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## INFORMAL REPORT

TECHNICAL EVALUATION REPORT

TMI ACTION--NUREG-0737 (II.D.1)
RELIEF AND SAFETY VALVE TESTING
SURRY, UNITS 1 AND 2

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# DOCKET NO. 50-280 and 50-281

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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 has examined the response of the licensee for the Surry Units 1 and 2, to the requirements of NUREG-0578 and NUREG-0737 and finds that the Licensee has provided an acceptable response, reconfirming that the General Design Criteria 14, 15 and 30 of Appendix A to 10 CFR 50 have been met.

FIN No. A6492--Evaluation of OR Licensing Actions-NUREG-0737, II.D.1

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# 1. INTRODUCTION

## 1.1 Background

Light water reactor experience has included a number of instances of improper performance of relief and safety valves installed in the primary coolant 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 (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 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 power operated relief valves, safety valves, block valves and associated piping systems. Virginia Electric and Power Company (VEPCO), the owner of Surry Units 1 and 2, was one of the utilities sponsoring the EPRI Valve Test Program. The results of the program are contained in a group of reports which were transmitted to the NRC by Reference 3. The applicability of these reports 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 Surry 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 Kressinger Development Laboratory which is part of the Combustion Engineering Test Facility 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 Surry Units 1 and 2 was submitted by the Virginia Electric and Power Co. to the NRC on July 1, 1982 (Reference 11). Supplementary information on the testing and evaluation of the block valves was submitted on September 1, 1982 (Reference 13). Request for additional information was sent to VEPCO by the NRC on February 9, 1984 (Reference 14), to which VEPCO responded on October 31, 1984 (Reference 15). A second request for information was sent to VEPCO on August 13, 1985 (Reference 16), to which the licensee responded on October 31, 1985 (Reference 17) and on February 26, 1986 (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 Surry Power Station used three (3) safety valves, two (2) PORVs and two (2) block valves in the overpressure protection system in each of its two units. The safety valves are the 6-in. Crosby Model HP-BP-86, 6K26 with loop seal internals. The PORVs are 2-in. Copes-Vulcan Model D-100-160 (17-4PH plug and cage). The block valves installed in line with the PORVs are 3-in. Velan Model B10-354B-13MS with Limitorque SMB-00-15 operators. Loop seals are included in the inlet piping to the safety valves. There are no water seals upstream of the PORVs.

The Surry safety valve was not specifically tested by EPRI. Similar values which were used in the EPRI tests include the Crosby HP-BP-86, 3K6 and 6M6 safety valves. These valves are similar in design and operational characteristics to the plant-specific valve but vary in orifice size. A comparison of the valve sizes is shown below.

| Valve | Model | Inlet Diameter (in.) | Outlet Diameter (in.) | Nozzle Bore Diameter (in.) | Rated Flow |
|---|---|---|---|---|---|
| Surry | 6K26 | 6 | 6 | 1.800 | 293,330 |
| EPRI Test | 3K6 | 3 | 6 | 1.531 | 212,182 |
| EPRI Test | 6M6 | 6 | 6 | 2.154 | 420,006 |

The difference in orifice size only affects the valve capacity but not valve behavior. The valves were tested on long inlet piping configuration with loop seals similar to the Surry safety valve piping arrangement. There are some differences in the valve body construction (i.e., cast vs. forged bodies), and disc holder type and material, but such variations are not expected to affect the valve operability. Therefore the results of the EPRI tests on the 3K6 and 6M6 valves can be used to demonstrate the operability of the Surry safety valves.

The PORVs used in the EPRI tests were Copes-Vulcan relief valve Model D-100 with 17-4PH plug and cage (Reference 8). The test valve is the same as the PORVs used at Surry except the thickness of the valve body (Reference 5). The Surry PORV is an older version of the same model which has a 2-in. valve body instead of the newer 3-in. valve body tested. The 3-in. valve body tested would experience a larger thermal effect than the 2-in. valve body, thus it would be more than sufficient to verify the in-plant valve. The 17-4PH plug and the D-100-160 actuators are the same in both the in-plant valves and the test valve. Therefore the EPRI test results for the Copes-Vulcan PORV are entirely applicable to the Surry PORVs.

The Velan block valves and operators are identified in the letter from R. C. Youngdahl of Consumers Power to H. Denton of NRC (Reference 12) and the block valve submittal (Reference 13) as follows.

| | Model Number | |
|---|---|---|
| | Block Valve | Operator |
| Surry | B10-354B-13MS | SMB-00-15 |
| EPRI Tests | B10-3054B-13MS | SB-00-15 |

The model numbers of the Velan block valves and operators are slightly different between those installed at Surry and those tested. The licensee has compared the design drawings of the Velan B10-354B-13MS and B10-3054B-13MS valves and confirmed that they are similar valves (Reference 16). The Limitorque operator SB-00-15 is a variation of the standard Model SMB-00-15. The SM-00-15 operator has an improved stem nut design which reduces the valve rigidity in high speed (over 36 in./min) and high temperature (900°F and up) services. Both units have the same torque, thrust ratings and gear ratios. At the temperature and stem speed used in the EPRI tests, the stem nut design makes very little difference in the valve operations. Therefore, the Velan block valves and operators used in the tests and installed at Surry are functionally identical.

Based on the above, the valves tested are considered to be representative of the in-plant valves at Surry Units 1 and 2 and to have fulfilled the part of the criteria of Items 1 and 7 as identified in Section 1.2 regarding applicability of the test valves.

## 4.2 Test Conditions

The Surry Units 1 and 2 are 3-loop PWRs designed by the Westinghouse Electric Corp. The valve inlet fluid conditions that bound the overpressure transients for Westinghouse designed PWR plants are identified in Reference 7. The transients considered in this report include the FSAR transients and the extended high pressure injection and cold overpressurization events. The expected fluid conditions for each of these events and the applicable EPRI tests are discussed in this section.

### 4.2.1 FSAR Steam Transients

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 Surry outlet piping is 500 psig (Reference 11). The loop seal temperature is approximately 200$^{o}$F.

The Surry safety valves were set by a Crosby field representative (Reference 16). The Surry safety valve ring settings are (-200 to -270, -18), relative to the upper limit of ring travel. The EPRI tests used to evaluate the in-plant safety valves are therefore selected on the basis of factory recommended ring settings. Among the EPRI tests (Reference 8), there are two loop seal-steam tests (Test No. 929 and 1406) on the Crosby 6M6 safety valve with loop seal internals in the long inlet configuration that are applicable to the Surry safety valves. The upper and lower valve rings in these tests were set at (-71, -18) and (-77 -18) referenced to the bottom of the disk ring which represent typical PWR plant ring setting. In these tests, the valve initially opened at 2530 and 2600 psia to clear loop seal water and then popped open on steam at 2717 psia (the pop open pressure for the other test could not be measured). The pressurization rates were 319 and 325 psi/s, and the backpressures were 250 and 710 psia respectively. The loop seal temperature for these two tests were 90 and 147$^{o}$F respectively.

No tests were performed with the Crosby 3K6 safety valve which were comparable to the Surry safety valve with cold loop seal. However one test (Test No. 506) was performed with steam internals in the long inlet configuration and typical plant ring setting. This test can be compared with another test of the 6M6 safety valve (Test No. 1411) to determine the general behavior of the Crosby safety valves at typical plant settings.

Test No. 506 was performed with the loop seal drained and valve rings set at a typical PWR setting (-55, -14). The pressurization rate was low (4.1 psi/s) and the back pressure was 455 psia. The valve initially opened at 2708 psia and popped open at the same pressure. Blowdown was 6.8% and the valve fluttered before achieving full closure. Test 1411 was performed with the loop seal drained and with typical plant ring setting (-77, -18). The pressurization rate was 300 psi/s and back pressure was 250 psia. The valve initially opened at 2410 psia and popped open at 2420 psia. The valve behavior was stable and the blowdown was 8.2%.

The above comparison demonstrates that the 3K6 and 6M6 valves with typical valve ring settings had similar behavior during steam discharge and the blowdown was relatively low (maximum 8%). Thus the 6K26 in-plant safety valve which is similar to the 3K6 and 6M6 test valves should perform similarly. A summary of test data compared to the FSAR steam discharge inlet conditions is presented in TABLE 4.2.1.

The PORVs are expected to open on steam at a pressure of 2350 psia. The maximum pressurizer pressure is predicted to be 2555 psia and the maximum pressurization rate is predicted at 200 psi/s. In the EPRI test on the Copes-Vulcan relief valve, the maximum steam pressure at valve opening

TABLE 4.2.1. SUMMARY OF TEST DATA FOR CROSBY 3K6 AND 6M6 SAFETY VALVE AND SURRY -1 AND -2 REQUIREMENTS

| Test Number | Inlet Piping or Test Type | Ring Setting | Pressure at Valve Opening (psia) | Peak Tank Pressure (psia) | Peak Back-Pressure (psia) | Percent Blowdown | Pressurization Rate (psi/s) | Valve Stability | Inlet Pressure Drop (psid) |
|---|---|---|---|---|---|---|---|---|---|
| FSAR Steam Transients | | | | | | | | | |
| 929 (6M6) | Loop seal | -71,-18 | 2600 | 2726 | 710 | 5.1 | 319.0 | Stable | 181 |
| 1406 (6M6) | Loop seal | -77,-18 | 2530 | 2703 | 250 | 9.4 | 325.0 | Stable | 181 |
| 506 (3K6) | Long inlet | -55,-14 | 2708 | 2709 | 455 | 6.8 | 4.1 | Flutter | 391 |
| 1414 (6M6) | Long inlet | -77,-18 | 2410 | 2664 | 245 | 8.2 | 300.0 | Stable | 181 |
| Surry (6K26) | Loop seal | -200,-18[1]<br>-280,-18[1] | 2485 | 2592 | 500 | <14.0 | 300.0 | --- | 171 |
| FSAR Liquid Transients | | | | | | | | | |
| 931a(6M6)[2] | Transition | -71,-18 | 2570 | 2578 | 725 | 12.7 | 2.5 | Stable | 181 |
| 931b(6M6)[2] | Water | -71,-18 | 2475 | 2475 | 700 | 4.8 | 2.5 | Chatter | 181 |
| 431a(3K6)[3] | Water | -45,-14 | 2342 | 2349 | 584 | 13.0 | 1.8 | Stable | 391 |
| Surry(6K26) | Loop seal | -200,-18[1]<br>-280,-18[1] | 2485 | 2575 | 500 | <14 | 74.0 | --- | 171 |

1 Factory ring settings

2 Liquid discharge 2355 GPM with saturated liquid

3 Liquid discharge 1370 GPM with liquid temperatures between 622 and $631^{\circ}F$

ranged from 2430-2505 psia. The back pressure developed at the outlet of the PORVs is not an important consideration for this type of relief valves because the operation of the air operated PORVs is not sensitive to back pressure (Reference 6). Therefore, the EPRI test inlet fluid conditions for the PORV steam discharge cases are representative of the plant specific transient condition.

### 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. Surry is one of the older nuclear power plants which were licensed prior to the issuance of Regulatory Guide 1.70, Revision 2 and were not required to consider the feedline break accident as part of the design basis. However, in response to the requirements of NUREG-0737, VEPCO had the feedline break accident for Surry analysed using the RETRAN System Transient Analysis code (Reference 18). The maximum pressure at the safety valve inlet during liquid discharge is predicted to be 2575 psia and the maximum pressurization rate is 74 psi/s. The range of liquid liquid relief temperatures is 624 to 626$^{o}$F, and the maximum liquid surge rate into the pressurizer is 1146 GPM.

During the EPRI tests, the Crosby 6M6 safety valve was subjected to two tests representative of liquid discharge conditions. Test 931a was a loop-seal transition test, while 931b was a water test. The peak tank pressure was 2578 psia, the pressurization rate was 2.5 psia, the liquid teperature was 641$^{o}$F, and the peak liquid discharge was 2355 GPM. One test was performed on the Crosby 3K6 safety valve with representative liquid water conditions (Test 431a). The ring settings were slightly higher (Less negative) which would tend to make the valve less stable which is conservative. The valve opened at 2342 psia, the peak pressure was 2349, pressurization rate was 1.8 psi/s, liquid temperature was 622$^{o}$F at the valve inlet, and 631$^{o}$F in the tank. The maximum steady state liquid flow was 1370 GPM. A summary of test data and FSAR Liquid inlet conditions is presented in TABLE 4.2.1.

### 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) shows that there is no fluid discharge through the safety valves for this (3-loop) plant because the maximum head of the safety injection pumps is below the set pressure of the safety valves, and only PORV discharge will take place in this event. The PORV generally opens with steam which is then followed by water discharge. The maximum pressure predicted at the PORV inlet is 2350 psia with temperature between 498 to 502°F. The pressurization rate ranges from 0 to 12 psi/s. The EPRI tests on the Copes-Vulcan relief valve contain water tests and transition test performed at valve opening pressure of 2535 to 2545 psia and at temperatures from 455 to 647°F. These tests are considered adequate to represent the PORV inlet fluid condition for the extended high pressure injection event.

### 4.2.4 Low Temperature Overpressure Transients

The potential fluid condition for the low temperature overpressurization events covers a wide range of pressure and temperature conditions and fluid states such as steam, water and steam water transitions. Low temperature overpressurization transients do not challenge the safety valves at the Surry plant (Reference 6), therefore only the operation of the PORVs and the block valves need to be considered. The high pressure water, steam and transition flow were previously discussed in the FSAR transients. Therefore, the tests which bound the high pressure conditions will not be repeated here. For the low pressure water discharge cases, the inlet fluid conditions for Surry Units 1 and 2 are given in the submittal, Reference 11. The expected pressure and temperature at the PORV inlet are 435 psia and 100 to 350°F.

There were two low pressure water discharge tests performed on the Copes-Vulcan PORV. The tests were conducted at an inlet pressure of 675 psia and at water temperatures of 105 and 442°F. These low pressure conditions together with the high pressure tests adequately envelope the expected fluid conditions of the low temperature overpressure events.

The block valves are required to operate over the same range of inlet fluid conditions as the PORVs. However the Velan block valve was only tested for the full pressure (up to 2500 psia) steam conditions in the EPRI tests. The question as to whether the block valve will perform satisfactorily on water was not directly addressed. The Velan block valve is a gate valve with stellite coated disk and seats. The Westinghouse Electric Co. has conducted an investigation on the opening and closing performance of gate valves of the similar type. Their tests (Reference 17) 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. Therefore, the required force for opening and closing the valve is nearly independent of the type of flow (i.e., water or steam). Furthermore, 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 conditions. In some instances, the friction force in water is lower than that in steam. Accordingly, it would take equal or less force to overcome the disk friction in water than in steam. Therefore, the full pressure steam tests are adequate to demonstrate the operability of the valve for the expected water conditions.

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 have been met, in that conditions for the operational occurrences have been determined and the highest predicted pressures were chosen for the test. The part of Item 7, which requires showing that the test conditions are equivalent to conditions prescribed in the FSAR, is also met.

## 4.3 Operability

### 4.3.1 Safety Valves

Of the EPRI steam tests performed on the Crosby 3K6 and 6M6 safety valves, there were two tests with the 6M6 safety valve which closely bounded the Surry valve steam transient conditions. These steam tests

(No. 929 and 1406) were performed with a cold loop seal on the 6M6 test valve (See Table 4.2.1). The ring setting used on this valve was comparable to the ring settings of the Surry safety valves in that they were typical PWR ring settings. In these tests, valve flutter or chattering was observed during the loop seal water clearing phase but stable steam flow was achieved afterwards. Rated flow was attained at 3% accumulation but only 95 to 97% of rated lift was obtained. The short period of valve instability during the loop seal water discharge is not considered unacceptable since the rated flow and steady steam discharge was achieved. The blowdown rates for the tests were 5.1 and 9.4% respectively. The licensee stated that the ring settings used for Surry safety valves were developed during the Crosby production tests for each of the valves and produced measured blowdowns of 4.5 to 5%. Furthermore, the Westinghouse Owners Group has conducted a study on the effect of the increased blowdown and concluded that for the Westinghouse design plants, blowdowns of up to 14% had no significant effect on the outcome of the safety analysis (Reference 19). Therefore, it would still be tolerable even if the blowdown rates for the Surry safety valves approach the 9% level as reported for one of the tests.

During the the loop-seal transition test 931a, the Crosby 6M6 safety valve performed stably. The liquid temperature durng liquid discharge was 641$^{o}$F, and the valve discharged 2355 GPM. Test 931a was followed by Test 931b which was a water test during which the valve also performed stably. One test was performed on the Crosby 3K6 safety valve with representative liquid water conditions (Test 431a). As shown in TABLE 4.2.1, the inlet conditions were also similar to the predicted conditions for Surry 1 and 2. The results of the tests on the 3K6 and 6M6 valves are similar. The difference in discharge rate is due to orifice size and indicate that the plant 6K26 vlave will be able to discharge the required amount of liquid during the bounding FSAR Liquid Discharge Transient.

The highest bending moment induced at the inlet flange of the Crosby 6M6 safety valve during the tests was 286,800 in.-lb and the valve performance was not affected. This bending moment is much higher than the maximum bending moment of 98,938 in-lb calculated for the Surry safety valve (Reference 15). This indicates that moment loading on the inlet flange of the safety valve has no effect on the operability of the Surry safety valve.

EPRI testing of the Crosby 6M6 valve was used to qualify the Surry Units 1 and 2 safety valves (the valves are similar in design and operation), therefore, the general performance of the Crosby 6M6 safety valve must be considered in the the evaluation of operability of the Surry 6K26 safety valve. During test 1415 (which had representative ring settings and a hot loop seal) the valve opened within ±3% of the set pressure, fluttered or chattered during loop seal discharge, stabilized on steam, and closed with a 6.2% blowdown. In the next test (Test 1419), the valve chattered on closure and was manually opened to terminate the test. The results of tests 1415 and 1419 indicate that inspection and maintenance are important to the continued reliable operability of Crosby safety valves.

4.3.2 Power Operated Relief Valve

The EPRI tests applicable to the Surry PORVs indicated that the valves opened and closed on demand. The steam flow rate observed in the tests were between 255,000 and 265,000 lb/h which exceeded the rated flow of 210,000 lb/h for the Surry PORVs. The opening and closing time were within the required opening and closing time of 2.0 s for the Surry PORVs (with the exception of one steam test recorded at 2.10 s at the Marshall Test Facility).

At the end of the planned evaluation cycles of the Copes-Vulcan PORV at the Marshall Test Facility, the valve was subjected to additional cycles of tests including full pressure tests and unpressurized actuations or opening/closing performance tests. During five full pressure tests conducted at the end of this series, the valve failed to achieve full closure. (The valve could only close to within 13% of the fully closed position.) The licensee indicated that no design change was contemplated at present. Routine maintenance of the PORVs will uncover such leakage problems and remedy will be made at the time.

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 Surry PORV associated with the combined effect of dead weight, thermal and valve discharge loads is 19,602 in-lb (Reference 15). This indicates that maximum presicted moment loading on the inlet flange of the Surry PORV will not effect operability, and functionality of the Surry PORV.

On the basis of the valve performance stated above and the modification made on the valve actuation system, proper operation of the Surry PORVs under the predicted fluid conditions is expected.

4.3.3 Block Valves

The Velan block valve was subjected to 21 cycles of steam tests against full flow at 2340 to 2500 psig nominal line pressure. The valve opened and closed on demand and the stroke times were recorded at 9.7 to 9.9 s. The test pressures were above the Surry PORV opening pressure of 2335 psig and the stroke times were within the specified stroke time of 10 s for the SB-00-15 valve operator.

The actuator motor rpm for the Surry block valve differed slightly from that of the test valves. According to the licensee, Reference 15, the internal gearing of the Surry actuator was adjusted to provide proper valve stroke and valve thrust. Therefore, the in-plant Velan valves are expected to provide similar performance as the test valves.

Tests for water flow for both the Velan and the Westinghouse block valves were not performed in the EPRI test program. As explained in Section 4.2 of this report, 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 has been indirectly demonstrated.

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. The Nuclear Regulatory Commission staff has agreed that meeting the licensing requirements of 10 CFR 50.49 for this circuitry is satisfactory and that specific testing per NUREG-0737 requirement is not required. In the October 31, 1985 submittal (Reference 17), VEPCO stated that the PORV control circuitry for Surry Units 1 and 2 were included in the 10 CFR 50.49 environmental qualification program. Therefore, the qualification of the PORV circuitry is considered complete.

### 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 have been met. Also, the qualification of the PORV circuitry under 10 CFR 50.49 is considered to satisfy Item 5 of Section 1.2. However, the results of the tests on the Crosby Model 6M6 safety valve, which was used by VEPCO to qualify the Crosby 6K26 safety valves, indicate that inspection and maintenance are important to the continued reliable operability of the Surry safety valves.

## 4.4 Piping and Support Evaluations

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

Pressurizer fluid conditions were selected for use in the thermal hydraulic and stress analyses such that the calculated pipe discharge forces would bound the forces for any of the FSAR, HPI and low temperature overpressurization events, including the single failure that would maximize the forces on the valve. The loss of load transient was selected as the limiting condition for this analysis.

The thermal hydraulic analysis was performed using the Stone and Webster Engineering Corp. (SWEC) computer code--WATSLUG. WATSLUG calculates the forcing functions for each piping segment based on the fluid characteristics such as velocity, pressure and density during the water slug discharge event. The program was verified by comparing solutions of a test problem to RELAP/MOD1 analysis results for the same problem. Further check-out was performed by comparing the structural NUPIPE-SW generated piping support reactions due to the WATSLUG output forcing functions calculated for a given EPRI test with the measured reactions for the same test. Based on the verification data provided by the licensee (Reference 17), these computer codes and the modeling are considered adequate for the piping thermal hydraulic and stress analysis.

The safety valve and PORV discharge cases were analyzed as two separate events. This is reasonable since the safety valves and PORVs have different set pressures, and will not lift simultaneously. In each valve discharge event, the single and multiple valve discharge cases were considered. For the safety valve piping analysis where three individual valves were involved, the maximum load on a given piping segment was selected from the results of one, two and three safety valve discharge cases. In the analysis of a multiple valve discharge case, all valves were assumed to actuate simultaneously. The loading on some portions of the piping could be further maximized if the valve openings were sequenced in such a manner that the initial pressure waves from the loop seal discharges reached the common header downstream of the safety valves simultaneously. This was not assumed in the analysis since the resulting change in stresses would not be large enough to alter the outcome of the analysis and the precise phasing of the valve openings postulated for the maximum condition is not likely to occur in the actual situation.

The safety valve input data used in the thermal-hydraulic analysis did not explicitly account for the ASME code derating factor, however, the slug flow analysis performed by using WATSLUG code is considered to be conservative by more than the derating factor. After passing through the valve, the slug had a density equal to 6 $lbm/ft^3$ due to flashing based upon the energy in the 400$^{o}$F (average temperature) upstream of the valve. The downstream slugs are treated as square-edged slugs which is conservative, and the slugs from each safety valve are combined into a larger square-edged slug which is conservative. The overall conservatism of the analysis is demonstrated in the results presented in Figure 3A.3A.6 of Attachment A of Reference 17. Therefore, the thermal hydraulic analysis is considered adequate for predicting the safety/relief valve discharge loads.

The structural analysis of the safety/relief valve piping was performed using NUPIPE-SW computer code. The NUPIPE computer code is a linear elastic piping structural analysis program widely used in the industry. It has been shown to be a suitable tool for the static, dynamic and thermal analyses of a piping system. Additional verification of the program was provided in the comparison made against the EPRI test results discussed above.

For the analysis of the valve discharge conditions, the input forcing functions were obtained from the thermal hydraulic analysis discussed previously. These time varying fluid forces caused by the valve discharge excitations were applied to the structural model at appropriate node points. The dynamic analysis was performed employing the modal superposition method.

The piping analysis was performed in accordance with the requirements of the ASA B31-1 code for pressure piping 1955 edition. The load combinations and stress limits for the safety and relief valve discharge stresses are based on the plant UFSAR and the EPRI recommendations (Reference 10). The load combination equations and stress limits are given below.

$$S_{LP} + S_{DL} + \text{SRSS}\ (S_{OBEI} + S_{OCC1}) \leq 1.2\ S_h$$

$$S_{LP} + S_{DL} + \text{SRSS}\ (S_{DBEI} + S_{OCC2}) \leq 1.8\ S_h$$

Where:

$S_{LP}$ = Longitudinal pressure stress

$S_{DL}$ = Deadload stress

$S_{OBEI}$ = Seismic stress due to OBE-inertia

$S_{DBEI}$ = Seismic stress due to DBE-inertia

$S_{OCC1}$ = Occasional stress due to relief valve discharge case

$S_{OCC2}$ = Occasional stress due to larger of relief valve or safety valve discharge cases

$S_h$ = Allowable stress at maximum operating condition.

The above load combination equations are consistent with the load combinations suggested by EPRI in Reference 10. The second equation uses $1.8S_h$ as the allowable stress which corresponds to a level C (emergency) condition stress limit. This stress limit is more conservative than the criterion given in Reference 10 which treats this load combination as a level D (faulted) condition. Therefore, the load combinations and stress limits are considered acceptable. In the October 31, 1985 submittal (Reference 17), VEPCO provided a maximum stress level summary for the two Surry units. The summary showed that the maximum calculated stresses were within the allowables for all load cominations.

Similar load combination equations were used for the piping supports. The allowable stress for the combined load condition is 1.33 times the basic allowable stress of the material based on the AISC Manual of Steel Construction. The load combination equations and stress limits used for the piping supports are considered acceptable. The October 31, 1985 submittal (Reference 17) also presented a list of attributes for two representative restrains assoiated withe the PSRV system. The calculated stresses/loads for the various restraint components of the system were compared to their respective allowables. There allowables used are those specified in the AISC Manual for Steel Construction (7th Ed, 1970) and manufacturer allowable limits. The comparison indicates that the stresses/loads for these two restraints which are typical of the others within the PSRV system, are within allowable stress/load limits.

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

# 5. EVALUATION SUMMARY

The Licensee for the Surry Units 1 and 2 has provided an acceptable response to the requires of NUREG-0737, and thereby reconfirmed that the General Design Criteria 14, 15, and 30 of Appendix A to 10 CFR 50 have been met. The rationale for this conclusion is given below.

VEPCO 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. 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.

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

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

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13. ABSTRACT *(200 words or less)*

Light water reactor operators have experienced a number of occurences 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 has examined the response of the licensee for the Surry Units 1 and 2, to the requirements of NUREG-0578 and NUREG-0737 and finds that the Licensee has provided an acceptable response, reconfirming that the General Design Criteria 14, 15 and 30 of Apprndiz A to 10 CFR 50 have been met.

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