August 22, 1984

DCR 016

Docket No. 50-289

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Mr. Henry D. Hukill, Vice President
and Director - TMI-1
GPU Nuclear Corporation
P. O. Box 480
Middletown, Pennsylvania 17057

Dear Mr. Hukill:

We have completed our review of NUREG-0737, Item II.D.1 "Relief and Safety Valve Testing". Our Safety Evaluation (SE) is enclosed. The SE was prepared with the assistance of our contractor, Idaho National Engineering Laboratory (EG&G) based on the references in the SE, including the results of generic studies and tests by the Electric Power Research Institute and your plant-specific submittals.

We find that your submittals demonstrate the ability of the reactor coolant system Power Operated Relief Valve (PORV), PORV Block Valve, safety valves, and associated piping to function under expected operating conditions for design-basis transients and accidents as defined in NUREG-0737, Item II.D.1.

As noted in our SE, you committed to remove the loop seals upstream from the safety valves and mount the valves directly on the pressurizer nozzle (SE page 8), install a heavier spring on the PORV (SE page 13), add a snubber in the safety valve piping and modify three supports in the safety valve lines (SE page 18). These modifications have been made and, therefore, we consider Item II.D.1 complete.

Sincerely,

John F. Stolz, Chief
Operating Reactors Branch No. 4
Division of Licensing

Enclosure:
Safety Evaluation

cc w/enclosure:
See next page

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SAFETY EVALUATION REPORT

TMI ACTION--NUREG-0737 (II.D.1)

RELIEF AND SAFETY VALVE TESTING FOR

THREE MILE ISLAND UNIT 1

DOCKET NO. 50-289

1. INTRODUCTION

1.1 Background

Light water reactor experience has included a number of instances of improper performance of relief and safety valves installed in the primary coolant systems. There have been instances of valves opening below set pressure, valves opening above set pressure and valves failing to open or reseat. From these past instances of improper valve performance, it is not known whether they occurred because of a limited qualification of the valve or because of a basic unreliability of the valve design. It is known that the failure of a power-operated relief valve (PORV) to reseat was a significant contributor to the Three Mile Island (TMI-2) sequence of events. These facts led the task force which prepared NUREG-0578 (Reference 1) and, subsequently, NUREG-0737 (Reference 2) to recommend that programs be developed and executed which would reexamine the functional performance capabilities of Pressurized Water Reactor (PWR) safety, relief, and block valves and which would verify the integrity of the piping systems for normal, transient and accident conditions. These programs were deemed necessary to reconfirm that the General Design Criteria 14, 15, and 30 of Appendix A to Part 50 of the Code of Federal Regulations, 10 CFR are indeed satisfied.

1.2 General Design Criteria and NUREG Requirements

General Design Criteria 14, 15, and 30 require that (1) the reactor primary coolant pressure boundary be designed, fabricated, and tested so as to have an 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 (Reference 2), which was issued for implementation on October 31, 1980. As stated in the NUREG reports, each pressurized water reactor Licensee or Applicant shall:

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

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

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

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

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

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

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

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

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

In response to the NUREG requirements previously listed, a group of utilities with PWRs requested the assistance of the Electric Power Research Institute (EPRI) in developing and implementing a generic test program for pressurizer power operated relief valves, safety valves, block valves and associated piping systems. General Public Utilities Nuclear Corporation (GPUN), the owner of Three Mile Island Unit 1 (TMI-1), was one of the utilities sponsoring the EPRI Valve Test Program. The results of the program are contained in a group of reports which were transmitted to the NRC by Reference 3. The applicability of these reports are discussed below.

EPRI developed a plan (Reference 4) for testing PWR safety and relief valves under conditions which bound actual plant operating conditions. EPRI, through the valve manufacturers, identified the valves used in the overpressure protection system 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 over pressure protection would be required (Reference 6).

The utilities participating in the EPRI Safety and Relief Valve Test Program also obtained information regarding the performance of PORV block valves (Reference 9). A list of valves used or intended for use in participating PWR plants was developed. Seven block valves believed to be representative of the block valves utilized in the PWR plants were selected for testing. Additional tests were performed by Westinghouse Electro-Mechanical Division (WEMD) on valve models they manufacture (Reference 14).

EPRI contracted with Babcock and Wilcox Company (B&W) to produce a report on the inlet fluid conditions for pressurizer safety and relief valves in Babcock and Wilcox designed plants (Reference 7). Since TMI-1 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. Only steam tests were conducted at the Marshall Station. Block valves, therefore, were only tested for full flow, full pressure, steam conditions at Marshall. Water flow tests were performed by WEMD on four valve models they manufacture. Conditions ranged from 60 to 600 gpm and 1500 to 2600 psi differential pressure. Additional PORV tests were conducted at the Wyle Laboratories Test Facility located in Norco, California. Safety valves were tested at the Kressinger Development Laboratory which is part of the Combustion Engineering Test Facility located in Windsor, Connecticut. The results for the relief and safety valve tests are reported in Reference 8. The results for the block valves tests are reported in References 9 and 14.

The primary objective of the EPRI/C-E Valve test Program was to test each of the various types of primary system safety valves in pressurized water reactor plant service for the full range of fluid conditions under which they may be required to operate. The conditions selected for test (based on analysis) were limited to steam, subcooled water and steam to water transition. Additional objectives were to (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 requirement of Item 6 of Section 1.2 to provide test data to the NRC.

3. PLANT SPECIFIC SUBMITTAL

A preliminary assessment of the adequacy of the overpressure protection system was submitted by GPUN on April 16, 1982 (Reference 10). A final evaluation report followed on December 3, 1982 (Reference 11). This transmittal included an enclosure, TMI-1 Pressurizer Relief System-Piping Support Evaluation. Requests for additional information (References 12 and 17) were submitted to GPUN by the NRC on July 5, 1983 and February 15, 1984. GPUN responded on September 9, 1983 (Reference 13) and on March 20, 1984 (Reference 18).

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

4. REVIEW AND EVALUATION

4.1 Valves Tested

The TMI-1 Plant 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 inch Velan gate valve, Model F9-454B-13MS, with a Model SMB-00-10 Limitorque operator. The generic test program included tests on similar models except for the minor differences discussed below.

The Dresser Model 31739A safety valve used in the generic test program contains no material changes or dimensional changes to moving parts or porting/orificing from the safety valve installed in the plant. The tested valve is thus considered to be structurally and functionally identical to the in-plant valves.

The Dresser PORV installed at the TMI-1 Plant was originally a dash 1 (31533VX-30-1) design with a bore diameter of 1-3/32 inch. The test valve was a dash 2 design with a bore size of 1-5/16. The dash 2 design was precipitated by the 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 the operability. The TMI-1 plant valve has since been modified to incorporate the changes to the internals of the dash 2 design. The difference in bore diameter will only affect capacity and not operability. The valve was tested in the vertical position which corresponds to the plant configuration. The test valve is, therefore, considered an adequate representation of the in-plant valve.

The Velan block valve used at TMI-1 is a 2 1/2 inch gate valve Model Number F9-454B-13MS, and has a Limitorque operator SMB-00-10. The valve is installed in the vertical position. Two Velan valves, both 3 inch 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. Both valves were tested in the horizontal position. The plant and test valves are of the same style, internal design, and operation. They differ in size, pressure rating, and valve ends, which have no effect on the operability. The plant valve and test valves are designed for either vertical or horizontal orientation and the horizontal test are considered applicable for the vertical orientation. The larger 3 inch valve requires a higher 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 inch 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 TMI-1 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 original design of the B&W nuclear steam supply system provided for the safety valves to be mounted directly on the pressurizer nozzle. During the plant construction the decision was made to include a loop seal upstream of the TMI-1 valves to provide protection to the valve seats especially from H_2 and steam cutting. The EPRI test demonstrated that better valve performance and lower dynamic forces would be achieved without the loop seals and with short inlet piping. The problem of H_2 and steam cutting has been shown to be very minor at similar plants and the recent problem with corrosion observed on the TMI-PORV has been minimized by eliminating the residual sulfur in the reactor coolant system, deletion of the sodium thiosulfate tank, frequent chemical monitoring and valve inspection. As a result GPUN has elected to remove the loop seals and return to the original design; that is, mount the safety valves directly on the pressurizer nozzles. The Dresser safety valves used at TMI-1 are not designed to use different internals for steam or loop seal service and modifications to the valves as a result of the piping change were not necessary. Therefore the EPRI tests with the short inlet piping are applicable to the TMI-1 plant.

Reference 7 addresses the accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70 Rev. 2 except for the reactor coolant pump shaft seizure which is addressed in Reference 13. Bounding conditions for valve operation are identified (Reference 7) for the relief valve and for the safety valves. The bounding conditions were selected by considering the limiting cases from FSAR, extended high-pressure injection and cold overpressure events.

For the PORV, the FSAR events result in only steam discharge. Although some events result in peak pressure higher than opening set point, 2450 psig, the valve opens quickly so that the increase in pressure during the opening cycle is minimal. Testing with saturated steam at set pressure is, therefore, considered adequate. The Dresser PORV is a pilot operated valve and back pressure 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 is not reported in the submittal (Reference 11). The PORV discharge pipe routing is similar to the safety valves. The design flow, 100,000 lb/hr, is less than the design flow of a safety valve, 218,000 lbm/hr. The 4 inch discharge pipe is smaller than the 6 inch pipe for the safety valves. From these data the conclusion is reached that the net effect will result in a lower backpressure and, therefore, the expected backpressure for the PORV is less than the 500 psia reported for the safety valve. Testing of the valve (Reference 8) included numerous steam test with opening pressures close to the TMI-1 set pressure and back pressures as high as 760 psia which adequately bounds the expected conditions for the PORV.

For extended High Pressure Injection (HPI) events the initial opening of the PORV will be on steam but subcooled liquid could possibly follow. HPI events can, therefore, result in full pressure steam to water transition and water (400°F to 650°F) discharge (Reference 7). A full pressure steam to water transition test and full pressure liquid tests with temperatures ranging from 447°F to 647°F were included in the test series. The tests were run using the same discharg· ›ipe orifice which developed the high backpressures of 450 psia to 500 psia for the steam tests so that the expected backpressure was adequately represented. The HPI events are, therefore, considered to have been adequately represented by the tests.

The PORV is used for cold overpressure protection. For cold overpressurization events the valve is expected to operate over a range of inlet conditions. These include opening on 550 psig steam with a possible transition to saturated water and opening on subcooled water with temperatures ranging from 338 to 448°F (Reference 7). Opening on steam with possible transition to water is considered to be adequately represented by the full pressure, 2496 psia, steam to water transition test and the subcooled water conditions are considered to be adequately represented by the 689 psia 112°F water test (Reference 6).

For the safety valves, the FSAR events are bounded by maximum pressurizer pressure of 2662 psig and a pressurization rate of 175 psi/second. The possible fluid state on opening is steam only (Reference 7). Only testing with the short inlet piping is applicable to TMI-1 since GPUN has elected to remove the loop seals and place the safety valve on top of the pressurizer nozzles. Six steam test with the plant ring settings which had peak pressures exceeding 2662 psig were included in the test series (Reference 8). Tests with back pressures as high as 866 psi were included which bounds the maximum back pressure of 500 psia expected at TMI-1.

For HPI events the initial opening of the safety valve will be on steam. Subsequent opening could possibly be on subcooled liquid. HPI events can, therefore, result in full pressure steam to water transition and water (400°F to 640°F) discharge. Surge line insurge rates as high as 11,500 lbm/minute may occur (Reference 7). The test series included a full pressure steam to water transition test as well as three water tests with temperatures of 4?4°F to 608°F (Reference 8). During the transition and liquid testing, the same back pressure orifice was used that produced back pressure of 600 psia for the steam test so that the back pressures expected at TMI-1 were adequately represented. These conditions are considered to be sufficiently close to the conservatively selected bounding conditions to adequately demonstrate the performance of the valves.

Cold overpressurization events do not challenge the safety valves in the TMI-1 plant (Reference 6); therefore, no test conditions for these events need to be included in the program to verify the adequacy of the safety valves.

For the block valve only full pressure steam, 2500 psig, tests were performed (Reference 9). The block valve, however, is required per II.D.1 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 and, therefore, is nearly the same for water or steam and nearly independent of the flow (Reference 14). The full pressure steam tests, therefore, are adequate to demonstrate operability of the valve for low pressure steam and the required water conditions.

The test sequences and analyses described above, demonstrating that the test conditions bounded the conditions for the plant valves, verify that Items 2 and 4 of Section 1.2 have been met, in that conditions for the operational occurrences have been determined and the highest predicted pressures were chosen for the test. The part of Item 7, which requires showing that the test conditions are equivalent to conditions prescribed in the FSAR, is also met.

4.3 Operability

The PORV is required to operate over a range of conditions since it is expected to be challenged per II.D.1 for three different class of events; that is FSAR events, HPI events, and cold overpressure protection events.. The conditions for each class of events are discussed in Section 4.2. For FSAR events the valve opens on steam at 2450 psig. Numerous tests were conducted with these conditions (Reference 8). The valve opened and closed on demand for all of the steam test. For one series of test the bellows for the pilot valve stem developed several partially failed welds. The failure did not effect the valve performance and the manufacturer concluded that the failure did not have the potential to develop a condition that would effect performance. The bellows was replaced and performed without failure for the additional cycles. The measured flow of the test valve, when modified to account for difference in pressure and orifice area, demonstrated that the plant valve at set pressure would pass, within a few percent, its rated flow.

The PORV did fail to close and had a delayed closure for test conditions of low temperature water rapidly followed by 650°F water (Reference 8). These tests were intended to be representative for plants which have a cold loop seal before the relief valve and are not applicable to the TMI-1 plant (Reference 13).

For HPI events, the initial opening of the PORV will be on steam but subcooled liquid, 400°F to 650°F, could follow. A full pressure steam to water transition test and several full pressure liquid tests with water temperatures ranging from 447°F to 647°F were included in the test series. The valve opened and closed on demand without incident for these tests (Reference 8).

For cold overpressurization events the PORV is expected to operate over a range of inlet conditions. These conditions are considered to be adequately represented by the full pressure steam to water transition test and the 689 psia and 112°F subcooled water tests. The valve opened and closed without incident for both tests (Reference 8).

The TMI-1 PORV is a pilot operated valve that uses system pressure to hold the disc tight against the seat. Because of this design, the valve manufacturer, Dresser Industries, had cautioned that the PORV block valve should be closed when the reactor coolant system pressure was below 1000 psig to avoid damaging the PORV disk and seat by steam wirecutting. The TMI-1 submittals did not specify that such precautions were being employed. The NRC staff became concerned that the plant valve may become sufficiently damaged that its performance would no longer be represented by the test valve and, by Reference 17, requested clarification. GPUN indicated in their response, Reference 18, that a heavier spring would be installed to make the PORV more leak tight at low pressures. The heavier spring has been endorsed by the manufacturers and should prevent weeping and wirecutting. At full system pressure the spring force is small relative to the force from system pressure and, therefore, the TMI-1 valve is considered to be adequately represented by the test valve.

Bending moments expected to be induced across the PORV at TMI-1 were shown not to impair the valve operation. The maximum expected bending moment resulting from dead weight plus thermal expansion, plus operating basis earthquake plus PORV discharge was calculated as 8,200 in-lbs (Reference 11, Attachment 1). A test was conducted with a moment of 25,500 in.-lb imposed across the valve during opening and closing and the valve functioned satisfactorily (Reference 8).

Based on the test results described above and the proposed modification using a heavier spring, the demonstration of relief valve operability is considered acceptable.

The safety valves are also required to operate over a range of full pressure steam, steam to water transition, and subcooled water. As described in Section 4.2, tests were conducted with the short piping configuration and with the reference ring settings to be used at the TMI-1 plant over this full range as part of the EPRI PWR Test Program (Reference 8). For all steam tests the valve opened near the set point and had stable operation. The valve reached rated lift and passed rated flow at 3% accumulation for all tests except two. For one test the valve did not reach rated lift but did pass rated flow. For the other test, a very high back pressure test, the valve did not reach rated lift and flow at 3% accumulation but did reach rated lift and flow at 6% accumulation. For the water tests and the water flow of the transition test the valve partially opened, was stable and had adequate water flow. For one test with simulated high water makeup rate, the valve did not pass sufficient flow to terminate the pressure accumulation of the test facility. The valve did, however, pass much higher flow, 534,000 lb/hr, than the total maximum insurge rate, 264,000 lb/hr, of the TMI-1 pressurizer for the bounding extended high pressure injection event which is the limiting liquid flow case (Reference 13).

For all applicable safety valve tests, the observed blowdown exceeded the 5% design value. The maximum blowdown, 19.1%, was for the steam to water transition test. GPUN has submitted a B&W analysis (Reference 13, Attachment 2) that adequately demonstrates that blowdown as high as 20% can be tolerated without impeding natural circulation cooling because of hot leg voiding.

Bending moments expected to be induced across the TMI-1 safety valves were shown not to impair the valve operation. The maximum expected bending moment resulting from dead weight plus thermal expansion plus operating basis earthquake plus safety valve discharge was calculated as 20,500 in-lbs (Reference 12, Attachment 1). During the tests moments as high as 231,000 in.-lbs occurred during opening and closing of the valve and the valve operated satisfactorily (Reference 8).

For the test performance to be a valid demonstration of the plant safety valve stability, the test inlet piping must have a flow resistance at least as great as the plant. The plant valves are mounted directly on the pressurizer nozzles. The test facility inlet piping included a venturi and reducing flange and, therefore, had a higher flow resistance than the plant.

Based on the test results described above, the demonstration of safety valve operability is considered acceptable.

The block valve must be capable of closing over a range of steam and water conditions. As described in Section 4.2, high pressure steam test are adequate to bound operation over the full range and as described in Section 4.1, the tests with the 3 inch valve and SMB-000-10 operator conservatively demonstrate the operability of the plant valve. The test valve was cycled successfully at full steam pressure with full flow and was shown to open and close successfully with torque settings as low as 82 ft-lbs (Reference 9). The plant valve torque setting is 98 ft-lbs and, therefore, the tests are considered to adequately demonstrate acceptable valve operation.

The above test results, demonstrating that the valves operated satisfactorily, verify that the part of Item 1 of Section 1.2 which requires conducting tests to qualify the valves and that part of Item 7 which requires the effects of discharge piping on operability be considered have been met.

NUREG-0737, Item II.D.1 (item 5 of Section 1.2 above) states that the relief and safety valve qualification program should include qualification of the associated control circuitry. The licensee has made a determination that for TMI-1, the PORV and block valve control circuitry is not safety related and therefore does not have to be environmentally qualified in accordance with 10 CFR 50.49. The staff agrees with the licensee in that the PORV and block valve control circuitry need not be safety related for events which have a nexus to TMI-2. However, it is the staff position that the PORV and block valve circuitry performs a safety related function when these components are used to mitigate the consequences of the steam generator tube rupture accident to assure that offsite doses are maintained below the 10 CFR Part 100 limits. However, the circuitry of the PORV and block valve is not required to be environmentally qualified in accordance with 10 CFR 50.49 for a steam generator tube rupture accident if the circuitry will not be exposed to a harsh environment during this event.

4.4 Piping and Support Evaluation

4.4.1 Safety and PORV Inlet Configuration

The original design of the Babcock and Wilcox nuclear steam supply system for TMI-1 provided for safety valves to be mounted directly on the pressurizer nozzles. During the construction phase, a decision was made to install loop seals and a justification was provided at that time for the new design. GPUN, as a result of the EPRI tests, has removed the loop seals and installed the system as originally designed. For the current installation each of the safety valves and the relief valve have essentially independent lines. The two safety valves are mounted directly on separate pressurizer nozzles and have separate lines that join a common header to the reactor coolant drain tank about 1 1/2 ft above the tank. The block valve is mounted directly on another pressurizer nozzle and the PORV is mounted, in a vertical position, directly on the block valve. The PORV has a separate discharge line that connects to the common header to the reactor coolant drain tank just above the tank near the junction of the safety valve discharge lines. The piping and support evaluation of the discharge line (Reference 11) treats the current installation.

As discussed above, the original design for TMI-1 provided for mounting the safety valves and the PORV directly on the pressurizer. The analysis was completed at that time that demonstrated the structural adequacy of that configuration. Since the current configuration correpsonds with the original design, verification for this configuration was based on consideration that analysis for original system was previously reviewed and accepted by the NRC. The analysis considered the mechanical loads from earthquakes and valve discharge and the displacement loads from anchor movement resulting from the thermal expansion of the pressurizer. The loading combinations considered and the acceptance criteria for allowable stresses were those established for the NSSS (Reference 13). This method of verification is considered acceptable.

Reference 11, supplemented by Reference 13, addresses the adequacy of the discharge piping for the safety valves and the relief valve. The thermal-hydraulic analysis was performed using RELAP5. The RELAP5 control system was used to generate the forcing function concurrently with the RELAP5 thermal-hydraulic analysis execution. RELAP5 has been shown to be a suitable tool for the prediction of discharge loads (Reference 16). An additional verification for the use of RELAP5, including the procedure for generating the forcing function and the modeling technique used in the analyses of the TMI-1 piping, is provided in Reference 11.

In the TMI-1 analyses two RELAP5 models were used. One model included the piping for the two safety valves and the other the piping for the relief valve. Treating the two systems independently is considered acceptable since the safety and relief valves will not lift simultaneously and the common junction is a long distance from the valves.

Using the sequence of both safety valves opening simultaneously for the bounding case is considered acceptable, since there will be little interaction between the two discharge lines and since the controlling case for pipe loading has been determined not to be the initial wave spike but a later broader peak from the fluid momentum.

In the piping models the key parameters of node spacing, time step interval, choked flow locations and valve opening times were reviewed and were considered to be acceptable. The valve flow areas used in the models were conservatively chosen to produce flows corresponding to rated flows corrected for the 10% ASME derating and a 5% error. The method used to generate the wave force was to solve the acceleration term of the momentum balance equation. Since every RELAP5 computation time step was used in generating the force time histories, the method was considered acceptable.

The structural analysis was performed using the Gilbert/Commonwealth piping analysis computer code TPIPE. The use of this code is considered acceptable since it has been used and accepted for analysis of similar problems for several nuclear power plant projects. The three piping branches were assumed to be structurally independent, which is considered acceptable since the interaction of the three branch lines and the common header is isolated from the pressurizer connection by intermediate anchors on each branch. In addition, the common junction is located in a relatively stiff section of the pipe adjacent to the drain tank anchor and dynamic stresses in the region of the common junction are very low. The key parameters of lumped mass spacing and integration time step are acceptable. The use of zero damping is conservative and the determination of axial extension effects from peak pressures is adequate.

The loading combinations considered for the discharge portion of the piping and the acceptable service stress limits were those recommended by EPRI (Reference 19). The service stress limits used were those for ASME Section III Class 2 which are considered adequate to demonstrate that the discharge piping would not deform in a manner that would significantly restrict the discharge flow of the safety valves or the PORV. Acceptable operability of the safety valves and the PORV with the calculated moments was demonstrated by the EPRI tests (see Section 4.3). Service Stress Limits A were specified for sustained loads during the normal operation. Service Stress Limits B were specified for sustained loads during normal operation plus discharge loads from PORV opening. Service Stress Limits C were specified for sustained loads during normal operation combined with loads from the operation basis earthquake and the discharge loads from the PORV opening and were also specified for sustained loads during normal operation combined with discharge loads from the two safety valves opening. Service Stress Limits D were specified for two load combinations. The first was the limiting combination of sustained load plus design basis pipe break combined with the safe shutdown earthquake and maximum loads of either discharge of the safety valves or the PORV. The second was the load combination of sustained loads plus loads from loss of coolant accident combined with safe shutdown earthquake and the maximum loads of either discharge of the safety valves or the PORV.

The results of the analysis showed that addition of a snubber in the safety valve piping is necessary to maintain the stress levels within the acceptance criteria. Also, three supports in the safety valve lines are shown to require modification. The piping analysis is considered to adequately verify the acceptability of the piping system provided the identified modifications are made, since meeting the service stress limits of Section III Class 2 adequately demonstrates that the piping will not deform in a manner that would restrict the discharge flow from the valves.

The structural analysis indicated that the valve flange loads imposed by the discharge piping exceed the allowable loads listed in the vendor catalog for the safety valves and exceed those considered in the previous design for the PORV. These loads are considered acceptable since the valve vendor, Dresser Industries, has evaluated the loads and by letter dated November 11, 1982, has notified GPUN that the loads do not result in stresses above the vendor criteria for valve performance.

The analysis discussed above, demonstrating that a bounding case has been chosen for the piping evaluation, verifies that Item 3 of Section 1.2 has been met and the analysis of the piping and support system verifies that Item 8 of Section 1.2 has been met because the identified modifications have been made.

5. EVALUATION SUMMARY

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

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

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

Further, the prototypical tests and the successful performance of the valves and associated components demonstrated that this equipment has been constructed in accordance with high quality standards (General Design Criterion 30).

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