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Idaho National Engineering Laboratory

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INFORMAL REPORT

TECHNICAL EVALUATION REPORT

TMI ACTION--NUREG-0737 (II.D.1)
RELIEF AND SAFETY VALVE TESTING
TURKEY POINT UNITS 3 AND 4

C. Y. Yuan
G. K. Miller
C. L. Nalezny

EG&G Idaho

Work performed under DOE Contract No. DE-AC07-76ID01570

Prepared for the

U.S. NUCLEAR REGULATORY COMMISSION

DISCLAIMER

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TURKEY POINT UNITS 3 AND 4
DOCKET NOS. 50-250 AND 50-251

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Idaho National Engineering Laboratory
EG&G Idaho, Inc.
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ABSTRACT

Light water reactors have experienced a number of occurrences of improper performance of safety and relief valves installed in the primary coolant system. As a result, the authors of NUREG-0578 (TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations) and subsequently NUREG-0737 (Clarification of TMI Action Plan Requirements) recommended that programs be developed and completed which would reevaluate the functional performance capabilities of Pressurized Water Reactor (PWR) safety, relief, and block valves and which would verify the integrity of the piping systems for normal, transient, and accident conditions. This report documents the review of these programs by the Nuclear Regulatory Commission (NRC) and their consultant, EG&G Idaho, Inc. Specifically, this report documents the review of the Turkey Point Units 3 and 4 Licensee response to the requirements of NUREG-0578 and NUREG-0737. This review found the Licensee had not provided an acceptable response, which would reconfirm that 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 (1) the reactor primary coolant pressure boundary be designed, fabricated, and tested so as to have extremely low probability of abnormal leakage, (2) the reactor coolant system and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions are

not exceeded during normal operation or anticipated transient events, and (3) the components which are part of the reactor coolant pressure boundary shall be constructed to the highest quality standards practical.

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

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

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

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

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

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

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

7. Submit a correlation or other evidence to substantiate that the valves tested in a generic test program demonstrate the functionability of as-installed primary relief and safety valves. This correlation must show that the test conditions used are equivalent to expected operating and accident conditions as prescribed in the Final Safety Analysis Report (FSAR). The effect of as-built relief and safety valve discharge piping on valve operability must be considered.

8. Qualify the plant specific safety and relief valve piping and supports by comparing to test data and/or performing appropriate analysis.

2. PWR OWNER'S GROUP RELIEF AND SAFETY VALVE PROGRAM

In response to the NUREG requirements previously listed, a group of utilities with PWRs requested the assistance of the Electric Power Research Institute (EPRI) in developing and implementing a generic test program for pressurizer safety valves, power operated relief valves, block valves, and associated piping systems. The Florida Power & Light Co. (FPL), the owner of Turkey Point Units 3 and 4, was one of the utilities sponsoring the EPRI Valve Test Program. The results of the program, which are contained in a series of reports, were transmitted to the NRC by Reference 3. The applicability of these reports is discussed below.

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

EPRI contracted with Westinghouse Corporation to produce a report on the inlet fluid conditions for pressurizer safety and relief valves in Westinghouse designed plants (Reference 7). Since Turkey Point Units 3 and 4 were designed by Westinghouse, this report is relevant to this evaluation.

Several test series were sponsored by EPRI. PORVs and block valves were tested at the Duke Power Company Marshall Steam Station located in Terrell, North Carolina. Additional PORV tests were conducted at the Wyle Laboratories Test Facility located in Norco, California. Safety valves (SRVs) were tested at the Kressinger Development Laboratory which is part of the Combustion Engineering Test Facility located in Windsor, Connecticut. The results of the relief and safety valve tests are reported in Reference 8. The results of the block valve tests are reported in Reference 9.

The primary objective of the EPRI/C-E Valve Test Program was to test each of the various types of primary system safety valves used in PWRs for the full range of fluid conditions under which they may be required to operate. The conditions selected for test (based on analysis) were limited to steam, subcooled water and steam to water transition. Additional objectives were to (1) obtain valve capacity data, (2) assess hydraulic and structural effects of associated piping on valve operability, and (3) obtain piping response data that could ultimately be used for verifying analytical piping models.

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

3. PLANT SPECIFIC SUBMITTAL

A preliminary assessment of the adequacy of the Turkey Point, Units 3 and 4, overpressure protection systems was submitted by FPL on April 1, 1982 (Reference 11). Final evaluation reports covering the PORV, PORV Block Valve, Safety Valve and piping were submitted on July 9, August 13 and September 1 of 1982 (Reference 12, 13, and 14). A request for additional information was transmitted to FPL by the NRC on July 19, 1985 (Reference 15). FPL responded on June 26, 1986 (Reference 16).

The response of the overpressure protection system to Anticipated Transients Without Scram (ATWS) and the operation of the system during feed and bleed decay heat removal are not considered in this review. Neither the Licensee nor the NRC have evaluated the performance of the system for these events.

4. REVIEW AND EVALUATION

4.1 Valves Tested

Turkey Point, Units 3 and 4, are three loop PWRs designed by the Westinghouse Electric Corp. Each unit is equipped with three (3) pressurizer safety valves, two (2) PORVs, and two (2) PORV block valves in its overpressure protection system. The safety valves are 4-in. Crosby Model HB-BP-86 4K26, spring loaded valves with loop seal internals. The design set pressure is 2485 psig and the rated steam flow capacity is 288,000 lbm/h. The PORVs are 2-in. Copes-Vulcan Model D-100-160 globe valves with 17-4 PH plugs and cages. The PORV opening set pressure is 2335 psig and the rated steam flow capacity is 210,000 lbm/h. The PORV block valves are 2-in. Velan Model B10-3054B-13M gate valves with Limitorque SMB-000-5 motor operators. The inlet pipe to the safety valves and the PORVs include cold loop seals.

The Crosby 4K26 safety valve used at Turkey Point 3 and 4, was not specifically tested in the EPRI safety and relief valve test program. Two similar valves, which were tested by EPRI, are the Crosby HP-BP-86 3K6 and 6M6 valves. These valves are of the same basic design but vary in orifice size and flow capacity. A comparison of the size and capacity of these valves is shown below.

Valve	Model	Inlet Diameter (in.)	Outlet Diameter (in.)	Nozzle Bore Diameter (in.)	Rated Flow (lbm/h)
Turkey Point 3,4	4K26	4	6	1.800	288,000
Test	3K6	3	6	1.536	212,182
Test	6M6	6	6	2.154	420,006

The difference in orifice size only affects the valve capacity but not valve behavior. Other differences, such as body construction (cast or forged) and disk holder type and disk materials, do not have a significant

effect on the valve operability. The results of the EPRI tests on the Crosby 3K6 and 6M6 safety valves are, therefore, adequate to demonstrate the operability of the Turkey Point 3 and 4 safety valves.

The Copes-Vulcan PORV tested by EPRI is essentially the same as the Copes-Vulcan PORVs installed at Turkey Point 3 and 4. The Turkey Point PORVs have a 2-in. NPS body with 17-4PH plug and cage. The test valve used the 17-4PH plug and cage, but the valve body was 3-in. NPS. The difference in body size will not effect operability and, thus, the test valve adequately represents the plant valve.

The PORV block valve used in the EPRI tests was the Velan Model B10-3054B-13MS gate valve, identical to those used at Turkey Point 3 and 4. Although the operators used for the Turkey Point block valves are smaller than those used in the EPRI tests, they are considered adequate for the proper operation of the in-plant block valves. (This will be explained later in Section 4.3.4 on valve operability.) Therefore, the applicable EPRI test results can be used to evaluate the operability of the Turkey Point PORV block valves.

Based on the above, the valves tested are considered to be representative of the in-plant valves at Turkey Point 3 and 4, 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

Turkey Point, Units 3 and 4, are three 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 in those sections of the report applicable to a three loop plant. The transients considered in this report include FSAR transients, 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.

The Turkey Point 3 and 4 safety valve ring settings were set at the factory, and are therefore considered representative of factory ring settings for the Crosby 3K6 and 6M6 test valves. The calculated in-plant inlet pressure drop when the safety valve opens is 433 psi, compared to the test 3K6 test valve pressure drop of 391 psi, which indicates that the in-plant valve may have a stability problem. Due to the combination of high inlet pressure drop and inlet pipe length the valve may start chattering upon initial opening and may not be able to pass the required amount of steam. The Turkey Point 3 and 4 safey valves are mounted on cold loop seals. The water at the valve inlets is approximately 120°F. The maximum predicted backpressure for the Turkey Point 3 and 4 safety valves is 493 psia.

4.2.1 <u>FSAR Steam Transients</u>

The limiting event resulting in steam discharge through the safety valves and through both safety valves and PORVs is the locked rotor accident. When the safety valves actuate with the PORVs closed, the safety valves are expected to experience a peak pressure of 2592 psia and a maximum pressurization rate of 216 psi/s (data for three-loop plants from Reference 7).

Test conditions similar to the valve inlet conditions described above were selected from the EPRI tests on the Crosby 3K6 and 6M6 safety valves for the qualification of the 4K26 safety valves installed at Turkey Point 3 and 4. These safety valves were tested in long inlet piping configuration with loop seals similar to the Turkey Point safety valve installation, but with lower inlet pressure drop.

The Crosby 3K6 safety valves were tested with both steam and loop seal internals and long and short inlet configurations. Four tests were performed with drained loop seals (Steam Tests 506, 508, 516, and 518). Two of these were with factory ring settings (-55, -14) and two with lowered ring settings (-115, -14). Four tests were also performed with a cold loop seal and lowered ring settings (Tests 525, 526, 529, and 536).

The ring settings were (-115, -14). In addition, two steam tests were performed with short inlets and factory ring settings. In these tests, the peak pressure ranged from 2456 to 2709 psia and the pressurization rate ranged from 2.6 to 286 psi/s. These tests bound all the steam transient inlet conditions for the Turkey Point 3 and 4 safety valves which have the factor ring settings and long inlets with cold loop seals.

The Crosby HP-BP-86, 6M6, safety valve was subjected to eleven loop seal-steam tests with a long inlet configuration in the EPRI testing program. Three of these tests (No. 929, 1406, and 1415) were applicable to Turkey Point 3 and 4 because the valve ring settings in these tests were representative of the plant valve ring settings and cold loop seals were used. In these tests, the peak pressure ranged from 2675 to 2760 psia and the pressurization rate ranged from 319 to 360 psi/s. The test conditions for the Crosby 3K6 and 6M6 valves identified above bound these of the plant valves. Therefore, the test inlet fluid conditions for the loop-seal steam tests were representative of the expected conditions for FSAR transients resulting in steam discharge from the safety valves.

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

4.2.2 FSAR Liquid Transients

The only FSAR transient resulting in liquid flow through the PORV and safety valves is a main feedline break. Based on a review of three loop plants in Reference 7, the safety valves can be expected to pass liquid following a main feedline break. The maximum pressure will be 2575 psia, pressurization rate will be less than 8.0 psi/s, maximum surge rate will be less than 2010.8 GPM, and range of liquid temperatures will be 620.1°F to 672.0°F.

One water test was conducted on the Crosby 3K6 safety valve (431a). Test 431a was run with a short inlet and ring settings slightly higher that the factory recommendend settings (-45, -14). The fluid temperatures during the test were between 622°F and 631°F, the pressurization rate was 1.8 psi/s, the maximum pressure was 2342 psia, and the valve passed 1370 GPM. One transition test from steam to water was run on the 6M6 safety valve (test 931a). The test valve had factory recommended ring setting. The fluid temperature during the steam to water transition was 673°F, peak pressure was 2578 psia, and the valve passed 2355 GPM. The 3K6 and 6M6 valves were stable during the water and steam/water transition tests and the fluid conditions at the valve inlet bound those expected for a Westinghouse three-loop plant and should bound Turkey Point 3 and 4.

4.2.3 Extended High Pressure Injection Event

The limiting extended high pressure injection transient is a spurious actuation of the safety injection system at power. According to Reference 7, no steam or liquid discharge through the safety valve is expected for three-loop Westinghouse PWRs such as Turkey Point 3 and 4. For the PORVs, both steam and liquid discharges may occur. The maximum pressure is predicted to be 2352 psia at temperature between 498°F and 502°F. The pressurization rate is less than 12 psi/s. The steam discharge conditions for the PORV is bounded by the FSAR steam transient condition discussed in Section 4.2.1. Liquid discharge would not take place until the pressurizer becomes water solid. According to Reference 7, this would not occur until

at least 20 minutes into the event which allows ample time for the operator to take appropriate action to terminate the water injection. Therefore the potential for liquid discharge in an extended HPI can be disregarded.

4.2.4 Low Temperature Overpressurization Transient

Only the PORVs are ued for low temperature overpressure protection (LTOP). The safety valves are not affected in this transient. The low pressure set point of Turkey Point 3 and 4 is 415 psig at a maximum temperature of 285°F. In Reference 16, the licensee stated that the peak pressure and temperature at the PORV inlet during a low temperature overpressurization event are less than 675 psia and 442°F.

There were two low pressure and temperature water test performed on the Copes-Vulcan PORV with stellited plug and 17-4PH cage similar to the in-plant valves. The tests were conducted at an inlet pressure of 675 psia and at temperatures of 105°F and 442°F respectively. These inlet fluid conditions of the EPRI tests adquately envelop the expected inlet fluid transients of Turkey Point 3 and 4.

4.2.5 PORV Block Valve Fluid Conditions

The PORV block valves are required to operate over the same range of fluid conditions as the PORVs. In the EPRI tests, the block valve was only tested at full pressure (to 2500 psia) steam conditions. The operability of the block valves under water flow conditions was not directly addressed in the EPRI tests. However, the Westinghouse gate valve closing tests (Reference 9) demonstrated that the required torque to open or close the valve depended almost entirely on the differential pressure across the valve disk and was insensitive to the momentum load. Therefore, the required force is nearly independent of the type of flow (i.e., water or steam). Furthermore, according to the friction tests done by Westinghouse on a stellite coated specimen, the friction coefficient between stellite surfaces is approximately the same for steam and water tests. In some instances, the friction force in water media is lower than in steam. The Velan block

valves have stellite coated disk and seats. The force required to overcome disk friction in steam is essentially equal to the force required in water. Therefore, the steam tests are adequate to demonstrate the operability of the block valves for expected water conditions.

The test sequences and analyses described above demonstrate that the test conditions bound the conditions for the plant valves. They also verify that Items 2 and 4 of Section 1.2 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 required showing that the test conditions are equivalent to conditions prescribed in the FSAR, is also met.

4.3 Operability

4.3.1 Safety Valves

The objective of the EPRI test program was to demonstrate operability of all safety and relief valves by testing a representative selection of all valves in operation. Operability of valves not tested was to be demonstrated by interpolating or extrapolating data from similar valves that were tested and performed satisfactorily. Since the Crosby 4K26 safety valves was not tested, the test results obtained with a smaller valve (3K6) and a larger valve (6M6) were used to demonstrate the operability of the in-plant 4K26 valve.

The EPRI test results identified in Section 4.2.1 show that when the Crosby 3K6 valve is mounted on the long inlet piping, it is only marginally stable with factory recommended ring settings. during test 506, the valve fluttered slightly but closed approximately 45 sec after opening. However, during test 508, the valve started chattering approximately 4 seconds after the valve opened, and the test had to be terminated. The computed inlet pressure drop when a Turkey Point 3 and 4 4K26 valve opens (433 psi) is higher than when the test 3K6 valve opens (391 psi). The computed pressure increase when a Turkey Point 4K26 valve closes is 244 psi which is also

higher than the pressure increase when the test 3K6 valve closes. A number of tests were run with the long inlet piping with lower ring settings which were (more) stable, which indicates that stability is a function of ring settings. However, the functional relationship between stability and ring settings was not provided by the licensee.

The 6M6 valve does perform stably with factory ring settings when installed on a long inlet pipe. However, the computed inlet pressure drop on opening (263 psi) is also much lower than for the 3K6 valve or for the Turkey Point 3 and 4 4K26 valve. The computed pressure increase when the 4K26 valve closes is also higher than the 6M6 safety valve.

The results of the EPRI tests indicate that the Turkey Point 3 and 4 safety valves will not be as stable as the test 3K6 safety valves with the original factory recommended ring settings and inlet pipe configuration because of the larger inlet pressure drop when the valve opens. It therefore cannot be concluded that the Turkey Point 3 and 4 4K26 safety valve will operate stably as installed.

The blowdown with factory ring settings ranged between 6.8 and 10.9% for 3K6, and between 5.1 and 9.4% for the 6M6. One hundred four percent of rated steam flow was achieved at 6% accumulation for the 3K6 and 109% of rated flow was achieved at 3% accumulation for the 6M6 valve.

While the 6M6 valve had better performance than the 3K6 valves because of the poor performance of the 3K6 valve in the applicable tests, there appeared to be a generic problem with the Crosby 6M6 (and 3K6) safety valves when the entire test series on the 6M6 valve is considered. In the EPRI tests, a total of 17 steam and water discharge tests were performed on the 6M6 valves in the loop seal inlet configuration. During the course of the tests, the valve was inspected 9 times because of the repeated problems encountered in valve closure. In most cases, galled guiding surfaces and damaged internals were found and the damaged parts were either refurbished or replaced before the next test. The valve performed well after each repair but started to chatter on closing in the subsequent test and the

test had to be terminated. This pattern of valve behavior suggests that inspection and maintenance of valves are important to ensure continued reliable operability of Crosby safety valves. This problem was not addressed in the Turkey Point Unit 3 and 4 submittal.

During the loop-seal transition test 931a, the Crosby 6M6 safety valve performed stably. The liquid temperatures during liquid discharge was 641°F, and the valve discharged 2355 GPM. Also during test 431a, the Crosby 3K6 safety valve performed stably. the liquid temperature was ranged from 622°F to 631°F, and the valve discharged 1370 GPM. Since the 3K6 valve is smaller than the plant 4K26 valve, and there are three plant safety valves at each unit, Turkey Point 3 and 4 has sufficient relief capacity for any transient involving liquid discharge.

The maximum bending moment induced at the outlet of the 3K6 safety valve (during EPRI Test 406) was 123,000 in.-lb which had no effect on valve operability. This bending moment is approximately equal to the maximum bending moment of 126,000 in. lb reported by the licensee for the Turkey Point 4K26 safety valve which has a larger inlet. However, it should be noted that the reported bending moment for the plant 4K26 safety valve does not include the effects of a safe shutdown earthquake (SSE). Thus it cannot be stated with certainty that the moment loading on the safety valve will not affect the operability of the safety valves at Turkey Point 3 and 4.

4.3.2 Power Operated Relief Valve

The EPRI tests on the Copes Vulcan PORV indicated that the valve fully opened and closed on demand and within the required opening and closing time of 2.0 s. Disassembly after testing showed that cage to body gasket had partially washed out but no damage was found that would affect valve performance. The lowest steam flow rate observed in the tests (255,600 lbm/h) is much higher than the rated flow of 210,000 lbm/h for the Turkey Point PORVs. 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 Turkey Point PORV was not provided, thus it cannot be stated that the moment loading on the PORVs will not affect the operability of the safety valves at Turkey Point 3 and 4.

NUREG-0737 II.D.1 requires qualification of associated control circuitry as part of the safety/relief valve qualification. The Nuclear Regulatory Commission staff has agreed, however, that meeting the licensing requirements of 10 CFR 50.49 for this electrical equipment is satisfactory and that specific testing per the NUREG-0737 requirement is not required. In Reference 16, FLP stated that the safety related electrical equipment required for PORV operation and/or monitoring have been determined to be within the scope of 10 CFR 50.49 and have been qualified for the environmental conditions under which the equipment must function. Therefore, it can be concluded that the PORV circuitry meets the qualification requirements of NUREG-0737 Item II.D.1.

4.3.3 PORV Block Valves

The Velan PORV block valve was cycled 21 times against full steam flow in the EPRI testing program. Steam pressure upstream of the block valve varied from 2440 psia to 2500 psia during the opening cycles and between 2340 psia and 2410 psia during the closing cycles. The stroke times of the test valve were 9.7 s to 9.9 s, which are within the required stroke time of 10.0 s. Tests for water flow through the Velan block valve were not performed in the EPRI test program. As explained in Section 4.2.5 of this report, the valve behavior under water flow conditions is expected to be similar to that of the full pressure steam tests. Thus, the operability of the valves for liquid flow condition has been indirectly demonstrated.

The Turkey Point 3 and 4 PORV block valves use a smaller Limitorque operator (SMB-000-5) than the SB-00-15 and SMB-000-10 operators used in the EPRI tests. The torque switch of the SMB-000-5 operator has been set at its highest value in order to provide sufficient thrust to operate the block valve. The SMB-000-5 operator for the in-plant block valve can deliver a torque of 100 ft-lb or more, when its torque switch is set at 3.0. In the supplementary tests on the Velan block valve, a number of tests were

performed at lower torque settings to investigate valve operability at reduced operator torque. It was found that the test valve opened and closed satisfavtorily even when the torque switch in the SB-00-15 or the SMB-00-10 operator was set at its lowest value of 1.0, which corresponded to a torque of 82 ft-lb. Based on the operator torque used in the tests, the amount of torque that the SMB-000-5 operator can deliver is more than adequate to assure the operability of the Turkey Point 3 and 4 block valve.

The test results discussed above indicate that with the factory recommended ring settings, and the inlet piping configuration, the Turkey Point 3 and 4 safety valves are not expected to operate stably, and neither the safety valves or PORVs have been shown to be operable with the maximum moments that they are expected to be exposed to during operating and accident conditions. The block valves have been shown to operate satisfactorily under the expected operating and accident conditions.

Therefore, the part of Item 1 of Section 1.2 of this report, which requires tests to qualify the valves has been satisfied for the safety valves, PORVs and the block valves in so far as the applicable test valves are concerned. However, because of the differences between in plant and test valve installation, the tests qualify the block valves but not the safety valves and the PORVs. Item 2 requiring determination of expected operating conditions using accidents and operational transients listed in Regulatory Guide 1.70, Revision 2 and Item 4, requiring the use of the highest test pressure predicted by conventional safety analysis procedures were both met. Item 5, which requires qualification of the PORV control circuits has been met.

4.4 Piping and Support Evaluation

This evaluation covers the piping and supports upstream and downstream of the safety valves and PORVs extending from the pressurizer nozzles to the pressurizer 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.

4.4.1 Thermal Hydraulic Analysis

Pressurizer fluid conditions were selected for use in the thermal hydraulic analysis such that the calculated pipe discharge forces would bound the forces for any of the FSAR, HPI and low temperature overpressurization events, including the single failure that would maximize the forces on the valve.

The valve operating conditions considered in the thermal hydraulic analysis were as follows, each involved discharge of cold loop seals:

1. Both PORVs open simultaneously with the safety valves closed

2. All three safety valves open with the PORVs closed

3. PORVs open simultaneously, while the pressurizer pressure continues to rise; all safety valves open subsequently.

In the first case, the two PORVs opened at a pressure of 2419.75 psia (or 3% above the set point) with zero pressurization rate and steam in the pressurizer at saturated temperature. The water in the cold loop seal at the valve inlet was 120°F. In the second case, the three SRVs opened at a pressure of 2574.25 psia (or 3% above the set point) with zero pressurization rate and steam in the pressurizer at saturation temperature. The water in the cold loop seal at the valve inlet was 120°F. The third case is not bounding, and so was not actually analyzed. In this case, the system initially behaves as the first case until the pressure reaches the setpoint of the safety valves. The safety valves then open against a larger backpressure than that of the second case, thereby producing lesser loads.

A solid water case was not analyzed for these plants because the discharge of cold water seals is expected to produce more severe loading.

The thermal hydraulic analysis was performed using the RELAP5/MOD1 computer program and an Ebasco inhouse postprocessor, CALPLOTFIII. RELAP5 calculates the thermal hydraulic properties of the fluid as a function of time, such as pressure, temperature, and density at each control volume and at each junction. The CALPLOTFIII program uses the fluid parameters from RELAP5 analysis to generate the force time histories to be applied at each pipe segment. RELAP5 is a thermal-hydraulic analysis program widely used in the industry and has been shown to be a suitable tool for the prediction of piping discharge loads (Reference 17). The CALPLOTFIII program was verified by using the program to calculate transient forces from the EPRI/CE test 1411 safety valve actuaion, as presented in Reference 17. The forces calculated by CALPLOTFIII duplicated the forces given in Reference 17 for test 1411.

The valve flow rates used in the analysis were calculated by the RELAP5/MOD1 program. Flow areas were adjusted to give a flow rate through the safety valve of 356,400 lbm/hr. This is 121% of the original flow rating of the safety valve (295,000 lbm/hr) and 111% of the revised flow rating of the safety valve (320,000 lbm/hr). The calculated flow rate through the PORVs was 266,400 lbm/hr, which is 174% of the valve flow rating (153,000 lbm/hr). The flows used in the analysis suitably account for the ASME Code derating of the safety valves and possible uncertainties in flow rates through the valves.

4.4.2 Stress Analysis

The structural analysis was performed initially using the Ebasco computer code PIPESTRESS 2010. This elastic analysis showed that overstresses occurred in the piping system for both PORV and safety valve discharges of cold loop seals. Since the overstress ratios were highest for safety valve actuations, the analysis involved safety valve discharge was extended into the plastic region of stress-strain relationships. This was accomplished by using the calculated fluid forcing functions as input to the PLAST computer code. The intent of this analysis was to show adequacy of the system when stress levels exceed yield stress.

The plastic analysis indicated plasticity in the first elbow downstream of the pressurizer and the elbow downstream of the safety valve. To determine whether the plasticity was acceptable, the licensee arbitrarily established 70 percent of the ultimate moment capacity of the pipe as an allowable bending moment. The licensee stated that the ASME Code allows use of this value, and proceeded to use the "Method of Gerber" to compute the moment capacity of the pipe. The maximum calculated bending moment as a percentage of computed moment capacity for safety valve actuation was 65 percent. Since this is less than the prescribed allowable of 70 percent, the licensee concluded that the piping system is adequate for safety valve actuation. The licensee further concluded that the system is adquate for PORV actuation, since the PORV elastic overstress ratios were not as high as those for safety valve actuation. He rationalized that the safety valve discharges serve as the bounding event.

The licensee is correct in stating that the ASME Code allows the use of plastic analysis when elastic stress limits are exceeded. ASME Code Section III Paragraph NB-3200 permits the use of plastic analysis, specifying that the loadings are not to exceed two-thirds of the plastic analysis collapse load. Use of these ASME Code criteria, however, requires that all of the design be consistent with Code criteria. For example, the plastic analysis collapse load must be determined in accordance with (or more stringently than) the method specified by the licensee (i.e. Method of Gerber). In addition, any effects of plastic strain concentration on fatigue, ratcheting, or buckling behavior of the piping must be considered. Without addressing these other Code requirements, the licensee has not established defendable acceptance criteria for the plastic analysis.

A second and more important difficulty with the licensee's structural analysis is that it does not include a faulted load combination, per recommendations of the EPRI Submittal Guide (Reference 10), in which safe shutdown earthquake (SSE) loading is combined with a safety valve actuation. Adding SSE loads to the bending moments produced by safety valve actuations would almost certainly increase the tatal plastic bending moment beyond the licensee's allowable of 70 percent ultimate capacity. As

previously mentioned, the maximum calculated moment as a percentage of ultimate capacity for safety valve actuations alone was already 65 percent.

A third problem with the licensee's analysis relates to the loading on supports during PORV actuation. In the response to the NRC's request for additonal information (Reference 16), the licensee summarized the reaction on supports during PORV and safety valve actuations in Tables 4.4 and 4.4A, respectively, and stated that all reactions were within restraint capacities. However, the transient loads listed in these tables in severalinstances exceeded the listed design load or support capacity. Also, the information in Table 4.4A did not compare support loads for a faulted condition (safety valve actuation plus SSE loads) with the allowable support loads. This load combination would likely intensify overstresses already indicated in this table.

Because of the problems described above, the licensee has not presented a structural analysis that satisfactorily demonstrates adequacy of the piping system for PORV or safety valve transients involving discharges of cold loop seal water. In addition, the licensee has not properly accounted for all load combinations recommended in the EPRI PWR Safety and Relief Test Program Guide (Reference 10).

Therefore, while the licensee has appropriately selected the transient condition that produces maximum dynamic forces on the PORV and safety valve piping systems, which meets Item 3 of Section 1.2. The structural analysis, though, does not acceptably meet the Item 8 requirement that the PORV and safety valve piping and supports be qualified by performing appropriate analysis.

5. EVALUATION SUMMARY

The Licensee for the Turkey Point, Units 3 and 4, has not 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 have been met. The rationale for this conclusion is given below.

The Florida Power and Light Co. participated in the development and execution of an acceptable Relief and Safety Valve Test Program designed to qualify the operability of prototypical valves and to demonstrate that their operation would not invalidate the integrity of the associated equipment and piping. The subsequent tests were successfully completed under operating conditions which by analysis bounded the most probable maximum forces expected from anticipated design basis events. The generic test results and piping analyses showed that the valves tested functioned correctly and safely for all relevant steam discharge events specified in the test program and that the pressure boundary component design criteria were not exceeded. Analysis and review of the test results and the licensee's justifications indicated direct applicability of the prototypical valve and valve performances to the in-plant valves and systems intended to be covered by the generic test program.

The test results demonstrated the need for inspection and maintenance of the safety valves following each lift involving loop seal or water discharge for continued reliable operability of the safety valves. However, the licensee has not described the methods by which continued operability of the safety valves will be assured. In addition, the pressure drop when the safety valves open and pressure increase when the safety valves close are larger than the corresponding values for the test valves, which suggests that the Turkey Point 3 and 4 safety valves may not perform stably.

Thus, while the requirements of Item II.D.1 of NUREG-0737 have been partially met (Items 1-6, and part of Item 7 in Paragraph 1.2), the part of Items 7 which requires consideration of the effect of as-built discharge piping on valve operability, and Item 8 which requires qualification of

piping and supports have not been met. Therefore the licensee has not demonstrated by testing and analysis, that the reactor primary coolant pressure boundary will have a low probability of abnormal leakage (General Design Criterion No. 14) and that the reactor primary coolant pressure boundary and its associated components (piping, valves, and supports) have been designed with sufficient margin such that design conditions are not exceeded during relief/safety valve events (General Design Criterion No. 15).

However, 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 reactors have experienced a number of occurrences of improper performance of safety and relief valves installed in the primary coolant system. As a result, the authors of NUREG-0578 (TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations) and subsequently NUREG-0737 (Clarification of TMI Action Plan Requirements) recommended that programs be developed and completed which would reevaluate the functional performance capabilities of Pressurized Water Reactor (PWR) safety, relief, and block valves and which would verify the integrity of the piping system for normal, transient, and accident conditions. This report documents the review of these programs by the Nuclear Regulatory Commission (NRC) and their consultant, EG&G Idaho, Inc. Specifically, this report documents the review of the Turkey Point Units 3 and 4 Licensee response to the requirements of NUREG-0578 and NUREG-0737. This review found the Licensee had not provided an acceptable response, which would reconfirm that General Design Criteria 14, 15, and 30 of Appendix A to 10 CFR 50 have been met.

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