Enclosure 3

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TECHNICAL EVALUATION REPORT
TMI ACTION--NUREG-0737 (II.D.1)
RELIEF AND SAFETY VALVE TESTING
ST. LUCIE, UNIT 2
DOCKET NO. 50-389

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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 St. Lucie 2 Licensee response to the requirements of NUREG-0578 and NUREG-0737. This review found the Licensee has not provided an acceptable response and, therefore, has not reconfirmed that the General Design Criteria 14, 15, and 30 of Appendix A to 10 CFR 50 were 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 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 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. Florida Power and Light Co. (FPL), the owner of St. Lucie 2, was one of the utilities sponsoring the EPRI Valve Test Program. The results of the program, which are contained in a series of reports, were transmitted to the NRC by Reference 3. The applicability of these reports is discussed below.

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

EPRI contracted with Combustion Engineering (CE) to produce a report on the inlet fluid conditions for pressurizer safety and relief valves in CE designed plants (Reference 7). Since St. Lucie 2 was designed by CE, 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, which is 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/CE 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 submittal for the adequacy of the overpressure protection system was submitted by FPL on March 22, 1983 (Reference 10). The submittal included the safety valves, PORVs, block valves, and the safety valve/PORV piping analyses required. A request for additional information (Reference 11) was submitted to FPL by the NRC on July 11, 1985. FPL responded to this request on March 18, 1986 (Reference 12). A second request for information was sent to FPL on June 10, 1987 (Reference 13). FPL responded to this request on November 6, 1987 and February 5, 1988 (References 14 and 15).

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

St. Lucie 2 is a CE designed PWR that utilizes three safety valves, two PORVs, and two PORV block valves in the overpressure protection system. The safety valves are Crosby Model HB-BP-86 3K6 valves. The plant safety valves are mounted directly on the pressurizer nozzles. The safety valves all have the same 2500 psia set pressure. The PORVs are Garrett Model 3750010 3 x 8-inch right angle solenoid actuated valves with a 2400 psia set pressure. St. Lucie 2 does not have loop seals upstream of the PORVs. The block valves are Westinghouse Model 0300GM88FNH gate valves with Limitorque SB-00-15 operators.

The Crosby 3K6 valve was one of the valves tested by EPRI. The plant and test valve parameters are identical; therefore, the test valve is considered to adequately represent the plant valves.

The Garrett PORVs installed at St. Lucie 2 are 3 x 8-inch right angle valves. The valve tested by EPRI was a Garrett 3 x 6-inch straight through design with a 2.150 in. bore seat diameter. The valve manufacturer states that although the right angle valve differs from the straight through design in terms of hydrodynamic efficiency and external loading the test results from the Garrett/EPRI test valve are considered fully applicable to the CE Garrett valve (Reference 16). Therefore, the test valve is considered an adequate representative of the in-plant valves.

The St. Lucie 2 block valves are Westinghouse Model 0300GM88FNH (Series 88) wedge type gate valves with a Limitorque SB-00-15 operator. The Westinghouse gate valve tested by EPRI was a Model MOD03000GM88FNB0D0 (Series 88) with Limitorque operator SB-00-15. FPL used EPRI and Westinghouse data to qualify their block valves (References 9 and 17). The test valve and the St. Lucie 2 valves are similar enough to be representative.

Based on the above, the safety valve, PORV and block valve tested are considered to be applicable to the in-plant valves at St. Lucie 2 and to have fulfilled that part of the criteria of Items 1 and 7 as identified in Section 1.2 regarding applicability of test valves.

4.2 Test Conditions

The valve inlet fluid conditions that bound the overpressure transients for CE designed PWR plants are identified in Reference 7. The transients considered in this report include FSAR, extended high pressure injection (HPI), and low temperature overpressurization events.

For the safety valves, only steam discharge was calculated for FSAR type transients (including a feedwater line break). The peak pressure predicted was 2752 psia and the maximum pressurization rate was 92 psi/s. A maximum backpressure of 308 psia is developed at the safety valve outlet (Reference 18). St. Lucie 2 has the safety valves mounted directly on a pressurizer nozzle. FPL stated in Reference 12 that the plant valve adjusting rings will be set to (-55, -14) for all the plant safety valves. These positions are relative to the level position.

Three steam tests with the Crosby 3K6 valve were run with ring settings of (-55, -14). These were tests 408, 411, and 442. These tests were run on the short (82 in.) inlet piping configuration which is more representative of the St. Lucie 2 configuration. In these tests the peak pressure ranged from 2462 to 2683 psia, the pressurization rate ranged from 2.5 to 314 psi/s, the peak backpressure ranged from 602 to 678 psia, the opening pressure ranged from 2462 to 2505 psia (-1.5 to +.2% of nominal set pressure), the blowdown ranged from 10.3 to 10.9%, and the valve flow rate ranged from 121 to 122% of the rated flow rate at 3% accumulation. These conditions are close enough to the conservatively predicted inlet conditions for St. Lucie 2 to adequately represent expected valve performance.

Review of the CE inlet conditions report (Reference 7) showed that water did not reach the valve during FSAR transients or an extended high pressure injection (HPI) event. The cutoff head for the St. Lucie 2 HPI

pumps is below the safety valve setpoint so that an extended HPI event would not challenge the safety valves.

There was a concern that the extended safety valve blowdown (blowdown greater than 5%) observed during the EPRI tests could result in the pressurizer level increasing to the safety valve inlet. CE, in Reference 18, analyzed the loss-of-load (LOLD) transient assuming 20% blowdown. Other conservative assumptions were also made to maximize pressurizer level swell. The LOLD was chosen because it provided the design basis for sizing pressurizer safety valves. The 20% blowdown is conservative since the blowdown observed in the applicable EPRI tests ranged from 10.3% to 10.9%. This analysis showed the pressurizer level did not reach the inlet to the safety valves. Thus, the steam inlet condition was maintained.

The two Garrett PORVs at St. Lucie 2 do not have loop seals. The peak pressure and pressurization rate for the PORVs during FSAR type transients are given in Reference 7 and are, 2752 psia and 92 psi/s, respectively. The maximum backpressure for the PORVs is 308 psia (Reference 12).

The test PORV was subjected to thirteen steam tests. In the steam tests, the opening pressure ranged from 2415 psia to 2760 psia. Backpressures ranged from 335 psia to 825 psia. The testing of the Garrett PORV was performed at opening pressures that bound the peak pressures indicated in Reference 7 for St. Lucie 2 during an FSAR transient (2415 to 2760 psia versus 2752 psia). The test conditions are representative of the plant conditions.

As with the safety valves, the CE inlet conditions report (Reference 7) indicated that water did not reach the PORV during FSAR transients or an extended HPI event. The cutoff head for the St. Lucie 2 HPI pumps is below the PORV set pressure so that an extended HPI event would not challenge the PORVs.

The PORVs are used for low temperature overpressure protection (LTOP) at St. Lucie 2 with set pressures of 460 and 490 psia. The CE inlet conditions report, Reference 7, included a plant-specific analysis of LTOP

for St. Lucie 2. For low temperature overpressure protection, the valve inlet conditions include steam, steam to water transition, and liquid at pressures up to 477 psia with temperatures ranging from 100 to 417°F and a pressurization ramp rate up to 80 psi/s. The peak pressures noted above are based on analyses that assumed the pressurizer was liquid full. The presence of a steam bubble in the pressurizer, which is the recommended mode of operation during low temperature operation, would limit the peak pressure when the PORV opened on steam, but this condition was not specifically analyzed. Thus, peak pressure during steam discharge was bounded using the liquid full analyses. The steam discharge conditions are considered to be adequately represented by the high pressure tests discussed above. Steam to water transition is also considered to be adequately represented by the two high pressure transition tests, 104-GA-S/W and 104-GA-8S/W. Water discharge during a LTOP transient is represented by the low pressure (~685 psia) water tests with fluid temperatures ranging from 104 to 447°F.

Verification of block valve operability was shown by EPRI testing representative Westinghouse valves in their test program (References 9 and 17). These tests verified that the test block valves could be opened with full 2400 psig pressure on the valves and closed with full flow and pressure through the valves. These conditions are representative of the plant conditions expected during actual valve operation (References 10 and 12).

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 Valve Operability

As discussed in the previous section, the Crosby 3K6 safety valves at St. Lucie 2 are required to operate with steam inlet conditions only. The EPRI test program tested the Crosby 3K6 valve for the required range of conditions. During FSAR transients the PORVs are required to only pass steam. The PORVs are used for LTOP and in this mode may be required to pass steam, steam to water transition, and water. The test valve was subjected to the required conditions. The block valves are also required to operate for steam and liquid flow conditions.

For the applicable safety valve tests (408, 411, and 442) with the plant (-55, -14) ring settings, the test valve opened at pressures from 2462 to 2505 psia (-1.5% to +.2% of the nominal set pressure), had stable behavior, and closed with 10.3% to 10.9% blowdown. In these tests the valve passed from 121% to 122% of rated flow at 3% accumulation. This indicates the valve was able to perform its safety function of opening, relieving pressure, and closing with these ring settings (-55, -14).

A maximum bending moment of 133,000 in-lb was applied to the Crosby 3K6 safety valve discharge flange without impairing valve operation. This bounds the maximum expected bending moment of 72,718 in-lb at the plant (Reference 12).

For a test to be an adequate demonstration of safety valve stability, the test inlet piping pressure drop should exceed the plant pressure drop. The measured stagnation pressure drop for the reference test (#411) was 211 psid. The plant specific stagnation pressure drop was calculated to be 106 psia indicating the plant valves should be as stable as the test valves.

As noted above, the valve blowdown for the Crosby 3K6 safety valve during the applicable tests ranged from 10.3% to 10.9%. A CE analysis for a St. Lucie 2 LOLD with 20% blowdown showed that the pressurizer level would not reach the safety valve inlet. This bounds the blowdown observed in the tests. Also, the hot leg remained subcooled during the LOLD analysis with the extended blowdown indicating adequate core cooling was maintained.

Since the ring settings used with the St. Lucie 2 safety valves were used in the EPRI tests they are deemed acceptable. The test valve flow rate was 121 to 122% of its rated value. This indicates the valves at St. Lucie 2 will provide adequate overpressure protection.

Based on the test results discussed above, demonstration of safety valve operability is considered adequate.

The Garrett PORV opened and closed on demand for all applicable tests. Inspection of the valve after testing at the Marshall Steam Station showed the valve in acceptable condition.

FPL said in Reference 12 that the manufacturer stated that their PORV would operate on low pressure steam as well as low pressure water down to a pressure of 100 psig with zero backpressure. This is well within the operating requirements of St. Lucie 2 PORVs.

A bending moment of 33,200 in-lb was induced on the discharge flange of the test valve without impairing operability. The maximum bending moment calculated for the St. Lucie 2 PORVs is 83,635 in-lb. FPL noted in Reference 12 that the manufacturer's report stated the maximum bending moment allowed on the Garrett valve without effecting valve operation is 193,200 in-lb. Based on the manufacturers recommended bending moment and the valve performance during EPRI tests, under the full range of expected inlet conditions, the demonstration of PORVs operability is considered adequate.

The EPRI tests of representative Westinghouse Series 88 block valve and its Limitorque SB-00-15 operator under all typical plant conditions demonstrated the block valve operability (Reference 9). The test valve opened and closed adequately with a torque setting of 3.75 (182 ft-lb). All cycles produced fully open and closed valve positions. The results of these tests indicate satisfactory block valve performance for the plant valves. FPL made manufacturer recommended changes to the valve gear drive and motor wiring before these valves were installed at St. Lucie 2.

NUREG-0737, Item II.D.1, requires qualification of the associated control circuitry as part of the safety/relief valve qualification. FPL qualified the reactor building PORV control circuitry equipment under 10 CFR 50.49; this equipment includes PT-1102 A, B, C, and D, PT-1103, PT-1104, PORV acoustic flow monitoring system, electric cables, and electric penetrations (Reference 12). All other electrical components for indication

and control, are located in mild environment areas in the Reactor Auxiliary Building. This meets the NUREG-0737 Item II.D.1 requirements for qualification of the associated PORV control circuitry.

The presentation above, demonstrating that the safety valves, PORVs and block valves operated satisfactorily, verifies that the portion of Item 1 of Section 1.2 that requires conducting tests to qualify the valves and that part of Item 7 requiring that the effect of discharge piping on operability be considered were met. Qualifying the PORV control circuitry under 10 CFR 50.49 meets Item 5.

4.4 Piping and Support Evaluation

In the piping and support evaluation (References 10 and 12), the safety valve and PORV inlet piping and the piping between the valve discharge flanges and the pressurizer relief tank were analyzed (including supports). The piping and safety related supports were analyzed for the requirements of the ASME Code, Section III, 1971 Edition, including Summer 1973 Addenda. Piping upstream of the valves was considered Class 1 and downstream piping was considered Safety Class 2. Non-safety supports were designed in accordance with the ANSI B31.1 Code.

Three transients were analyzed (Reference 10). The first case assumed the two PORVs opened and the three safety valves remained closed. The second case assumed the PORVs did not open and the three safety valves opened simultaneously. The third case assumed the two PORVs opened and the pressure continued to increase until the three safety valves opened simultaneously. The valve inlet condition was always saturated steam.

In Reference 15, FPL provided the results of a thermal-hydraulic analysis that considered water through the PORV during a LTOP transient. Because of questions on the LTOP transient analysis, there is a potential problem regarding the St. Lucie, Unit 2, discharge piping. However, analysis of the PORV discharge piping for LTOP conditions is not considered by the NRC staff to be in the scope of the Item II.D.1 review. Therefore, this issue will be reviewed separately, at a later time, and will not impact the NUREG-0737, Item II.D.1, review for St. Lucie 2.

The thermal-hydraulic analysis was performed with RELAP5/MOD1. RELAP5/MOD1 was shown to be a suitable tool for performing the thermal-hydraulic analysis of valve discharge transients in Reference 19. RELAP5 output is input to CALPLOTFIII for calculation of piping loads. CALPLOTFIII was verified to accurately calculate these types of loads by comparing its results to hand calculated loads and to GE 4-inch pipe blowdown test results (Reference 12).

A RELAP5 model representing the piping from the pressurizer to the quench tank was assembled and is shown in Figure 4 of Reference 12. The safety valves were assumed to open in 6 ms and the PORVs in 130 ms. The safety valve and PORV opening times were based on the minimum pop time measured in the EPRI tests. The flow rates for the safety valves and PORVs were based on EPRI test data. The safety valve flow rate used was 122% of the rated value and the PORV flow rate used was 106% of its rated value. The safety valve and PORV rated for rates are 213,000 and 395,000 lbm/h, respectively. Reference 14 provided additional information on the thermal-hydraulic analysis. Node sizes in the piping model immediately downstream of the safety valves ranged from 0.58 to 1.0 ft while those in the model immediately downstream of the PORVs ranged from 0.9 to 1.0 ft. Information on the node sizes for the other components in the safety valve and PORV discharge piping model was not given. The time step used in the analysis varied with transient time. However, the time step used in the time period when the maximum forces were calculated, 2×10^{-4} s, is considered adequate.

Based on the information provided by FPL, the adequacy of the thermal-hydraulic analysis could not be determined because sufficient information on the node sizes in the piping model was not given.

The dynamic structural analysis was performed with the PIPESTRESS 2010 computer program. PIPESTRESS 2010 is a benchmarked program discussed in the St. Lucie 2 FSAR.

In Reference 12, FPL stated the load combination used in the stress analysis are consistent with the Standard Review Plan, Section 3.9.3, Rev. 1. The load combinations included the operating basis earthquake

combined with the maximum values of safety valve and PORV loads and compared to Level B limits. The design basis earthquake was combined with the maximum values of safety valve and PORV loads and the stresses compared to Level C limits. The loading used in the design of the restraints was the combination of the worst thermal, dead weight, seismic and safety valve/PORV discharge loads. The restraint design considered this combination and normal allowable stress values (Reference 15). These load combinations are acceptable and meet the criteria presented in Reference 20.

The key input parameters for the structural analysis were reviewed and found to be adequate. Time steps ranged from 2×10^{-4} to 1×10^{-3} s. Damping was 1% for the upset condition and 2% for the emergency condition. FPL stated in Reference 15 that the mass spacing and time step used in the analysis were considered adequate for calculations with a cutoff frequency of 100 Hz. However, the actual cutoff frequency used in the analysis was 72 Hz. The contributions to the loads from frequencies beyond the cutoff frequency were taken into account by utilizing the left out force option in PIPESTRESS 2010. FPL stated that for this reason the loads calculated with a cutoff frequency of 100 Hz would not be much different from those in the analysis submitted. Based on these considerations, the structural analysis is considered adequate.

However, from the information provided by FPL, it is not possible to conclude the St. Lucie 2 piping and supports meet applicable code allowables. This is because question 2a of Reference 13 requested FPL to provide a table comparing the calculated piping and support stresses and loads with the code allowables for the most highly loaded locations. FPL's response did not provide the requested comparison. FPL only stated that the allowable stresses were compared to the predicted pipe stresses and support loads.

The analysis discussed above demonstrated that a bounding case was chosen for the piping configuration. Therefore, Item 3 of Section 1.2 was met. However, the analysis discussed above for the piping and support system has not verified Item 8 was met.

5. EVALUATION SUMMARY

The Licensee for St. Lucie 2 has not provided an acceptable response to the requirements of NUREG-0737. Therefore, it has not been reconfirmed that the General Design Criteria 14, 15, and 30 of Appendix A to 10 CFR 50 were met. The rationale for this conclusion is given below.

5.1 NUREG-0737 Items Fully Resolved

Based on the following information provided by the Licensee, the requirements of Item II.D.1 of NUREG-0737 were partially met (Items 1 to 7 in Section 1.2).

The Licensee participated in the development and execution of an acceptable Relief and Safety Valve Test Program designed to qualify the operability of prototypical valves and to demonstrate that their operation would not invalidate the integrity of the associated equipment and piping. The subsequent tests were successfully completed under inlet 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.

Therefore, 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).

5.2 NUREG-0737 Items Not Resolved

Based on the Licensee's submittal, the following requirements of NUREG-0737, Item II.D.1, as shown in Section 1.2, were not met.

Item 8: Item 8, which requires qualification of the piping and supports, was not met. This is because sufficient information on the node sizes in the piping model (other than immediately downstream of the valves) was not given. Also, FPL did not provide a requested table comparing the calculated piping and support stresses and loads with the code allowables for the most highly loaded locations. FPL only stated that the allowable stresses were compared to the predicted pipe stresses and support loads.

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) were designed with sufficient margin such that design conditions are not exceeded during relief/safety valve events (General Design Criterion No. 15).

6. REFERENCES

1. TMI-Lessons Learned Task Force Status Report and Short-Term Recommendations, NUREG-0578, July 1979.

2. Clarification of TMI Action Plan Requirements, NUREG-0737, November 1980.

3. R. C. Youngdahl ltr. to H. D. Denton, Submittal of PWR Valve Test Report, EPRI NP-2628-SR, December 1982.

4. EPRI Plan for Performance Testing of PWR Safety and Relief Valves, July 1980.

5. EPRI PWR Safety and Relief Valve Test Program, Valve Selection/Justification Report, EPRI NP-2292, December 1982.

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

7. Valve Inlet Fluid Conditions for Pressurizer Safety and Relief Valves in Combustion Engineering-Design Plants, EPRI NP-2318, December 1982.

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

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

10. Letter R. E. Uhrig, Florida Power and Light Co., to D. G. Eisenhut, NRC, "St. Lucie Unit 2, Docket No. 50-389, Post-TMI Requirements, Item II.D.1-Final Response," L-83-168, March 22, 1983.

11. Letter E. J. Butcher, NRC, to J. W. Williams of Florida Power and Light Co., "Request for Additional Information re Response to TMI Action Item II.D.1 'Relief and Safety Valves Test Requirements,'" July 11, 1985.

12. Letter C. O. Woody, Florida Power and Light Co., to A. C. Thadani, NRC, "St. Lucie Units 1 and 2, Docket Nos. 50-335 and 50-389, Relief and Safety Valve Test Requirements," L-86-114, March 18, 1986.

13. Letter E. G. Tourigny, NRC, to C. O. Woody, Florida Power and Light Co., "Request for Additional Information re Response to TMI Action Item II.D.1, Relief and Safety Valves Test Requirements," June 10, 1987.

14. Letter C. O. Woody, Florida Power and Light Co., to USNRC Document Control Desk, "St. Lucie Unit Nos. 1 and 2, Docket Nos. 50-335 and 50-389, TMI Action Item II.D.1, Request for Addition Information," L-87-446, November 6, 1987.

15. Letter C. O. Woody, Florida Power and Light Co., to USNRC Document Control Desk, "St. Lucie Units. 1 and 2, Docket Nos. 50-335 and 50-389, TMI Action Item II.D.1, Request for Addition Information," L-88-59, February 5, 1988.

16. Summary Report on the Operability of Power Operated Relief Valves in CE Designed Plants, CEN-213, June 1982.

17. EPRI Summary Report: Westinghouse Gate Valve Closure Testing Program, prepared by Westinghouse Electro-Mechanical Division, March 31, 1982.

18. Summary Report on the Operability of Pressurizer Safety Valves in CE Designed Plants, CEN-227, December 1982.

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

20. EPRI PWR Safety and Relief Valve Test Program, Guide for Application of Valve Test Program Results to Plant-Specific Evaluations, Rev. 2, Interim Report, July 1982.

SALP INPUT

FACILITY NAME: St. Lucie 1 and 2

SUMMARY OF REVIEW/INSPECTION ACTIVITIES

Relief and Safety Valve Testing (Tac Nos. 44617 and 51605)

Review of Primary relief and safety valve testing. In addition to generic testing program, NUREG-0737, Item II.D.1 required licensee to assess the applicability of tests to plant-specific configuration including analysis of attached discharge piping.

NARRATIVE DISCUSSION OF LICENSEE PERFORMANCE - FUNCTIONAL AREA

SAFETY ASSESSMENT/QUALITY VERIFICATION

AUTHOR: C. G. Hammer

DATE: 1/9/89

1. Assurance of Quality, Including Management Involvement and Control
   The content of the submittals indicates that there was significant attention given to this TMI item. However, there appeared to be a lack of understanding or appreciation of the detail or nature of the needed information in the initial submittal. Responses to two rounds of staff questions provided most of the needed additional information; however, there remain significant items not fully resolved.