasco to combine with the predicted fluid thrust on the slug traps. Ebasco had performed several RELAP5 calculations using different loop seal temperatures. The piping dynamic analysis was performed by selecting from the Ebasco analysis results the appropriate fluid force time histories which enveloped the highest loads on the upstream and downstream pipes. In addition, static loads were calculated by Westinghouse to represent the maximum loading on the slug traps and its connecting pipes when the loop seal water was diverted into the slug traps. These static forces were applied to the structural model to perform a static analysis to obtain the piping stresses and reactions. This is acceptable because the forces generated on the piping and slug traps when the loop seal water simmers through the safety valves is expected to be very small compared to the forces generated when the loop seal water is pushed through the discharge piping without a slug trap in the system. The static and dynamic structural analyses of the piping system will be discussed in the next section.

4.4.2 Structural Analysis

The dynamic responses of the piping system due to safety valve and PORV discharge transients were calculated using the modal superposition method. The fluid force time histories generated by Ebasco were used as forcing functions on the structure. The Westinghouse series of structural analysis programs, namely 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 Ebasco and the modal data from WESTDYN. The structural displacements were then used by WESTDYN2 to compute the piping internal loads and support reactions.

The WESTDYN series of structural programs mentioned above was previously reviewed and approved by the NRC (Reference 22). 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 23).

The piping upstream of the safety valves and PORVs was analyzed in accordance with requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB of the 1971 Edition and all editions and addenda through the Summer 1979 Addendum. The downstream piping was in compliance with the requirements of ANSI B31.1, Power Piping Code, 1973 Edition through Summer 1975 Addendum. The load combination equations and stress limits used for the evaluation of upstream and downstream piping were identical to those recommended by the Piping Subcommittee of the PWR Pressurizer Safety and Relief Valve Test Program (Reference 10). The piping stress summaries for the upstream, Class 1, piping contain the upset, faulted and fatigue stress evaluation results (Reference 13). All stresses are found to be within the allowable limits. The emergency condition stresses, which represent the normal plus safety valve discharge condition, were not provided in the submittal. However, the faulted condition stresses, which represent the normal plus SSE and safety valve discharge stresses are relatively low. They are not only within the faulted condition stress limits but also lower than the emergency condition allowables. It is, therefore, concluded that the emergency condition is covered by the faulted condition evaluation.

The pipe supports for the upstream and downstream piping were analyzed to meet the requirements of the ANSI B31.1 Code, 1973 Edition, supplemented by the Seventh Edition of the AISC Code (February 1969). References 14 and 15 presented the load combinations and allowable loads used for the piping support evaluation. These are, in general, consistent with those recommended by EPRI (Reference 10). One exception is discussed below. Reference 15 also presented a comparison of the calculated support loads with the corresponding allowable for the most critically stressed support components. This comparison showed the calculated loads were all less than the allowable loads. For all cases where the limiting load combination included a seismic load, the calculated loads were less than 1.33 times the AISC allowable (in fact, many met the basic AISC allowable without the 1.33 increase allowed for seismic loads). For those cases where the limiting load combination did not include a seismic load, the calculated loads were less than 1.0 times the AISC allowable.

A concern was raised about the definition of the emergency load combination presented in Reference 14. The emergency load combination used in the Shearon Harris support analysis was defined as the combination of the normal plus SSE plus PORV discharge. The emergency load combination in the EPRI guide, Reference 10, was defined as the combination of the normal plus safety valve discharge. The service limit for the emergency condition in the EPRI guide was level C. In the ASME code, a level C limit allows for a 50% increase in the normal allowable. The faulted load combination that was analyzed for Shearon Harris included normal plus SSE plus safety valve discharge loads. As noted above, all load combinations that included a seismic load met at least 1.33 times the basic AISC allowable. Because the basic AISC allowable is similar to the ASME normal allowable, this indicates the emergency load combination as defined by EPRI is bounded by the faulted load combination analyzed for Shearon Harris. This is acceptable.

According to the results of EPRI tests performed on the Crosby 6M6 safety valve, high frequency pressure oscillations of 170-260 Hz occurred in the piping upstream of the safety valve as a loop seal water slug passed through the valve. This raised a concern that these oscillations could potentially excite high frequency vibration modes in the inlet piping that could contribute to higher bending moments in the piping. This phenomenon was not accounted for in the structural analysis of the system. The piping between the pressurizer and safety valves in the EPRI tests, however, was composed of 8-in. Schedule 160 and 6-in. Schedule XX while that at Shearon Harris is 6-in. Schedule 160. Since the test piping did not sustain any discernible damage during pressure oscillations occurring in the tests, it is expected that the plant piping also would not incur damage during similar oscillations. Thus, a specific analysis for these pressure oscillations is not necessary for this plant.

4.4.3 Piping and Support Summary

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

5. EVALUATION SUMMARY

The Licensee for the Shearon Harris Nuclear Power Plant 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. However, the Licensee should develop a method to ensure continued operability of the safety valves following any lift of the valves involving discharge of loop seal water or water. The rationale for this conclusion is given below.

CP&L participated in the development and execution of an acceptable Relief and Safety Valve Test Program designed to qualify the operability of the 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 justifications indicated direct applicability of the prototypical valve and valve performances of the in-plant valves and systems intended to be covered by the generic test program. The plant-specific piping also was shown by analysis to be acceptable.

The test results demonstrated the need for inspection and maintenance of the safety valves following each lift involving loop seal or water discharge to ensure continued operability of the safety valves. The Licensee for Shearon Harris should develop a method to ensure continued operability of the safety valves such as formal procedures for inspection and maintenance of the safety valves following each valve actuation involving discharge of the loop seal or water.

The requirements of Item II.D.1 of NUREG-0737 (Items 1-8 in Paragraph 1.2) will be considered met when the Licensee develops a method to ensure continued operability of the safety valves that is acceptable to the NRC staff and, will thereby have 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). 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).

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