ENCLOSURE

EGG-NTA-7739
Rev. 1

TECHNICAL EVALUATION REPORT
TMI ACTION--NUREG-0737 (II.D.1)
RELIEF AND SAFETY VALVE TESTING
CRYSTAL RIVER UNIT 3
DOCKET NO. 50-302

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August 1988

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Prepared for the
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555
Under DOE Contract No. DE-AC07-76ID01570
FIN No. A6492

8810040039 26pp. XA

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 Crystal River Unit 3 Licensee response to the requirements of NUREG-0578 and NUREG-0737. This review found the Licensee has not provided an acceptable response and, thus has not reconfirmed that 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

CONTENTS

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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 Corp. (FPC), the owner of Crystal River Unit 3 (CR-3), 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 Babcock & Wilcox (B&W) to produce a report on the inlet fluid conditions for pressurizer safety and relief valves in B&W designed plants (Reference 7). Since CR-3 was designed by B&W, 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/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 o provide test data to the NRC.

3. PLANT SPECIFIC SUBMITTAL

A preliminary assessment of the adequacy of the overpressure protection system was submitted by FPC on July 01, 1981 (Reference 10) and August 7, 1981 (Reference 11). Additional assessment of the Pressurizer Safety and Relief Valve Piping was transmitted March 31, 1982 (Reference 12), June 30, 1982 (Reference 13) and November 1, 1982 (Reference 14). A request for additional information (Reference 15) was submitted to FPC by the NRC on October 18, 1984. FPC responded to this request on February 17, 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

CR-3 utilizes two safety valves, one PORV, and one block valve in the overpressure protection system. Both safety valves are Dresser Model 31739A. The PORV is a Dresser Model 31533VX-30. The block valve is a 2-1/2 in. Velan bolted bonnet gate valve with a Limitorque SMB-00-10 operator. There are no loop seals between the pressurizer and the safety valves or the PORV. Also, each valve is connected separately with the quench tank.

The Dresser 31739A safety valve used at CR-3 was one of the valves tested by EPRI; therefore the EPRI test results are directly applicable to the CR-3 safety valves.

The Dresser PORV installed at CR-3 has dash 2 internals (31533VX-30-2) and a bore diameter of 1-5/32 in. The test valve was also a dash 2 design but with a bore size of 1-5/16. The dash 2 design resulted from a need to improve the seat tightness and included modifications to the internals, the body, and the inlet flange. The body and flange modifications were not of a nature that would affect operability. The difference in bore diameter will only affect capacity and not operability. The test valve is, therefore, considered an adequate representation of the in-plant valve.

The Velan block valve used at CR-3 is a 2 1/2 in. bolted bonnet gate valve and with a Limitorque SMB-00-10 operator. Two Velan valves, both 3 in. gate valves, Model B10-3954-13MS, were tested by EPRI (Reference 9). One was tested with a Limitorque operator SB-00-15 and the other tested with a Limitorque operator SMB-000-10. FPC, in their submittal (Reference 16), compared the CR-3 block valve with both the EPRI test valves and TMI-2 block valve. It stated that the CR-3 block valve is not appreciably different from the valves tested by EPRI and is similar to the TMI-2 block valve. In addition, the CR-3 block valve operated during the February 26, 1980 transient. The 3 in. EPRI test valve requires a larger force to operate and the SMB-000-10 operator is a smaller operator with the same starting torque as the plant valve, so the tests with this operator on a 3 in. valve are a conservative demonstration of the operability of the plant valve.

Based on the above, the valves tested are considered to be applicable to the in-plant valves at CR-3 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 B&W 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. Reference 7 addresses those transients listed in Regulatory Guide 1.70, Rev. 2, which potentially challenge the PORV or safety valves in B&W plants. The conditions in the report that are applicable to CR-3 are those identified for B&W 177-FA plants.

For the safety valves, only steam discharge was calculated for FSAR type transients. The peak pressure was 2677 psia and the maximum pressurization rate was 175 psi/s. According to Reference 17, the maximum backpressure developed during FSAR accidents and transients for CR-3 is 520 psia. Since CR-3 does not have loop seals upstream of the safety valves, testing of the Dresser safety valves with the short inlet piping is applicable.

Eight applicable steam tests (Tests 316, 318, 320, 322, 324, 326, 328, and 1104a) with a short inlet pipe were performed with the 31739A valve which had a peak pressure of 2720 psia and a peak pressurization rate of 333 psi/s. The ring settings for these tests were (-48, -40, +11 and -48, -60, +11) bound those for the CR-3 valves (-48, -50, +11). Test 320 had a backpressure of 866 psia. The highest backpressure for the other seven tests was 676 psia. These conditions bound those expected at CR-3.

For extended HPI events (which include feedwater line breaks and steam line breaks) the safety valves will initially open on steam with transition to subcooled water calculated. A peak pressure of 2515 psia was calculated with liquid temperatures ranging from 400 to 640°F. A peak liquid surge rate of 11,520 lbm/min (at 640°F) will occur. Pressurization rates from 0 to 65 psi/s are expected.

For the 31739A valve, testing included a steam to water transition test at 2489 psia and saturated conditions. Three water tests at pressures ranging from 2389 to 2749 psia and with water temperatures of 414 to 608°F were run. During these tests, the 31739A valve passed at least 1128 GPM (~8,000 lbm/min) with 539°F water and 2492 GPM (~16,000 lb/min) with 649°F. The transition and water tests were run with pressurization rates from 1.8 to 3.2 psi/s. Although these represent the lower end of the range of pressurization rates calculated for B&W plants, they are adequate to represent expected inlet conditions at CR-3. These conditions are sufficiently close to the conservatively selected bounding conditions to adequately demonstrate valve performance.

For the PORV, FSAR events result only in steam discharge. Although Reference 7 indicated the PORV should be tested at a peak pressure higher than the opening set point, 2465 psia, the valve opens quickly enough that the increase in pressure during the opening cycle is minimal. Additionally, the peak pressure listed in Reference 7 was based on an analysis in which the PORV was assumed to be inoperable. Testing with saturated steam at set pressure is, therefore, considered adequate. The Dresser PORV is a pilot operated valve and the backpressure developed at the outlet is of potential importance to valve operability. The ability of the valve to operate at backpressures at least as high as those expected in service should be demonstrated. The expected backpressure for the PORV was not reported by FPC. However, the PORV discharge pipe routing is similar to the safety valves. The PORV rated flow, 100,000 lb/h, is <30% of the rated flow of the safety valve, 317,973 lbm/h. The 4 inch discharge pipe of the PORV has approximately 44% the flow area of the 6 inch pipe for the safety valves. From these data the conclusion is reached that the expected backpressure for the PORV is less than the 520 psia which bounds the safety valve. Testing of the valve (Reference 8) included numerous steam tests with opening pressures close to the CR-3 set pressure and backpressures as high as 760 psia which adequately bound the expected conditions for the PORV.

For extended HPI events (which include feedwater line breaks and steam line breaks) the initial opening of the PORV will be on steam but subcooled liquid could follow. HPI events can, therefore, result in steam to water transition and water (400 to 650°F) discharge at a maximum pressure of 2500 psia (Reference 7). A steam to water transition test and liquid tests with temperatures ranging from 447 to 647°F and pressures of approximately 2500 psia were included in the test series. The tests were run using the same discharge pipe orifice which developed backpressures ranging from 175 to 415 psia for the steam tests so that the expected backpressure was adequately represented. The HPI events were, therefore, adequately represented by the tests.

The PORV is used for low temperature overpressure protection (LTOP). For LTOP events, the valve is required to open on 565 psia steam. Reference 7 indicates transition and water flow will not occur at CR-3 during low temperature overpressurization events. Opening on steam is considered to be adequately represented by the full pressure steam tests discussed above.

For the block valve only full pressure steam, 2480 psia, tests were performed (Reference 9). The block valve, however, is required to open and close over a range of steam and water conditions. The required torque to open or close the valve depends almost entirely on the differential pressure across the valve disk and is rather insensitive to the momentum loading. Therefore, the required torque is nearly the same for water or steam and nearly independent of the flow. The full pressure steam tests, therefore, are adequate to demonstrate operability of the valve for low pressure steam and the required water conditions. The TMI-2 valve is similar to the CR-3 valve and its operability during the TMI-2 accident can be used to evaulate the operability of the CR-3 block valve. In addition, the CR-3 block valve operated during the February 26, 1980 event. This can also be used to evaluate the valve's operability.

The test sequences and analyses described above demonstrate that the test conditions bounded the conditions for the plant valves. The test results on safety valve, PORV and block valve plus the operating experience of the block valve and 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

The CR-3 safety valves (Dresser 31739A) were tested by EPRI and the test conditions enveloped the expected CR-3 valve conditions as discussed Section 4.2. The valve ring setting used at CR-3 (-48, -50, + 11) are bounded by the two ring settings used in the eight short inlet pipe tests. In seven of the eight tests (all but Test 320), the test valve functioned acceptably. Test 320 had an excessively high back pressure (866 psia) and the valve did not reach rated lift or flow rate. Because the peak backpressure in the other tests, 676 psia, still bounds the expected backpressure at CR-3, 520 psia, and the valve operated acceptably, the CR-3 safety valves are expected to operate acceptably.

Blowdowns for the eight Dresser 31739A safety valves tested by EPRI ranged from 7.0 to 16.9% so that the measured blowdown generally exceeded the design blowdown of 5%. A B&W analysis (Reference 18) has shown blowdown up to 20% does not impede natural circulation due to hot leg voiding. Therefore, having the observed blowdown exceed the design blowdown is considered acceptable.

The maximum bending moment applied to the discharge flange of the Dresser 31739A test valve during the eight applicable tests was 230,913 in-lb. Valve operability was not impaired by the application of this moment. The maximum moment computed by FPC for the CR-3 safety valve is 25,099 in-lb (Reference 16). However, this moment does not include seismic loading or all the FSAR transient conditions. Therefore, the maximum expected moment on the plant valve may not be bounded.

For the test performance to be a valid demonstration of plant safety valve stability, the test inlet piping must have a pressure difference at least as great as the plant. The plant valves are mounted directly on a pressurizer nozzle and thus have the minimum pressure drop possible, Therefore, the plant valves should be as stable as the test valve.

During the 414°F water test (Test 1114) the 31739A valve was stable but only achieved partial lift. The valve did not pass enough flow to prevent the test pressure from accumulating. However, the amount of liquid the valve discharged was more than the amount predicted to be discharged during a steam line break at 400°F. In addition, there are two safety valves at the plant, which gives CR-3 more than sufficient relief capacity. Under conditions typical of the FWLB, 2515 psia and water flow at temperatures of 602 and 640°F, the 31739A test valve on the short inlet configuration passed the required flow in two out of three tests and over 80% of the required flow during the third test. CR-3 has sufficient relief capacity at these conditions because two valves are installed at the plant.

Based on the test results discussed above, demonstration of safety valve operability is considered adequate. However, since the maximum expected moment under all possible transients was not provided, safety valve operability was not adequately demonstrated since valve performance might be reduced by excessive moments on the inlet and/or outlet flanges.

The Dresser PORV opened and closed on demand for all nonloop seal tests. Inspection of the valve after testing at the Marshall Steam Station showed the bellows had several welds partially fail. The failure did not affect valve performance and the manufacturer concluded the failure did not have a potential impact on valve performance. The bellows was replaced and did not fail during any of the additional test series.

A bending moment of 25,500 in-lb was induced on the discharge flange of the test valve without impairing operability. The maximum bending moment calculated for the CR-3 PORV is 15,552 in-lb (Reference 16). However, this moment does not include seismic loading or all the FSAR transient conditions. Therefore, the maximum expected moment on the plant valve may not be bounded.

The CR-3 PORV is a pilot operated valve that uses system pressure to hold the disk tight against the seat. At one point Dresser Industries recommended the block valve be closed at system pressures below 1000 psig to avoid steam wirecutting of the PORV disk and seat. Testing by Dresser later showed the 1000 psig pressure limit to be overly conservative and that the PORV as designed was qualified to system pressures of 50 psig and below. Below 50 psig Dresser recommends that the PORV block valve be closed to prevent leaking. In addition, Dresser provides heavier springs to be used under the main and pilot disks to ensure closure if the plant is to operate below 50 psig. However, leakage is not a problem at all plants, and the plant start up procedures for Crystal River 3 (Paragraph 6.4.6.12) require that the valve be cycled twice at 205-215 psig to ensure its proper operability. Failure of the PORV to operate properly will force the plant to remain at that pressure until a decision is made on how to restore the valve to its proper condition (Reference 16).

The valve performance during EPRI tests, under the full range of expected inlet conditions, and the CR-3 start-up procedures, demonstrate that the PORV is capable of discharging the required steam and liquid flow rates. However, since the maximum expected moment under all possible transients was not provided, PORV operability was not adequately demonstrated since valve performance might be reduced by excessive moments of the inlet and/or outlet flanges.

The PORV block valve must be capable of closing over a range of steam and water conditions. As described in Section 4.2, high pressure steam tests are adequate to bound operation over the full range of inlet conditions. The TMI-2 experience, the tests with the 3 in. Velan valve and SMB-000-10 operator, and the CR-3 operating experience all provide data to evaluate valve performance. The test valve was cycled successfully at full steam pressure with full flow. A similar valve/operator combination operated satisfactorily during the TMI-2 accident. In addition, the plant valve operated satisfactorily during a transient at CR-3. Based on the performance of the test valve, the TMI-2 valve, and the plant valve, the CR-3 PORV block valve is considered operable.

NUREG-0737 II.D.1 requires qualification of associated control circuitry as part of the safety/relief valve qualification. In Reference 15, the NRC requested information demonstrating that the PORV control circuitry is qualified. In Reference 16, FPC stated that qualification of the PORV and its control circuitry is not required by NRC regulation 10 CFR 50.49, and further stated that they did not believe that the NRC intended the PORV circuitry to be qualified under NUREG-0737 Item II.D.1. The FPC response is considered unsatisfactory because an electrical system malfunction initiated the February 26, 1980 transient that challenged the CR-3 high pressure injection system and a safety valve (Reference 16). The electrical system malfunction also produced a signal which opened the PORV and held its pilot valve operator open. This indicates the PORV is subject to spurious actuations which can challenge the plant safety systems. In addition, the response to question 2 in Reference 16 clearly states that the limiting inlet conditions for the PORV include extended HPI operation following an FSAR steam line break. This would expose the PORV to a harsh environment during which it could malfunction and cause additional challenges to plant safety systems. On the basis of the submittals by FPC, the PORV control circuitry is not considered qualified, and, therefore, does not satisfy the requirements of NUREG-0737 Item II.D.1.

The presentation above demonstrates that the valves operated satisfactorily and verifies the portion of Item 1 of Section 1.2 that requires conducting tests to qualify the valves was met. However, that part of Item 7 requiring the effect of discharge piping on operability be considered and Item 5 requiring qualification of the PORV control circuitry were not met.

4.5 Piping and Support Evaluation

In the piping and support evaluation, the safety/relief valve piping and supports between the valve discharge flanges and the pressurizer relief tank were analyzed for the requirements of the ANSI B31.1 Power Piping Code, 1967 Edition with code case N-7.

The thermal-hydraulic analysis was performed with the program RELAP4/MOD5. The THRUST code was used to generate fluid force histories from the RELAP4/MOD5 output. RELAP4/MOD5 was benchmarked against the output of RELAP5/MOD1 in Reference 16 and was shown to provide satisfactory thermal-hydraulic results. Furthermore, the ability of RELAP5/MOD1 to calculate the system thermal-hydraulic response was verified through simulations of EPRI/CE tests (Reference 21). Verification of the TRUST code was also provided in Reference 16. Therefore it can be concluded that RELAP4/MOD5 and the THRUST code will produce acceptable calculations of piping loads due to safety and relief valve discharge.

The Licensee stated that the thermal-hydraulic analysis for each individual valve and associated discharge piping was performed separately. Because the discharge piping of the safety valves and PORV do not join at a common header, the effects of simultaneous actuation of the valves is not an important consideration in the analyses for CR-3.

The transient conditions for which the piping and supports must be qualified should have been based on those summarized in Reference 7. The safety valves are required to pass steam, transition from steam to saturated water, and subcooled water. The bounding conditions from Reference 7 are:

Steam line break:

Safety valve set point = 2500 psig.
Steam flow, transition to saturated water, followed by subcooled water.
Liquid temperature range = 517 to 602°F.
Liquid insurge into pressurizer = 6555 lb/min at 602°F,
and 6019 lb/ min at 517°F.
Maximum steam discharge = 366,350 lb/h (based on EPRI tests)

Rod ejection accident at HZP:

Safety valve set point = 2575 psig.
Saturated steam discharge.
Maximum pressurizer pressure = 2662 psig.
Pressurization rate = 175 psi/s.

The bounding transients involving the PORV are presented in Reference 7. They include FSAR transients, extended HPI events, and low temperature overpressurization events that result in steam flow, transition from steam to saturated water, and subcooled water. Because the PORV was assumed to be inoperable during the analyses used to determine the inlet conditions presented in Reference 7, the bounding conditions for the PORV analyses are the same as those for the safety valves.

The forces generated from these conditions bound those from all other conditions expected at the plant.

In Reference 16, the Licensee stated that the safety valve piping was evaluated using the inlet conditions for the February 20, 1980 transient because subcooled water discharges are normally controlling due to the much higher discharge rate through the valve. The inlet conditions used in the analysis are:

Safety valve opening pressure = 2410 psig
Pressurization rate = 0, pressurizer maintained at 2410 psig
Safety valve inlet temperature = 560°F (subcooled water)
Liquid discharge rate = 1,544,280 lb/h
Safety valve opening time = 0.040 s

Based on the information presented in Reference 21, it is not clear that the liquid discharge analyzed provides the bounding loads on the piping and supports. Figure 2 in Reference 21 shows design forces computed in 1974 that are as high as those computed in 1980 with no explanation as to why the 1980 calculation are actually bounding. In addition, EPRI test data indicates that the safety valve opening times ranged from 0.007 to 0.043 s. Therefore, it can not be concluded that the bounding transients that provide the maximum expected loads on the piping and supports due to a safety valve discharge were considered.

In Reference 16, the Licensee stated that the thermal-hydraulic analysis for the PORV discharge piping was based on the following design conditions.

Pressurizer pressure = 2315 psia

Pressurizer temperature = saturated steam at 2315 psia

PORV flow rate = 117,000 lb/h

Pressurizer pressure = constant 2315 psia.

Subsequent to the analysis it was determined that the PORV flow rate is 165,900 lb/h. However, the analysis was not redone, because the Licensee assumed the piping loads were proportional to flow rate.

Again, few details were provided by FPC on the calculations performed. It did not appear that a liquid discharge case was analyzed for the PORV piping. According to FPC this would produce the maximum piping and supports loads.

The piping structural analysis was performed using the computer code PIPDYN II. This code was originally developed by the Franklin Institute. The Licensee did not provide verification of this code to the EPRI test data. It was stated that the code was verified to an alternate structural problem with a published solution; however, data comparing the results of these analyses were not provided. The Licensee stated that the results of the piping load cases that were analyzed were combined by absolute summation. This is generally more conservative than the SRSS combination method recommended by EPRI for combining transient and seismic loads. The stress information provided implies that the results will envelope the load cases recommended by EPRI in Reference 19. However, complete information concerning the piping model and results for all recommended EPRI load cases were not provided. Thus, structural adequacy for all load cases could not be confirmed.

The information supplied indicates only that the piping supports were evaluated. It was stated that two supports would require modification to withstand the loads imposed during the 2-26-80 transient (which is used as the limiting transient). However, because this load was not considered a design condition and because no physical damage was apparent after the Febrary 26, 1980 event, no modifications were made. Detailed information was not provided regarding analytical methods utilized for the support analyses. The Licensee has not demonstrated support structural adequacy for all load cases recommended by EPRI.

Based on the information provided by FPC, it cannot be concluded that a bounding case was chosen for the piping and support evaluation and that the thermal-hydraulic and structural analyses were adequate to qualify the pressurizer piping and supports. Therefore, Items 3 and 8 of Section 1.2 were not met.

5. EVALUATION SUMMARY

The Licensee for Crystal River Unit 3 has not provided an acceptable response to the requirements of NUREG-0737, which would reconfirm that the General Design Criteria 14, 15, and 30 of Appendix A to 10 CFR 50 were met with regard to the safety valves and PORV. 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, 2, 4, 6, and part of Item 7 in Section 1.2).

The Licensee participated in the development and execution of an acceptable relief and safety valve test program to qualify the operability of prototypical valves and to demonstrate that their operation would not invalidate the integrity of the associated equipment and piping. The subsequent tests were successfully completed under inlet conditions which, by analysis, bound the most probable maximum forces expected from anticipated design basis events. The test results showed that the valves tested functioned correctly and safely for all steam and water discharge events specified in the test program that were applicable to Crystal River Unit 3 and that the pressure boundary component design criteria were not exceeded. Analysis and review of both the test results and the Licensee justifications indicated the performance of the prototypical valves and piping can be extended to the in-plant valves and piping.

Therefore, the prototypical tests and the successful performance of the valves demonstrated that this equipment was constructed in accordance with high quality standards, meeting 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 3: Item 3, which requires the dynamic forces on the safety valves and PORV be maximized, was not met. The safety valve piping was only analyzed for water flow conditions and the PORV piping for steam discharge conditions. Based on the information supplied by FPC, it is not clear the dynamic forces on the piping system were maximized using these transient conditions.

Item 5: Item 5, which requires the PORV control circuitry be qualified, was not met. FPC stated in its submittals that it did not believe that the NRC intended the PORV circuitry to be qualified under NUREG-0737, Item II.D.1. However, this is indeed the NRC staff position on this item. The FPC response is considered unsatisfactory because an electrical system malfunction initiated a plant transient that opened the PORV and held its pilot valve operator open. This indicates the PORV is subject to spurious actuations. In addition, FPC's submittal clearly states the limiting inlet conditions for the PORV include extended HPI operation following an FSAR steam line break. This would expose the PORV to a harsh environment during which it could malfunction and challenge plant safety systems.

Item 7: That part of Item 7 that requires consideration of the effect of as-built discharge piping on safety valve and PORV operability was not met. This is because the maximum expected bending moment on the CR-3 safety valves and PORV, as supplied by the Licensee, did not include seismic loads or all the FSAR transient conditions. Therefore, the maximum expected moment on the plant valves may not be bounded. Thus, operability of the safety valves and PORV with the maximum expected applied moment could not be assured.

Item 8: Item 8, which requires qualification of the piping and supports, was not met. This is because, based on the information provided by FPC, it cannot be concluded that the thermal-hydraulic and structural analyses were adequate to qualify the pressurizer piping and supports. Also, sufficient information on the verification of the structural analysis code was not presented.

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).

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