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TECHNICAL EVALUATION REPORT
TMI ACTION--NUREG-0737 (II.D.1)
MILLSTONE NUCLEAR POWER STATION, UNIT 3
DOCKET NO. 50-423

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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 systems. As a result, the authors of NUREG-0578 (TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations) and subsequently NUREG-0737 (Clarification of TMI Action Plan Requirements) recommended that programs be developed and completed which would reevaluate the functional performance capabilities of Pressurized Water Reactor (PWR) safety, relief, and block valves and which would verify the integrity of the piping systems for normal, transient, and accident conditions. This report documents the review of these programs by the Nuclear Regulatory Commission (NRC) and their consultant, EG&G Idaho, Inc. Specifically, this review examined the response of the Licensee for the Millstone Nuclear Power Station, Unit 3, to the requirements of NUREG-0578 and NUREG-0737 and finds that the Licensee provided an acceptable response, reconfirming that the General Design Criteria 14, 15, and 30 of Appendix A to 10 CFR 50 were met.

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TECHNICAL EVALUATION REPORT

TMI ACTION--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 a basic unreliability of the valve design. It is known that the failure of a power-operated relief valve (PORV) to reseat was a significant contributor to the Three Mile Island (TMI-2) sequence of events. These facts led the task force which prepared NUREG-0578 (Reference 1) and, subsequently, NUREG-0737 (Reference 2) to recommend that programs be developed and executed which would reexamine the functional performance capabilities of Pressurized Water Reactor (PWR) safety, relief, and block valves and which would verify the integrity of the piping systems for normal, transient, and accident conditions. These programs were deemed necessary to reconfirm that the General Design Criteria 14, 15, and 30 of Appendix A to Part 50 of the Code of Federal Regulations, 10 CFR, are indeed satisfied.

1.2 General Design Criteria and NUREG Requirements

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

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

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

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

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

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

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

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

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

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

2. PWR OWNERS' GROUP RELIEF AND SAFETY VALVE PROGRAM

In response to the NUREG requirements previously listed, a group of utilities with PWRs requested the assistance of the Electric Power Research Institute (EPRI) in developing and implementing a generic test program for pressurizer power operated relief valves, safety valves, block valves, and associated piping systems. The Northeast Nuclear Energy Co. (NNECO), owner of the Millstone Nuclear Power Station, Unit 3, was one of the utilities sponsoring the EPRI Valve Test Program. The results of the program are contained in a group of reports which were transmitted to the NRC by Reference 3. The applicability of these reports 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. Representative valves were selected for testing with a sufficient number of the variable characteristics that their testing would adequately demonstrate the performance of the valves used by utilities (Reference 5). EPRI, through the Nuclear Steam Supply System (NSSS) vendors, evaluated the FSARs of the participating utilities and arrived at a test matrix which bounded the plant transients for which overpressure protection would be required (Reference 6).

EPRI contracted with Westinghouse Electric Corp. to produce a report on the inlet fluid conditions for pressurizer safety and relief valves in Westinghouse designed plants (Reference 7). Since Millstone, Unit 3, was designed by Westinghouse this report is relevant to this evaluation.

Several test series were sponsored by EPRI. PORVs and block valves were tested at the Duke Power Company Marshall Steam Station located in Terrell, North Carolina. Additional PORV tests were conducted at the Wyle Laboratories Test Facility located in Norco, California. Safety valves were tested at the Combustion Engineering Company, Kressinger Development Laboratory, located in Windsor, Connecticut. The results 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 (a) obtain valve capacity data, (b) assess hydraulic and structural effects of associated piping on valve operability, and (c) obtain piping response data that could ultimately be used for verifying analytical piping models.

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

3. PLANT SPECIFIC SUBMITTAL

The plant specific evaluation of the adequacy of the overpressure protection system for Millstone, Unit 3, was submitted by NNECO to the NRC on October 1, 1985 (Reference 11). The NRC transmitted a request for additional information to the utility on March 18, 1987 (Reference 12) to which NNECO responded on July 21, 1987 (Reference 13). Reference 14, on August 28, 1987, provided the Licensee's final response to questions on the plant backpressure in the safety and relief valve discharge piping and the piping thermal hydraulic and structural analyses.

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

Millstone, Unit 3, is a four-loop PWR designed by the Westinghouse Electric Co. It is equipped with three (3) safety valves, two (2) PORVs, and two (2) PORV block valves in its overpressure protection system. The safety valves are 6-in. Crosby Model HB-BP-86, 6M6, spring loaded valves with steam internals. (The original safety valves had loop seal internals and were later modified to use steam internals of the Crosby flexi-disc design.) The design set pressure is 2500 psia and the rated steam flow capacity is 420,000 lbm/h. The PORVs are 3-in. Garrett, straight-through, power operated relief valves, Garrett Part No. 3750014. The PORV opening set pressure is 2350 psia and the rated steam flow capacity is 210,000 lbm/h. The PORV block valves are 3-in. ALOYCO Model N-6225-EMO-SP motor operated gate valves with Limitorque SMB-00-25 motor operators. The inlet piping to the safety valves includes a drained loop seal; the PORV inlets have no loop seals.

A Crosby 6M6 safety valve identical to those installed at Millstone, Unit 3, was tested by EPRI. The test results from this valve are directly applicable to the corresponding valves in Millstone, Unit 3.

The Garret PORV tested by EPRI was a Model 3224718-2 solenoid controlled PORV with straight-through body configuration. This valve was specifically designed for the EPRI tests to represent the two production models of the valve, one of which is the Model 3750014 PORVs installed at Millstone, Unit 3. The only differences between the test valve and the in-plant valves are in the cage and seat flow areas, inlet and outlet pipe sizes, and housing configuration. These factors only affect the relief capacity of the valve and do not affect valve performance. The test valve used a Garrett developmental solenoid that had the same size electromagnetic coil, the same switching and override mechanisms, and similar body style as the production units. However, the test solenoid used materials that were different from the production solenoid. These differences in materials do not affect the operability of the solenoid. Therefore the test results of the Garrett PORV are considered applicable to Millstone, Unit 3.

The Millstone PORV block valve was not tested by EPRI. The Licensee provided the results of in-situ testing of the plant specific block valve to demonstrate the adequacy of the valve.

Therefore, those parts of the criteria of Item 1 and 7 as identified in Section 1.2 of this report regarding applicability of the test valves are fulfilled.

4.2 Test Conditions

As stated above, Millstone, Unit 3, is a four-loop PWR designed by the Westinghouse Electric Corp. The valve inlet fluid conditions that bound the overpressure transients for Westinghouse designed PWR plants are identified in Reference 7. The transients considered in this report include FSAR, extended high pressure injection, and cold overpressurization events. The expected fluid conditions for each of these events and the applicable EPRI tests are discussed in this section.

4.2.1 FSAR Steam Transients

For Millstone, Unit 3, the limiting events for FSAR transients resulting in steam discharge through the safety valves alone and in steam discharge through both the safety and relief valves are the loss of load event (for maximum pressurizer pressure) and the locked rotor event (for the maximum pressurization rate).

When the safety valves actuate alone, the maximum pressurizer pressure and maximum pressurization rate are predicted to be 2555 psia and 144 psi/s, respectively. The maximum developed backpressure in the outlet piping is 466 psia. The plant valve rings are set at the factory ring settings recommended by Crosby, the valve manufacturer.

Among the tests conducted by EPRI on the Crosby 6M6 safety valve, only one test (Test No. 1411) is directly applicable to the plant specific steam discharge condition. This test was performed with a drained loop seal and with Crosby recommended ring settings. This is representative of the plant drained loop seal inlet piping configuration and factory set ring settings. The peak tank pressure was 2664 psia; the pressurization rate was 300 psi/s and the peak backpressure was 245 psia. To envelope the plant backpressure of 466 psia, a cold loop seal test, Test 929, will be used. The cold loop seal test can be used because the maximum backpressure is not developed until the loop seal has been discharged and full steam flow developed. In Test 929, the peak backpressure was 710 psia. The pressurization rate in this test, 319 psi/s, and the peak tank pressure, 2726 psia, also bound the plant conditions.

For FSAR transients resulting in steam discharge through both the safety valves and PORVs, the maximum pressure predicted is 2532 psia and the maximum pressurization rate is 130 psi/s. These fluid parameters represent the limiting condition for steam discharge through the PORVs at this plant.

The Garrett test PORV was subjected to thirteen steam tests in the EPRI testing program. In these tests, the maximum pressure at the valve inlet was 2760 psia which bounds the predicted maximum pressure of 2532 psia. The highest backpressure developed in the discharge pipe was 825 psia in the full pressure steam tests performed at the Marshall Facility. The backpressures developed during the two steam tests conducted at Wyle Laboratories were 580 and 623 psia. These test values are higher than the backpressure of 428 psia predicted for the Millstone, Unit 3, PORVs. Therefore the EPRI test inlet fluid conditions for the PORV in steam discharge are representative of the plant specific transient conditions.

4.2.2 FSAR Liquid Transients

For most Westinghouse plants, the limiting FSAR transient resulting in liquid discharge through the PORVs and safety valves is the main feedline break accident (Reference 7). However, in Reference 13, the Licensee stated that the feedline break analysis in the Millstone, Unit 3, FSAR, Section 15.2.8, showed that the pressurizer does not fill for the cases analyzed, which included both offsite power available and the loss of offsite power. Therefore, water relief through either the safety valves or the PORVs is not required for this accident.

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. Both the safety valves and the PORVs may be challenged by steam and liquid discharge. When the safety valves actuate, the maximum pressure is predicted to be 2507 psia and the liquid temperature ranges from 567 to 572°F. When the PORVs discharge, the maximum pressure is predicted to be 2353 psia with liquid temperatures between 565 and 569°F (Reference 7). The pressurization rate in both cases is within 4 psi/s. The steam discharge conditions for the safety valves and PORVs are bounded by the FSAR steam transient conditions 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

The PORV is used for low temperature overpressure protection (LTOP) during reactor start-up and shutdown operations. The PORV control is put in the LTOP mode whenever the reactor coolant temperature falls below 350°F. In the LTOP mode, the PORV setpoint varies as a function of the reactor coolant temperature. The possible fluid states at the PORV inlet include steam, steam to water transition, and water conditions. The steam discharge condition is bounded by the high pressure set point of 2350 psia at 650°F. The maximum pressure at the PORV inlet during water discharge is predicted to be 750 psia at temperature ranging from 70 to 350°F. The steam to water transition pressure ranges from 455 to 2350 psia with the fluid at the saturation temperature.

For steam discharge through the PORV, the high pressure steam tests discussed in Section 4.2.1 would cover the low pressure steam conditions predicted for a low temperature overpressurization transient. For water discharge conditions, there were two low pressure and low temperature water tests performed on the Garrett PORV. The tests were conducted at pressures of 683 and 686 psia and water temperatures of 104 and 447°F, respectively. These tests were performed at a slightly lower pressure than predicted for the in plant valves, however, they do demonstrate the operability of the valves at comparable pressures. In addition, there was a transition test performed at 2760 psia and 682°F. These tests bound the low temperature overpressure transient at Millstone, Unit 3. The EPRI test results can therefore be used to evaluate the performance of the in-plant PORV in the low temperature overpressurization transient.

4.2.5 PORV Block Valve Fluid Conditions

The ALOYCO PORV block valve used at Millstone, Unit 3, was not tested by EPRI. NNECO tested the Millstone, Unit 3, PORV block valves in-situ. The conditions were saturated steam at 2248 psia. The use of an in-situ test is considered adequate to meet the requirements of NUREG-0737, Item II.D.1 because the intent of the NUREG was to qualify the valve through testing. In-situ testing or participation in the EPRI test program is adequate.

4.2.6 Test Conditions Summary

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 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 that requires showing the test conditions are equivalent to conditions prescribed in the FSAR was also met.

4.3 Operability

4.3.1 Safety Valves

As discussed in Section 4.2, the representative EPRI test for the steam discharge condition for the Millstone, Unit 3, safety valves was the drained loop seal test or the Crosby, 6M6, safety valve, Test No. 1411. The test valve opened within -4% of the set pressure (2410 psia) and closed with 8.2% blowdown. The valve performance was stable. Rated flow was exceeded at 3% accumulation (107%) and the maximum valve position was at 93% of rated lift. In Test 929, used to bound the valve behavior in a test with a backpressure greater than that predicted for the plant valves, the valve had stable performance on steam and closed with 5.1% blowdown. This adequately demonstrates valve operability at high backpressures.

The blowdown in Test 1411 was in excess of the 5% value specified by the valve manufacturer and the ASME Code. Thus, it must be demonstrated that these extended blowdowns will not impact plant safety and valve operability. Filling the pressurizer is not a concern from a valve operability standpoint since the Crosby 6M6 safety valves at Millstone, Unit 3, were shown in the EPRI tests to be operable with steam, steam/water transition, and water inlet conditions. Blowdown of 8.2% from a valve setpoint of 2500 psia should not present a challenge to plant protection equipment and, therefore, was not considered a safety concern. A second concern with extended blowdown is the possibility of voiding in the primary coolant system causing a significant loss of decay heat removal capability. In order to resolve this concern about plant safety, data from the Semiscale natural circulation (NC) test series was reviewed (Reference 15). This data applies to plants with U-tube steam generators. The NC test series showed that the various modes of NC (single-phase, two-phase, and reflux) were able to keep the core cool at decay heat levels of 1.5 to 5% and at primary system mass inventories of 100 to 55%. This indicates that if any voiding of the primary due to extended blowdown should occur, it would not endanger the core because forced circulation (early in the transient) and NC (late in the transient) would continue to remove the decay heat.

The inlet piping pressure differences on valve opening and closing were calculated for the Millstone, Unit 3, 6M6 safety valves and the 6M6 test valve. The pressure differences predicted for Millstone, Unit 3, are 246 psi during valve opening (pressure drop) and 144 psi during valve closing (pressure rise) which are lower than the pressure drop and pressure rise values of 263 and 181 psi for the test inlet configuration. Therefore, the plant valves should be as stable as the test valve.

The maximum bending moment on the plant safety valves was calculated to be 16,718 ft-lb. The maximum induced bending moment on the test valve was 24,896 ft-lb. Because the bending moment from the EPRI tests bounds the value calculated for the plant valves and the operation of the test valve was not affected by the induced bending moment, operability of the plant valves with the predicted bending moment was demonstrated.

4.3.2 Power Operated Relief Valves

The EPRI tests applicable to the Millstone, Unit 3, PORVs indicated that the test valve opened and closed on demand for all applicable steam tests. The test valve closed at 2310 and 2240 psia (for Tests 97 and 98 respectively) which was slightly lower than the required minimum closing pressure of 2320 psia. The lowest steam flow rate observed in the EPRI tests was 292,000 lbm/h which exceeds the rated flow of 210,000 lbm/h for the Millstone, Unit 3, PORVs.

In the reduced pressure water tests, which simulated the low temperature overpressurization transients, the test valve opened on demand and closed at pressures above the minimum predicted pressure of 455 psia.

The valve stroke times during the EPRI tests ranged from 0.24 to 1.80 s during the opening cycles and 0.58 to 1.95 s during the closing cycles which were within the required opening and closing times of 2.00 s.

The maximum bending moment induced on the discharge flange of the Garrett PORV during the EPRI tests was 33,200 in-lb. The operability of the test valve was not affected. In Reference 13, NNECO stated the maximum predicted bending moment for the PORVs was 115,992 in-lb. The PWR Safety and Relief Valve Test Program Valve Selection Justification Report (Reference 5) noted that the Garrett Straight-Through PORV was designed to operate with the maximum valve load. Garrett stated in Reference 5 that during a design analysis where a bending moment of 384,878 in-lb. was applied through the valve inlet the valve body deformation was 0.000116 inch. This compared to a minimum cage to body clearance of 0.012 inch. Consequently, even though the EPRI test program only subjected the Garrett test PORV to a 33,200 in-lb. bending moment, the plant valve is expected to operate with the higher induced moments expected during transient conditions. Similar reasoning was found acceptable in the Vogtle, Units 1 and 2, technical evaluation report (TER) (Reference 16).

4.3.3 Electric Control Circuitry

NUREG-0737, Item II.D.1, states that the control circuitry associated with the PORVs shall be qualified for design basis accidents and transients. Meeting the licensing requirements of 10 CFR 50.49 for this electrical equipment is considered satisfactory and specific testing per the NUREG-0737 requirements is not necessary. The Licensee for Millstone, Unit 3, included the PORV control circuitry in its 10 CFR 50.49 environmental qualification program (Reference 13), thus, satisfying the NUREG-0737 requirement.

4.3.4 PORV Block Valves

As noted in Section 4.2.5, the ALOYCO PORV block valve used at Millstone, Unit 3, was not tested by EPRI. NNECO tested the Millstone, Unit 3, PORV block valves in-situ and as discussed in Section 4.2.5 this approach is adequate. The conditions were saturated steam at 2248 psia. NNECO stated the PORV block valve was opened and then closed successfully against saturated steam flow conditions during the test. This test adequately demonstrated valve operability.

4.3.5 Operability Summary

The above discussion demonstrates the valves operated satisfactorily. This verifies the part of Item 1 of Section 1.2 that requires conducting tests to qualify the valves and the part of Item 7 that requires the effect of discharge piping on operability be considered were met. Also, the inclusion of the PORV control circuitry in the 10 CFR 50.49 program is considered to satisfy Item 5 of Section 1.2.

4.4 Piping and Support Evaluation

This evaluation covers the piping and supports upstream and downstream of the safety/relief valves extending from the pressurizer nozzle to the pressurizer relief tank. The piping was designed for deadweight, internal pressure, thermal expansion and transients, 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 evaluation are discussed below.

4.4.1 Thermal Hydraulic Analysis

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

A total of seven valve discharge cases were analyzed in the thermal hydraulic analysis; five cases involved steam discharge through the safety valves and PORVs and two cases involving water discharge through the PORVs. These analysis conditions were as follows.

Steam Discharge Conditions:

1. The two PORVs actuated simultaneously with the safety valves closed.
2. The three safety valves actuated simultaneously with the PORVs closed.
3. The PORVs and safety valves actuated in sequence. The three safety valves were assumed to open after the two PORVs had opened and reached steady state flow.
4. PORVs closed after reaching steady state flow with the pressurizer inlet pressure at 2400 psia.

5. PORVs closed after reaching steady state flow with the pressurizer pressure at 435 psia.

Water discharge conditions:

6. PORV opened with water discharge.

7. PORV closed after reaching steady state water discharge.

Cases 6 and 7 above simulated the low temperature overpressurization transient.

Analyses of the above valve actuation conditions were performed to generate the piping forces as a function of time. These forcing functions were used as input to the piping structural analysis computer code to calculate the piping loads and responses. The results showed that Case 2, the simultaneous opening of all safety valve on steam, produced the maximum piping forces on the upstream portion of the safety valve piping and the downstream piping from the safety and relief valves and Case 7, simulating the PORV closing during water discharge, produced the maximum forces on the piping upstream of the PORVs. For Case 2, the safety valves opened at 2500 psia and increased linearly to 2575 psia at a rate of 54 psi/s. The initial fluid conditions were saturated steam at 2500 psia. These conditions, except for the pressurization rate, bound those expected at the plant. The pressurization rate used in the thermal hydraulic analysis for Case 2, 54 psi/s, is less than the bounding, maximum pressurization given in Reference 7 for Westinghouse four loop plants, 144 psi/s. The overall analysis is still considered adequate, however, because the pressurization rate used is not expected to significantly affect the overall analysis results. Also, because the Millstone, Unit 3, safety valves do not have water loop seals the contribution of the valve discharge transient to the overall piping stresses and support loads is relatively small. The initial conditions for Case 7 were pressure of 800 psia and subcooled water at 350°F.

The thermal hydraulic analysis was performed using the Stone and Webster Engineering Corp. (SWEC) computer programs STEHAM for steam discharge conditions and WATHAM and WATAIRO for water discharge conditions. Adequate verification of the SWEC codes STEHAM, WATHAM, and WATAIRO was provided by NNECO in Reference 14. STEHAM computes fluid pressure, velocity, and density over time using the method of characteristics. Unbalanced forces are computed for bonded pipe segments by integrating the rate of change of the fluid momentum within the control volume. For open pipe segments, discharge blowdown forces are included. Time steps are selected internally based on input segment lengths and the instantaneous sound speed. The code was verified by Stone & Webster by comparing STEHAM results to those obtained with RELAP5/MOD1 and EPRI test results (both obtained from Reference 17). WATHAM is used to calculate waterhammer type problems and is based on the method of characteristics. WATHAM was verified by comparing the calculated results to hand calculated values for a sudden valve closure problem. Good agreement was obtained. WATAIRO calculates the one-dimensional transient flow field response and flow induced forcing functions in a piping system. The code uses a Runge-Kutta integration method to integrate the governing two-phase fluid flow equations. Hand calculations were used to verify WATAIRO and the comparison of the hand calculated values and the WATAIRO results were in excellent agreement. This information is sufficient to show the SWEC codes are adequate to model the thermal hydraulic portion of the valve discharge piping analysis.

The safety valve opening time used in the analysis of Case 2 was 0.02 s and the PORV closing time used in the analysis of Case 7 was 0.19 s. These times are representative of those found during the EPRI valve tests. The safety valve flow rate used in the analysis was adjusted to reflect the 90% derating of the safety valve required by ASME Code. The assumed valve flow rate was based on 111% of the rated flow. Time step size was on the order of 0.001 s. This time step size is consistent with the size used in the verification analyses. Node spacing used in the plant analysis also is consistent with that used in the verification analyses. Therefore, the thermal hydraulic analysis is considered adequate for predicting the safety/relief valve piping discharge loads.

4.4.2 Structural Analysis

The structural analysis of the safety/relief valve piping was performed using the NUPIPE-SW computer code. The NUPIPE computer code is a linear elastic piping structural analysis program widely used in the industry. It was shown to be a suitable tool for the static, dynamic, and thermal analyses of a piping system. Additional verification of the program was provided by comparing results with other programs benchmarked by the NRC.

The key structural analysis parameters of lumped mass spacing, integration time step, cutoff frequencies, and damping are adequate. Lumped mass spacing was selected so the model contained at least three mass points between restraints active in the same direction. Integration time steps less than 0.0008 were used to adequately address the input and to ensure a stable solution. The minimum cutoff frequency was 485 Hz. A damping factor of 1% was used. Therefore, the Millstone, Unit 3, structural analysis parameters are considered adequate.

The safety valve and PORV upstream piping were designed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, 1971 Edition, with Addenda through Summer 1973. The piping downstream of the safety valves and the PORVs is considered to be ANSI B31.1, Class 4, piping (1967 Edition with Addenda through Summer 1973). However, to protect QA Category I piping, the Licensee stated in Reference 13 that the piping downstream of the safety valves and the PORVs was qualified to ASME Section III, Class 3, piping standards in this evaluation. This is considered acceptable because, for these code editions and addenda, the ASME code for Class 3 piping is equivalent to the ANSI code. The piping stresses were calculated for the effect of normal operation load, earthquake (OBE and SSE), thermal expansion and transients, and piping discharge loads. The load combinations and allowables were taken from the Millstone, Unit 3, FSAR which is considered acceptable. The fluid discharge transient was combined with seismic stresses using the SRSS method and then added to all other stresses using absolute summation. The Licensee completed the piping analysis before the final piping erection. As a result, the Licensee installed the piping to meet the design requirements of this analysis.

The load combinations and allowables for the supports were taken from the Millstone, Unit 3, FSAR. According to the FSAR, the piping supports were designed in accordance with the AISC Manual of Steel Construction, Seventh Edition, except that the design limits of ASME Section III, Appendix XVII, 1974 Edition, were used for the faulted condition. This is conservative because the ASME code only allowed an increase of 1.2 times the basic allowable for the faulted condition whereas the AISC allows an increase of 1.33 times the basic allowable when a seismic load is included in the load combination. Use of load combinations and allowables from the plant FSAR is acceptable. The supports were designed by considering the service conditions separately using the allowable stress applicable to each individual service condition. The support design analysis was completed prior to the final piping and support erection. As a result, the Licensee installed the supports to meet the design requirements of this analysis.

Information supplied by NNECO in Reference 13 showed the piping met the applicable allowable at all locations. In Reference 14, the Licensee stated all support loads were also within code allowables. Therefore, the above analysis is adequate to qualify the plant specific piping and supports.

4.4.3 Piping and Support Summary

The discussion above demonstrates that a bounding loading case was chosen for the piping evaluation and verifies that Item 3 of Section 1.2 was met. The analysis of the piping and support system verifies that Item 8 was also met.

5. EVALUATION SUMMARY

The Licensee for Millstone, Unit 3, provided an acceptable response to the requirements of NUREG-0737, and thereby reconfirmed that the General Design Criteria 14, 15, and 30 of Appendix A to 10 CFR 50 were met. The rationale for this conclusion is given below.

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 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 plant specific piping and supports also were shown by analysis to be acceptable.

Thus, the requirements of Item II.D.1 of NUREG-0737 were met (Items 1-8 in Paragraph 1.2) and, thereby demonstrate by testing and analysis, that the reactor primary coolant pressure boundary will have a low probability of abnormal leakage (General Design Criterion No. 14) and that the reactor primary coolant pressure boundary and its associated components (piping, valves, and supports) were designed with sufficient margin such that design conditions are not exceeded during relief/safety valve events (General Design Criterion No. 15). Furthermore, the prototypical tests and the successful performance of the valves and associated components demonstrated that this equipment was constructed in accordance with high quality standards (General Design Criterion No. 30).

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