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TECHNICAL EVALUATION REPORT
TMI ACTION--NUREG-0737 (II.D.1)
RELIEF AND SAFETY VALVE TESTING
SEABROOK NUCLEAR STATION - UNITS 1 AND 2
DOCKET NOs. 50-443 AND 50-444

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ABSTRACT

In the past, safety and relief valves installed in the primary coolant system of light water reactors have performed improperly. 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 to: (a) reevaluate the functional performance capabilities of Pressurized Water Reactor (PWR) safety, relief, and block valves and (b) verify the integrity of the pressurizer safety and relief valve piping systems for normal, transient, and accident conditions. This report documents the review of those programs by the Nuclear Regulatory Commission (NRC) and their consultant, EG&G Idaho, Inc. Specifically, this report documents the review of the Seabrook Nuclear Station, Units 1 and 2, Licensee response to the requirements of NUREG-0578 and NUREG-0737. This review found the Licensee provided an acceptable response reconfirming that General Design Criteria 14, 15, and 30 of Appendix A to 10 CFR 50 have been met for the subject equipment. It should also be noted this review was made for both Units 1 and 2. However, the applicability of this review to Unit 2 is dependent on the verification that the Unit 2 as-built system conforms to the Unit 1 design reviewed in this report.

FIN No. D6005--Evaluation of CW Licensing Actions--NUREG-0737, II.D.1

Summary

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. This failure, plus other previous instances of improver valve performance, led the task force which prepared NUREG-0578 and NUREG-0737 to recommend that programs be developed to reexamine the functional performance capabilities of Pressurized Water Reactor (PWR) safety, relief, and block valves. The task force also recommended the programs verify the integrity of the pressurizer safety and relief valve piping systems for normal, transient, and accident conditions. This was deemed necessary to reconfirm that the General Design Criteria 14, 15, and 30 of 10 CFR 50, Appendix A, have indeed been satisfied for the subject equipment.

This report documents the review by EG&G Idaho, Inc., of the Seabrook Nuclear Station, Units 1 and 2, Licensee respons to the requirements of NUREG-0578 and NUREG-0737. The Licensee submittals were reviewed to determine the applicability of the test valves and test conditions to the plant valves and inlet conditions. The operability of the test valves was reviewed to determine the operability of the plant valves. The Licensee's analysis of the pressurizer discharge piping was reviewed to determine if acceptable stress limits were met for valve discharge transients.

The Licensee met the requirements of NUREG-0578 and NUREG-0737. The Licensee participated in the development and execution of an acceptable test program. The tests were successfully completed under operating conditions which bounded the most probable maximum forces expected from anticipated design basis events. The test results and piping analyses showed that the valves tested functioned correctly and safely for all steam and water discharge events specified in the test program that are applicable to Seabrook Nuclear Station, Units 1 and 2, and the pressure boundary component design criteria were not exceeded. Review of the Licensee's justifications indicated direct applicability of the test valve performance to the in-plant valves and systems intended to be covered by the test program. The plant specific piping was shown by analysis to be acceptable. Therefore, the Licensee reconfirmed that General Design Criteria 14, 15, and 30 of Appendix A to 10 CFR 50 have been met for the subject equipment.

This review was made for both Units 1 and 2 at Seabrook Nuclear Station. However, the applicability of this review to Unit 2 is dependent on the verification that the Unit 2 as-built system conforms to the Unit 1 design reviewed in this report.

PREFACE

This report was prepared for the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Reactor Regulation, by EG&G Idaho, Inc., NRC Regulatory Technical Assistance Unit.

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1. INTRODUCTION

1.1 Background

In the past, safety and relief valves installed in the primary coolant system of light water reactors have performed improperly. There were instances of valves opening below set pressure, valves opening above set pressure, and valves failing to open or reseat. From the 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 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 to: (a) reexamine the functional performance capabilities of Pressurized Water Reactor (PWR) safety, relief, and block valves and (b) verify the integrity of the pressurizer safety and relief valve piping systems for normal, transient, and accident conditions. These programs have been deemed necessary to reconfirm that General Design Criteria 14, 15, and 30 of 10 CFR 50, Appendix A, were indeed satisfied for the subject equipment.

1.2 General Design Criteria and NUREG Requirements

General Design Criteria 14, 15, and 30 require (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 operational occurances events; and (c) the components, which are part of the reactor coolant pressure boundary, be constructed to the highest quality standards practical.

To reconfirm the integrity of overpressure protection systems and thereby assure compliance to the General Design Criteria, 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 PWR 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 NRC staff review and evaluation, including criteria for success or failure of valves tested.

7. Submit a correlation, or other evidence, to substantiate the valves tested in a generic test program demonstrate the functionability of as-installed primary relief and safety valves. This correlation must show 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 also be considered.

8. Qualify the plant specific safety and relief valve piping and supports by comparing to [illegible] data and/or performing appropriate analyses.

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. Public Service Co. of New Hampshire (PSNH), the owner of the Seabrook Nuclear Station (SNS), Units 1 and 2, was one of the utilities sponsoring the EPRI Safety and Relief 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 those reports is discussed below.

Electric Power Research Institute developed a plan (Reference 4) for testing PWR safety and relief valves under conditions which bound actual plant operating conditions. Electric Power Research Institute, through the valve manufacturers, identified the valves used in the overpressure protection systems of the participating utilities and representative valves were selected for testing. The 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). Electric Power Research Institute, 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).

The utilities that participated in the EPRI Safety and Relief Valve Test Program also obtained information regarding the performance of PORV block valves (Reference 7). 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 8).

Electric Power Research Institute contracted with Westinghouse Corporation to produce a report on the inlet fluid conditions for pressurizer safety and relief valves in Westinghouse designed plants (Reference 9). Because SNS, Units 1&2, were designed by Westinghouse, this report is relevant to this evaluation.

Several test series were sponsored by EPRI. Power-operated relief valves 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 tested at Marshall only for full flow, full pressure steam conditions. Water flow tests were performed by WEMD on four valve models manufactured by them. 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 Combustion Engineering Company Kressinger Development Laboratory located in Windsor, Connecticut. The results of the relief and safety valve tests are reported in Reference 10. The results of the block valve tests are reported in References 7 and 8.

The primary objective of the EPRI 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 analyses) 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.

The EPRI test program was not designed to provide information on valve reliability. The EPRI program plan (Reference 4) states, "During the course of the specified tests, each valve will be subjected to a number of operational cycles. However, it should be noted that the test program, to be completed by July, 1981, is not intended to provide valve lifetime, cyclic fatigue or statistical reliability data."

NRC staff approval of the program is contained in Reference 11. Reference 11 states the staff has concluded the EPRI program produced sufficient generic safety valve and PORV performance information to enable utilities to comply with the plant specific information requirements in NUREG-0737, Item II.D.1. Transmittal of the test results meets Item 6 (provide test data to the NRC) of Section 1.2 in this report.

3. PLANT SPECIFIC SUBMITTAL

Public Service Co. of New Hampshire submitted their SNS, Units 1&2, evaluation report on the pressurizer safety valves, PORVs, PORV block valves, and piping on March 17, 1986 (Reference 12). Additional information on the piping analysis was submitted on June 1, 1986 in Reference 13. A request for additional information was transmitted to PSNH on March 31, 1987 (Reference 14), to which the Licensee responded on November 23, 1987 (Reference 15). Additional information was supplied by PSNH on May 8, 1989 and May 30, 1989 (References 16 and 17). The NRC requested the Licensee follow-up the May 30, 1989 conference call (Reference 17) with a letter documenting the information provided during the call.

4. REVIEW AND EVALUATION

4.1 Valves Tested

Seabrook Nuclear Station, Units 1 and 2, each utilize three safety valves and two PORVs in the overpressure protection system. In addition, each of them employ two PORV block valves. The safety valves are Crosby Model HB-BP-86 6M6 valves with steam internals. The PORVs are 3 in. by 6 in. Garrett straight through solenoid actuated valves. Only the PORVs have hot water seals upstream of the valves. The block valves are Westinghouse Model 3GM99 motor operated gate valves with Limitorque SB-00-15 motor operators.

The safety valve used at SNS, Units 1&2, the Crosby Model HB-BP-86 6M6 valve, was tested in the EPRI program. The safety valves at SNS, Units 1&2, are mounted on short vertical pipes to prevent the formation of water seals at the valve inlets. The valve internals are those designed for steam service. The valve was tested on a long inlet piping configuration with and without a loop seal, which bounds the SNS, Units 1&2, installation. The test valve had loop seal internals. Only the material used in the valve seats differs, and this does not affect valve operability within the limited number of cycles in the test program. In Reference 17, PSNH stated the ring settings for the Crosby 6M6 valves at SNS, Units 1&2, were factory set ring settings. The results from the EPRI tests with factory ring settings can, therefore, be used to demonstrate operability of the safety valve.

The Garrett PORVs used at SNS, Units 1&2, are of the same design as the valve tested by EPRI but have differences that do not affect valve operability. Those differences include inlet, outlet, seat, and cage flow hole areas. These differences affect flow capacity but not operabililty. Other differences that do not affect valve operability include internal versus external solenoid tubing and piloting of the cage directly on the valve body rather than indirectly to the body through the bonnet. Therefore, the test valve is considered to be representative of the plant valves.

The block valves used in SNS, Units 1&2, are the same design as the valve tested in the EPRI test program, a Westinghouse 3GM99 block valve. The valve was tested by EPRI in a horizontal configuration. The valve is designed for use in either a horizontal or vertical orientation. The plant valves have Limitorque SB-00-15 motor operators, which is the Limitorque operator used with the test valve. During EPRI testing, the 3GM99 block valve operator was rewired for limit closure on valve position rather than on torque and the yoke was redesigned. Public Service Co. of New Hampshire stated similar changes were made to the Seabrook valves. The test valve is, therefore, representative of the plant valves.

Based on the above, the test valves are considered to be applicable to the SNS, Units 1&2, valves and to have fulfilled the requirements of Items 1 and 7 of Section 1.2 in this report regarding applicability of the test valves.

4.2 Test Conditions

The valve inlet fluid conditions that bound the overpressure transients for Westinghouse-designed PWR plants are identified in Reference 9. The transients considered in this report include FSAR, extended high pressure injection, and low temperature overpressurization events. The plant specific conditions for these events discussed in this section are taken from Reference 9. The conditions applicable to SNS, Units 1&2, are those identified for a four-loop plant.

For FSAR transients resulting in steam discharge through the safety valves, the pressurizer experiences a peak pressure of 2555 psia (loss-of-load transient) and a maximum pressurization rate of 144 psi/s (locked rotor transient). The maximum expected backpressure is 560 psia.

In the EPRI testing program, the Crosby HB-BP-86 6M6 safety valve was subjected to two steam tests with a long inlet configuration. Of these tests, one test (1411) is applicable to the Crosby valves at SNS, Units 1&2, because the ring settings in this test (-77, -18) are representative of the plant ring settings and the test was performed with a drained loop seal. In this test the valve opening pressure was 2410 psia, the pop pressure was

2420 psia, and the peak tank pressure reached 2664 psia. The pressurization rate was 300 psi/s, the peak backpressure was 245 psia, and the blowdown was 8.2%. The test inlet fluid conditions for this steam test, except for the backpressure, are representative of the expected conditions for FSAR transients resulting in steam discharge for the safety valves. The Crosby 6M6 valve performance with high backpressure can be assessed using Test 929, a cold loop seal/steam test. The peak backpressure in this test, 710 psia, develops after the loop seal is discharged and full steam flow has been established. Other conditions for this test, peak tank pressure, 2726 psia, and pressurization rate, 319 psi/s, also bound the SNS, Units 1&2, inlet conditions.

For FSAR transients resulting in steam discharge, the PORVs will open at a pressure somewhat above the opening setpoint of 2350 psia. The maximum pressurizer pressure is 2532 psia (loss-of-load) and maximum pressurization rate is 130 psi/s (locked rotor) when the safety and relief valves actuate.

The Garrett test PORV was subjected to thirteen steam tests, one transition test, and two water seal simulation tests in the EPRI test program. In the steam tests, the maximum pressure at valve opening ranged from 2415 to 2760 psia. The valve opening pressure for the steam-water transition test was 2760 psia. The two water seal tests were conducted at initial pressures of 2755 and 2760 psia and inlet fluid temperatures of 130 and 293°F. The plant PORV water seal temperature is predicted to be about 250°F (Reference 15). The maximum back pressure for these tests ranged from 25 to 875 psia. The test fluid conditions in the steam and water seal tests on the PORVs are representative of FSAR transients.

The limiting FSAR transient, with respect to water flow through the safety valves and PORVs, is the feedwater line break (FWLB). The Westinghouse inlet conditions report (Reference 9) originally provided SNS, Units 1&2, inlet conditions for the FWLB transient. These conditions included maximum pressurizer pressure, 2504.9 psia, maximum liquid surge rate, 275.1 gpm, maximum pressurization rate, 3.0 psi/s, and liquid temperatures ranging from 568.7 to 584.1^{o}F. Subsequently, PSNH provided

revised FWLB liquid temperature conditions in Reference 16. This information indicated the liquid temperature would range from 603 to 605°F.

The Crosby 6M6 valve was subjected to one transition test (931a) with ring settings applicable to those at SNS, Units 1&2. This test included a loop seal upstream of the valve; however, with respect to valve operability, this test can be used to evaluate the plant valves without loop seals. The peak pressure and pressurization rate in the test were 2578 psia and 2.5 psi/s. The tank water temperature was 641°F. After the valve closed, the system was repressurized and the the valve cycled on 635°F water (Test 931b). In addition, one water test (932) was run with ring settings applicable to those in the plant valves. The peak pressure and pressurization rate was 2520 psia and 3.0 psi/s. The tank water temperature was 515°F. These conditions bound those at the plant.

The Garrett PORV was subjected to one transition test and three high pressure water tests. In the transition test, the peak pressure was 2760 psia and the water temperature was 653°F. In the water tests, the pressure ranged from 2640 to 2760 psia and water temperatures ranged from 249 to 648°F. The above conditions bound those expected for the plant PORVs.

The limiting extended High Pressure Injection (HPI) event is a spurious activation of the safety injection system at power. However, in this event, the PORVs and safety valves open on steam, and liquid discharge would not be observed until the pressurizer became water solid. According to Reference 9, this would not occur for at least 20 minutes into the event, which allows ample time for operator action. Thus, the potential for liquid discharge in extended HPI events can be disregarded.

Low temperature overpressurization (LTOP) events challenge only the PORVs since they are used to mitigate such transients. The fluid conditions for these events can vary between steam and subcooled water because of administrative requirements for maintaining a steam bubble in the pressurizer during low temperature operations. The plant specific range of potential fluid conditions for low temperature overpressure events was not

provided by PSNH. Low temperature overpressurization conditions in Reference 9 for similar four-loop Westinghouse plants were reviewed, and bounding conditions were selected to evaluate the performance of the SNS, Units 1&2, PORVs. These conditions include pressures from approximately 350 to 2350 psia and inlet fluid conditions varying from subcooled liquid to saturated steam.

In addition to the high pressure water, steam, and transition tests previously mentioned, the PORV was subjected to two low pressure water tests. The test pressures were 683 and 686 psia, while the valve inlet temperatures were 94 and 460°F. These test conditions, together with the test conditions in the high pressure tests, sufficiently encompass the range of expected fluid conditions for LTOP events at SNS, Units 1&2.

The block valves are required to operate over a range of fluid conditions (steam, steam-to-water, water) similar to those of the relief valves. However, the block valves were tested only under full pressure steam conditions (to 2485 psia). Based on testing preformed by Westinghouse (Reference 8), with similar internal materials under full pressure steam conditions, the required torque to open or close the valve: (a) depends almost entirely on the differential pressure across the valve disk, (b) is rather insensitive to momentum loading, (c) is nearly the same for water or steam, and (d) is nearly independent of the flow. Thus, full pressure steam tests are adequate to show valve operability for steam and water conditions.

Two transient conditions not part of the design basis are anticipated transients without scram (ATWS) and feed and bleed decay heat removal. The response of the overpressure protection system to 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.

The presentation above demonstrates that the test conditions bounded the conditions for the plant valves and verifies Items 2 and 4 of Section 1.2 in this report were met, in that conditions for the operational occurrences were determined and the highest predicted pressures were chosen for the tests. The presentation also verifies that the portion of Item 7,

which requires showing test conditions are equivalent to those prescribed in the FSAR, was met.

4.3 Operability

As discussed in the previous section, the safety valves and PORVs are required to operate over a range of full pressure steam, steam to water transition, and subcooled water fluid conditions. The valves were tested for the range of required conditions in the EPRI test program. The block valves are also required to operate for steam and liquid flow conditions. The valves were subjected to full pressure steam tests, the results of which apply also to liquid flow.

In one applicable steam test (1411), the safety valve opened at 2410 psia (-3.6% of the setpoint), was stable, and achieved 107% of rated steam flow at 3% accumulation and 92% of rated lift. The valve closed with 8.2% blowdown. In Test 929, the loop seal test used to bound the valve performance with high backpressures, the valve was stable on steam and achieved 110% of rated flow at 3% accumulation and 93% of rated lift. The valve closed with 5.1% blowdown. Thus, in the applicable tests, the valve performed its safety function of opening, relieving pressure, and closing.

A FWLB can result in high pressure and temperature liquid discharge through the safety valves. A loop seal/transition test (931a) and two water discharge tests (931b and 932) were used to bound the expected behavior of the plant valves. In Test 931a, the valve opened at 2570 psia (+2.8% of the set pressure), fluttered or chattered during loop seal discharge, stabilized during steam and water discharge, and closed. The valve blowdown was not available for this test. At 2415 psia with 641°F water, the valve passed 2355 gpm of liquid with the valve at 56% of rated lift. In Test 931b, the valve opened on 635°F water, chattered during opening, stabilized, and closed with 4.8% blowdown. The liquid flow rate in Test 931b was not recorded. In Test 932, the valve opened and immediately began to chatter. The valve chattered for 6.5 s before the test was terminated by manually opening the valve. This test used 515°F water. Because the pressurizer safety valves are designed for steam relief, valve chatter when passing highly subcooled water is not unexpected. The temperatures expected in a

FWLB at SNS, Units 1&2, (603 to 605°F) fall between the available test data at 640 and 515°F. However, based on engineering judgement, the SNS, Units 1&2, FWLB temperatures are close enough to the hot water EPRI tests to conclude the SNS, Units 1&2, safety valves will operate satisfactorily during a FWLB.

Bending moments as high as 298,750 in-lb (Test 908) were induced on the discharge flange of the Crosby 6M6 test valve, which had no adverse effect on valve performance. Since this applied moment exceeds the maximum estimated bending moment of 71,749 in-lb for the SNS, Units 1&2, valves (Reference 13), the performance of the plant valves is also expected to be unaffected by bending moments imposed during discharge transients.

As stated earlier, the observed blowdown in the applicable EPRI test was 8.2%, which exceeds the design value of 5%. Thus, it must be demonstrated that extended blowdown will not impact plant safety and valve operability. From a valve operability standpoint, filling the pressurizer with saturated water is not a concern. In the EPRI tests, the Crosby 6M6 safety valves at SNS, Units 1&2, were shown to be operable with steam, steam/water transition, and saturated water inlet conditions. Blowdown of 8.2% from a valve setpoint of 2500 psia should not present a challenge to plant protection equipment; therefore, this 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. To resolve this concern, three approaches were taken. First, if 8.2% blowdown occurs from a set pressure of 2500 psia, the primary pressure would decrease to 2295 psia. At 2295 psia, the saturation temperature is 655°F. The hot leg temperature would have to increase to this temperature before any hot leg voiding could occur. Therefore, significant voiding of the hot leg is not expected to occur due to the 8.2% versus 5% blowdown. Second, to consider the primary system response if voiding should occur, a NRC study of natural circulation (NC) test data was reviewed (Section 6.10.1 of Reference 18). The NRC study applies to PWRs with U-tube steam generators like SNS, Units 1&2. The study was based on NC data from experiments covering a wide range of possible accident or transient conditions that may occur in a PWR system; it also considered test facilities of widely different scale. NRC staff concluded the test data

showed the various modes of NC (single-phase, two-phase, and reflux) were able to keep the core cool as long as an adequate secondary heat sink is maintained and there is sufficient primary system mass inventory to keep the core covered with a two-phase mixture. Thus, 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 remove the decay heat. Finally, Westinghouse provided the results of an analysis that considered the effects of two or three stuck open safety valves (Reference 19). The analysis showed that even if this worst case condition for safety valve blowdown should occur (i.e., the valve sticks open) the emergency core cooling systems were able to keep the core covered and cool. Therefore, the extended blowdown observed in the EPRI tests does not impact plant safety or valve operability.

For the test to be an adequate demonstration of safety valve stability, the test inlet piping pressure drop should exceed the plant pressure drop. The test inlet pressure drop for the Crosby 6M6 valve on the loop seal configuration was 263 psid on opening and 181 psid on closing. The values calculated for the SNS, Units 1&2, safety valves were 122 and 80 psid for opening and closing, respectively (Reference 15). Therefore, the plant valves should be as stable as the test valves.

For all tests on the Garrett PORV, the valve opened and closed on demand. Total valve opening times were less than 1.24 s and closing times were less than 2.35 s. After testing was completed, the valve was inspected. Based on the limited number of cycles in the test program, there was no damage observed that would affect the future performance of the valve. Based on valve performance during testing, the PORVs were shown to operate under expected fluid transient conditions.

A bending moment of 33,200 in-lb was induced on the discharge flange of the Garrett test PORV, which has nearly the same valve body as the plant PORV. This moment had no adverse effect on valve performance. The maximum calculated bending moment for the SNS, Units 1&2, valves is 86,040 in-lb (Reference 13). However, the PWR Safety and Relief Valve Test Program Valve Selection Justification Report (Reference 5) stated that the Garrett straight through PORV is designed to operate with the maximum valve

deformation. A Garrett analysis of bending moments as large as 380,000 in-lb showed that valve operability would not be affected. Consequently, even though the EPRI tests only subjected the Garrett PORV to 33,200 in-lb, the valve is expected to operate with the higher induced moments expected under transient conditions.

The PORV block valve must be capable of closing over a range of steam and water conditions. As described in Section 4.2 of this report, high pressure steam tests are adequate to bound operation over the full range of inlet conditions. As described in Section 4.1 of this report, the tests conducted on the 3 in. Westinghouse Series 99 valve and SB-00-15 operator demonstrate the operability of the plant valve provided the plant block valve operator is adjusted to produce the maximum torque and wired for limit closure. The test valve was cycled successfully at full steam pressure with full flow. It was shown to open and close successfully with full operator torque (References 7 and 8). The plant block valves were modified to provide sufficient closing thrust as determined in the Westinghouse test program (Reference 15). Therefore, the tests are considered to have demonstrated acceptable valve operation.

NUREG-0737, Item II.D.1, states that the PORVs and their associated control circuitry shall be qualified for design basis accidents and transients. The EPRI test program included the PORV control circuitry attached directly to the valve in its test program (Reference 20), but did not include the circuits away from the valve (pressure sensing devices, cables, transmitters, etc.). The individual utilities still need to meet the NUREG-0737, Item II.D.1, requirements for the circuits away from the valve. Based on Reference 11, the NUREG requirement for environmental qualification of those circuits was to be met by including them in the program to meet the licensing requirements of 10 CFR 50.49. If the PORV control circuits are included in the 10 CFR 50.49 program, specific testing to meet the NUREG-0737 requirements is not necessary. The Licensee included the PORV controls in the SNS, Units 1&2, environmental qualification program (Reference 15). This meets the environmental qualification requirements for the control circuitry. With respect to the qualification of the control circuits during normal operation, testing of the PORV control circuits is required by the inservice testing program under 10 CFR 50.55a. Including

the circuits in this program meets the requirement to qualify the PORV control circuitry during normal operation.

The facts presented above demonstrate that Item 1 (conducting tests for valve qualification) and Item 7 (considering the affects of discharge piping on operability) of Section 1.2 in this report were met. Meeting the requirements of 10 CFR 50.49 and 50.55a are adequate to satisfy Item 5 of Section 1.2 in this report regarding the PORV control circuitry.

4.4 Piping and Support Evaluation

This evaluation covers the piping and supports upstream and downstream of the safety valves and PORVs extending from the pressurizer nozzle to the pressurizer relief tank. The piping was designed for deadweight, internal pressure, thermal expansion, earthquake, and safety and 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 analysis such that the calculated pipe discharge forces would bound the forces for any of the FSAR, HPI, and low temperature overpressurization events, including the single failure that would maximize the forces on the valve.

The safety valve and PORV discharge transients were analyzed in six separate cases. These cases included: (1) the relief and safety valves open sequentially at their respective set points, (2) the three safety valves discharge saturated steam and the relief valves remain closed, (3) the two relief valves discharge steam and experience a transition to saturated water plus a subsequent actuation during which 567°F water is discharged; the safety valves remain closed, (4) the three safety valves discharge 567°F water and the relief valves remain closed, (5) the two relief valves discharge 329°F water at 2400 psia and the safety valves remain closed, and (6) one relief valve discharges 329°F water at

2400 psia while the other relief valve and the safety valves remain closed. This approach is acceptable because it covers the type of valve actuations and conditions which are possible at SNS, Units 1&2.

Because water seals are maintained upstream of the PORVs, the water seal/steam discharge condition would generate the highest loads on the PORV piping system when the water seal is expelled and forced down the discharge piping. The steam discharge cases analyzed for the safety valves, combined with the water condition analyzed for the safety valves (which were representative of the coldest fluid temperatures expected at the valve inlet based on Reference 9), adequately represent the conditions expected for the safety valve piping system as discussed below. Therefore, the selection of these cases as the limiting conditions for the evaluation of the piping loads is considered adequate.

For these analyses, saturated steam at a maximum pressure of 2555 psia was assumed to be discharged through the safety valves. The conditions for the PORVs were saturated steam at a maximum pressure of 2532 psia. Hot water seals (250°F) were assumed upstream of the PORVs. For water discharge, the safety valve conditions were 567°F water at a maximum pressure of 2507 psia and the PORV conditions were 567°F water at 2403 psia and 329°F water at 2400 psia.

The thermal hydraulic analysis for SNS, Units 1&2, used 567°F water based on the FWLB water temperatures in Reference 9. As noted in Section 4.2 of this report, new FWLB water temperatures (603 to 605°F) were provided in Reference 16. During a conference call between the NRC staff, EG&G Idaho, Inc., and PSNH (Reference 17), PSNH stated the effects of the higher FWLB water temperatures on the piping thermal hydraulic analysis were assessed by a series of RELAP5/TULIP calculations (see below). The calculations looked at water temperatures of 605 and 650°F. This review showed the forces calculated using the 567°F water bound those expected from the 605 and 650°F water. Therefore, the forces for the original analysis bound the forces that would be generated based on the new FWLB temperature conditions and 650°F water, and a new analysis is not needed.

The thermal hydraulic analysis was performed using the RELAP5/MOD1 computer code. RELAP5 calculates the thermal hydraulic properties of the fluid as a function of time in each control volume and at each junction of the analysis model. The RELAP5 results are then input into the TULIP computer code to obtain the time histories of the fluid forces acting at the two ends of a pipe segment. RELAP5 is widely used in the industry and was shown to be an adequate tool for predicting piping discharge loads (Reference 21). The TULIP program generates force time histories from RELAP5 output. In Reference 15, the Licensee provided verification of TULIP's capability to generate force histories.

The key input parameters and assumptions made in the thermal hydraulic analysis, such as the valve flow area, the RELAP5 model node spacing, the valve opening time, time step size, etc., were reviewed and considered acceptable. The valve opening time for the safety valves was 0.01 s on steam and 0.02 s for water. These times are representative of those measured in the EPRI tests for these inlet conditions (valve opening time in the applicable steam test was 0.007 s and on water the valve opening time ranged from 0.012 to 0.021 s). The valve flow area used in the safety valve discharge analysis was calculated to produce the flow corresponding to 112% of the rated flow which is adequate for the Crosby valves used at SNS, Units 1&2. The PORVs were assumed to open in 0.01 s for steam and 0.625 s for water discharge. These opening times are faster than those measured in the EPRI tests and thus are conservative. The flow rate used in the analysis for the PORVs, 303,000 lbm/h at 2400 psia, is 144% of the valve rated flow at this pressure. This is considered to be conservative. The inlet pipes to the safety valves were modeled without loop seals while the water seals upstream of the PORVs were modeled. This represents the actual plant condition. The thermal hydraulic analysis is considered adequate for predicting the safety valve and PORV discharge loads.

4.4.2 Stress Analysis

The structural responses of the piping system due to safety valve/PORV discharge transients were calculated using the static method applying the maximum load for each leg enveloped from the six transient cases discussed in Section 4.4.1 of this report. The applicable load was applied to each

leg individually including the forces on adjacent legs using a dynamic load factor (DLF) of 1.5. The DLF of 1.5 was determined as follows. The fundamental frequency and period for each piping segment was determined. The duration time for the impulse force in each leg was determined and assumed to be a triangular pulse. Then the ratio of duration time and period was calculated for each segment and this ratio used to determine the DLF based on Reference 22. For the ratios found the DLFs ranged from 0.2 to 1.5. A DLF of 1.5 was used for conservatism. The peak forces for each segment were determined for the various transients analyzed. The peak forces were applied simultaneously to each segment even though the peak forces may not occur simultaneously. This approach is consistent with that taken in the SNS, Units 1&2, FSAR.

The static analysis was performed using the computer program ADLPIPE-D. The program was verified using NUREG/CR-1677 benchmark problems (References 23 and 24).

The piping upstream of the safety valves and PORVs was analyzed to the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division I, 1977 Edition with Addenda through December 31, 1977. The downstream piping was analyzed to the requirements of the ASME and ANSI B31.1 Power Piping Codes. The load combinations and stress limits for the upstream and downstream piping are equivalent to those recommended by EPRI (Reference 25). The piping stress summary presented by the Licensee compared the highest stresses in the piping against the applicable stress limits. All stresses were within the applicable stress limits.

The structural code governing the upstream support design is the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF, 1971 Edition with Addenda through Winter 1973. The load combination equations were consistent with the load combination equations in the EPRI Submittal Guide (Reference 25), and the resulting stresses were less than the code allowables. The downstream supports were discussed in Reference 17. The governing code for the downstream supports is the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF, Class 3, 1971 Edition with Addenda through Winter 1973. These downstream supports were analyzed according to the plant design requirements. This includes summing the maximum loads for

all the transients the supports undergo (normal, seismic, valve discharge, etc.) and comparing it to the allowable. The allowable load was taken to be approximately 80% of the ASME Code, Subsection NF, Class 3 allowable. The ASME allowable was increased by 1.33 to account for seismic effects if needed. Use of this approach is considered to be more conservative than using the load combinations and allowables recommended by EPRI. Public Service Co. of New Hampshire also stated that a load identified as a loss-of-coolant accident load in Reference 15 was actually the load due to valve discharge. Based on a review of the information provided by PSNH, all supports met code requirements.

4.4.3 Piping and Support Summary

The selection of a bounding case for the piping evaluation demonstrates the requirements of Item 3 of Section 1.2 in this report were met. The piping and support stress analysis verifies Item 8 was also met.

5. EVALUATION SUMMARY

The Licensee for SNS, Units 1&2, provided an acceptable response to the requirements of NUREG-0737, Item II.D.1. Therefore, the Licensee has reconfirmed that the General Design Criteria 14, 15, and 30 of Appendix A to 10 CFR 50 were met with regard to the safety valves, PORVs, and block valves. The rationale for this conclusion is given below.

The Licensee participated in the development and execution of an acceptable test program. The program was 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 operational occurances and design basis events. The generic test results and piping analyses showed that the valves tested functioned correctly and safely for all steam and water discharge events specified in the test program that are applicable to SNS, Units 1&2, and 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 performance to the in-plant valves and systems intended to be covered by the generic test program. The plant specific piping was shown by analysis to be acceptable.

Thus, the requirements of Item II.D.1 of NUREG-0737 were met (Items 1-8 of Section 1.2 in this report). Therefore, the Licensee demonstrated by testing and analysis for the subject equipment that: (a) the reactor primary coolant pressure boundary will have a low probability of abnormal leakage (General Design Criterion No. 14), (b) 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), and (c) the valves and associated components were constructed in accordance with high quality standards (General Design Criterion No. 30).

This review was made for both SNS, Units 1&2. However, the applicability of this review to Unit 2 is dependent on the verification that the Unit 2 as-built system conforms to the Unit 1 design reviewed in this report.

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