IDAHO NATIONAL ENGINEERING LABORATORY

INEL-94/0277

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February 1995

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Technical Evaluation Report TMI Action—NUREG-0737 (II.D.1) Relief and Safety Valve Testing Watts Bar Nuclear Plant, Units 1 and 2, Dockets 50-390 and 50-391

Idaho National Engineering Laboratory

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Lockheed
Idaho Technologies Company

Work performed under
DOE Contract
No. DE-AC07-94ID13223

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INEL-94/0277

TECHNICAL EVALUATION REPORT
TMI ACTION--NUREG-0737 (II.D.1)
RELIEF AND SAFETY VALVE TESTING
WATTS BAR NUCLEAR PLANT - UNITS 1 AND 2
DOCKET NOs. 50-390 AND 50-391

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February 1995

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Prepared for the
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
Under DOE Contract No. DE-AC07-94ID13223
FIN No. L1354, TAC No. M79992

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 development and completion of programs to do two things. First, they should reevaluate the functional performance capabilities of pressurized water reactor safety, relief, and block valves. Second, they should 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 Lockheed Idaho Technologies Company. Specifically, this report documents the review of the Watts Bar Nuclear Plant, Units 1 and 2, Applicant response to the requirements of NUREG-0578 and NUREG-0737. This review found the Applicant provided an acceptable response reconfirming they met General Design Criteria 14, 15, and 30 of Appendix A to 10 CFR 50 for the subject equipment. It should also be noted Lockheed Idaho performed this review for both Units 1 and 2. However, the applicability of this review to Unit 2 depends on verifying that the Unit 2 as-built system conforms to the Unit 1 design reviewed in this report.

L1354 - Technical Assistance for TVA Reviews

Summary

The failure of a power-operated relief valve (PORV) to reseat was a significant contributor to the Three Mile Island, Unit 2, sequence of events. This failure, plus other previous instances of improper valve performance, led the task force that prepared NUREG-0578 (TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations) and NUREG-0737 (Clarification of TMI Action Plan Requirements) to recommend development of programs to reexamine the functional performance capabilities of pressurized water reactor 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. The task force deemed this necessary to reconfirm that Licensees and Applicants indeed satisfied General Design Criteria 14, 15, and 30 of 10 CFR 50, Appendix A, for the subject equipment.

This report documents the review by Lockheed Idaho Technologies Company of the Watts Bar Nuclear Plant, Units 1 and 2, Applicant response to the above NUREG requirements. Lockheed Idaho reviewed: (a) the Applicant's submittals to determine the applicability of the test valves and test conditions to the plant valves and inlet conditions, (b) the operability of the test valves to determine the operability of the plant valves, and (c) the Applicant's analysis of the pressurizer discharge piping to determine if they met acceptable stress limits for valve discharge transients.

The Applicant met the requirements of NUREG-0578 and NUREG-0737. The Applicant participated in the development and execution of an acceptable test program. The Electric Power Research Institute successfully completed the tests under operating conditions that bounded the most probable maximum forces expected from anticipated design basis events. The valves tested functioned correctly and safely for all steam and water discharge events that are applicable to Watts Bar Nuclear Plant, Units 1 and 2. Also, the pressure boundary component design criteria were not exceeded. Review of the Applicant's justifications indicated direct applicability of the test valve performance to the in-plant valves and systems covered by the test program. The Applicant's analysis showed the acceptability of the plant specific piping. Therefore, the Applicant reconfirmed they met General Design Criteria 14, 15, and 30 of Appendix A to 10 CFR 50 for the subject equipment.

Lockheed Idaho completed this review for both Units 1 and 2 at the Watts Bar Nuclear Plant. However, the applicability of this review to Unit 2 depends on the Applicant verifying that the Unit 2 as-built system conforms to the Unit 1 design reviewed in this report.

PREFACE

Lockheed Idaho Technologies Company, National Nuclear Operations Analysis Department, at the Idaho National Engineering Laboratory prepared this report for the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation.

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TECHNICAL EVALUATION REPORT
TMI ACTION--NUREG-0737 (II.D.1)

RELIEF AND SAFETY VALVE TESTING
WATTS BAR NUCLEAR PLANT - UNITS 1 AND 2
DOCKET NOs. 50-390 AND 50-391

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 setpoint, valves opening above setpoint, and valves failing to open or reseat. From the past instances of improper valve performance, it is not known whether they occurred because of limited valve qualification or because of a basic unreliability in 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, Unit 2, sequence of events. These facts led the task force that prepared NUREG-0578 (Reference 1) and, subsequently, NUREG-0737 (Reference 2) to recommend development and execution of programs 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. The task force deemed these programs necessary to reconfirm that Licensees and Applicants indeed satisfied General Design Criteria 14, 15, and 30 of 10 CFR 50, Appendix A, 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 (RCS) 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 occurrences; and (c) the

components that 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 Nuclear Regulatory Commission (NRC), Division of Licensing, Office of Nuclear Reactor Regulation, issued the NUREG-0578 position as a requirement in a letter dated September 13, 1979, to all operating nuclear power plants. The NRC incorporated this requirement as Item II.D.1 of NUREG-0737, and they issued NUREG-0737 for implementation on October 31, 1980. As stated in the NUREG reports, each PWR Licensee or Applicant shall:

1. Conduct testing to qualify RCS 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 test 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. The test program covered pressurizer PORV block valves, PORVs, safety valves, and associated piping systems. The Tennessee Valley Authority (TVA), the owner of the Watts Bar Nuclear Plant (WBN), Units 1 and 2, was one of the utilities sponsoring the EPRI Safety and Relief Valve Test Program. The participating utilities transmitted the reports containing the results of the program to the NRC in Reference 3. Lockheed Idaho Technologies Company discusses the applicability of those reports below.

The Electric Power Research Institute developed a plan (Reference 4) for testing PWR safety and relief valves under conditions that bounded actual plant operating conditions. Through the valve manufacturers, EPRI identified the valves used in the overpressure protection systems of the participating utilities and selected representative valves for testing. The valves included enough of the variable characteristics so that their testing would adequately demonstrate the performance of the valves used by utilities (Reference 5). Through the nuclear steam supply system vendors, EPRI evaluated the FSARs of the participating utilities. They then developed a test matrix that bounded the plant transients requiring overpressure protection (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). The Electric Power Research Institute developed a list of block valves used or intended for use in participating PWR plants. They then selected for testing seven block valves to represent the block valves

used in PWR plants. Westinghouse Electro-Mechanical Division (WEMD) performed additional tests on valve models they manufacture (Reference 8).

The 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 Westinghouse designed WBN, Units 1 and 2, that report is relevant to this evaluation.

The Electric Power Research Institute sponsored several test series. They tested PORVs and PORV block valves at the Duke Power Company Marshall Steam Station located in Terrell, North Carolina. At the Marshall Station, EPRI conducted steam tests only. Therefore, EPRI tested block valves at Marshall only for full flow, full pressure steam conditions. However, WEMD performed water flow tests on four block valve models they manufacture. The Electric Power Research Institute conducted additional PORV tests at the Wyle Laboratories Test Facility located in Norco, California. They tested safety valves at the Combustion Engineering Company Kressinger Development Laboratory located in Windsor, Connecticut. In Reference 10, EPRI reported the relief and safety valve test results; References 7 and 8 contain the block valve test results.

The EPRI test program's primary objective was to test the various types of primary system safety valves used in PWRs for the full range of expected inlet conditions. Based on analyses, EPRI limited the conditions selected for testing 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 for verifying analytical piping models.

The Electric Power Research Institute did not design the test program 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."

Reference 11 contains the NRC staff approval of the EPRI test program. In Reference 11, the staff concluded the EPRI program produced enough 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

The TVA submitted their WBN, Units 1 and 2, evaluation report on the pressurizer safety valves, PORVs, PORV block valves, and a summary of the piping analysis on July 22, 1983 (Reference 12). The TVA submitted additional information on July 11, 1991 (Reference 13). The NRC transmitted requests for additional information to TVA on October 10, 1991, (Reference 14) and April 15, 1993, (Reference 15) to which the Applicant responded on December 26, 1992, July 19, 1994, and December 1, 1994, (References 16, 17, and 18).

4. REVIEW AND EVALUATION

4.1 Valves Tested

The overpressure protection systems at WBN, Units 1 and 2, each use three safety valves, two PORVs, and two PORV block valves. The safety valves are Crosby Model HB-BP-86 6M6 valves with steam internals. The PORVs are Target Rock Model No. 82UU-001 3 in. by 3 in. solenoid operated valves. Neither the safety valves or PORVs have water seals upstream of the valves. The block valves are Westinghouse Model 3GM88 motor operated gate valves with Limitorque SB-00-15 motor operators.

The EPRI program tested the safety valve used at WBN, Units 1 and 2, the Crosby Model HB-BP-86 6M6 valve. At WBN, Units 1 and 2, TVA mounted the safety valves on loop seal piping, but they modified the piping to be self-draining to prevent formation of a water seal at the valve inlets. The plant valves have steam internals. During testing, EPRI tested the valve on a long inlet piping configuration with and without a loop seal, and those configurations bound the WBN, Units 1 and 2, installation. The test valve had loop seal internals. Only the material used in the valve seats differs from

the steam internals, and this does not affect valve operability within the limited number of cycles in the test program. In Reference 16, TVA stated the ring settings for the Crosby 6M6 valves at WBN, Units 1 and 2, were factory set ring settings. Therefore, TVA can use the results from the EPRI tests with factory ring settings to demonstrate operability of the plant valves.

The Target Rock PORV used at WBN, Units 1 and 2, has the same general size, configuration, and principle of operation as the valve tested by EPRI. However, the plant and test valves have differences that both do and do not affect valve operability (References 16 and 17). The differences include inlet, outlet, bonnet, fixed core, valve body through bore, and magnetic sleeve changes. Target Rock changed the main disc, pilot disc, pilot seat insert, and sleeve materials and increased the main disc lift. They added expander rings under the piston rings in the plant valves to compensate for a loss of initial piston ring tension in high differential pressure service conditions. Target Rock also transferred guidance of the plunger and moveable core from the bonnet wall to the inner disc rod to compensate for scratches observed in the EPRI tests on the plunger, moveable core, and bonnet wall. Finally, they increased the nominal power input of the solenoid from 120 to 163 watts. Tennessee Valley Authority stated Target Rock based these changes on field experience and the EPRI tests. They stated these changes should improve the operability and performance characteristics of the plant valves and increase their flow capacity relative to the test valve. Therefore, Lockheed Idaho considers the test valve representative of the plant valves.

The block valves used in WBN, Units 1 and 2, are the same design as one of the valves tested in the EPRI test program, the Westinghouse 3GM88 block valve. The Electric Power Research Institute tested the valve in a horizontal configuration, and TVA installed the plant valves in the same position (Reference 16). The plant valves have Limitorque SB-00-15 motor operators; the test valve used the same Limitorque operator. During EPRI testing, the 3GM88 block valve fully closed only when EPRI set the operator to produce 182 ft-lb of torque. In Reference 13, TVA stated they changed the Unit 1 block valve operators to limit control closure rather than torque control closure to ensure complete valve closure. In Reference 17, TVA stated the 3GM88 operator supplied by Westinghouse is capable of providing 222 ft-lb of torque with 87% voltage based on the nameplate motor start torque and a

run/pullout efficiency of 60%. This exceeds the 182 ft-lb shown by EPRI to be adequate to open/close the 3GM88 block valve. It also exceeds the torque TVA calculated was required, 137 ft-lb, to operate the block valve under worst case conditions (Reference 17). For Unit 2, TVA committed to modifying the block valve operators before fuel load. Based on this information, the test valve represents the plant valves.

Based on the above, Lockheed Idaho considers the test valves applicable to the WBN, Units 1 and 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

As stated previously, Westinghouse designed WBN, Units 1 and 2. Reference 9 contains the valve inlet fluid conditions that bound the overpressure transients for Westinghouse-designed PWRs. In Reference 16, TVA stated they verified the inlet conditions in Reference 9 are still applicable to WBN, Units 1 and 2, except for the main feedwater line break (FWLB); that event was reanalyzed in 1991.

In Reference 17, TVA stated the new FWLB inlet conditions from the 1991 analysis are bounded by those in Reference 9, and that the differences between the two sets of inlet conditions do not affect the applicability of the EPRI tests to WBN, Units 1 and 2. Lockheed Idaho disagrees with TVA's statement in regards to the range of liquid temperatures at the valve inlet. The 1991 FWLB analysis for WBN, Units 1 and 2, resulted in temperature ranges of 603.7 to 624.6°F if the PORVs opened and 607.2 to 629.1°F if only the safety valves opened. The Reference 9 liquid temperature range was given as 630.8 to 637.0°F. Lockheed Idaho considers the 1991 results more limiting because of the colder liquid temperatures calculated in that analysis. Colder liquid temperatures were more challenging in the EPRI tests because valve operability problems generally occurred in the tests with colder water. Therefore, Lockheed Idaho will use the liquid temperature ranges from the 1991 FWLB reanalysis in the valve operability review.

The events considered in Reference 9 include FSAR, extended high pressure injection, and low temperature overpressurization events. The conditions applicable to WBN, Units 1 and 2, are those for four-loop plants. The following sections discuss the bounding inlet conditions for each of these events and the applicable EPRI tests.

4.2.1 FSAR Steam Discharge Transients

When the safety valves open alone, the limiting steam discharge conditions are peak pressurizer pressure, 2555 psia, and maximum pressurization rate, 144 psi/s. The peak pressure is from the loss-of-load event, and the peak pressurization rate is from the locked rotor event. The maximum expected backpressure is 550 psia (Reference 16).

In the test program, EPRI completed two steam tests on the Crosby 6M6 safety valve with a long inlet configuration and a drained loop seal. One of these tests (Test 1411) applies to the Crosby valves at WBN, Units 1 and 2, because the ring settings in this test (-77, -18) are representative of typical PWR plant ring settings. The ring settings represent the upper and lower ring positions measured from the level position referenced to the bottom of the disc ring. In Reference 16, TVA stated the ring settings used at WBN, Units 1 and 2, are -81 to -129 (upper ring) and -18 (lower ring), and these ring settings were determined by the valve manufacturer. Therefore, Lockheed Idaho considers the plant and test valve ring settings comparable.

In the applicable 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 and the peak backpressure was 245 psia. The test inlet fluid conditions for this steam test, except for the backpressure, are representative of the expected conditions for steam discharge.

To assess Crosby 6M6 valve performance with high backpressure, Lockheed Idaho will use Test 929, a cold loop seal/steam test. The peak backpressure in this test, 710 psia, develops after loop seal discharge, and full steam flow is established. Other conditions for this test, peak tank pressure, 2726 psia, and pressurization rate, 319 psi/s, also bound the plant 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). These results assume both the safety and relief valves open. The plant backpressure for the PORVs is 550 psia (Reference 16).

During testing, EPRI completed 15 steam tests on the Target Rock test PORV. In the steam tests, the maximum pressure at valve opening ranged from 2425 to 2521 psia. The maximum backpressure for these tests ranged from 155 to 520 psia. In these tests, the peak test pressure and backpressure are slightly lower than those expected at the plant. However, Lockheed Idaho considers the EPRI test conditions in the PORV steam tests adequate to represent FSAR steam transients. In addition, TVA stated (Reference 16) that they contracted with Target Rock to provide additional PORV tests with backpressure variations from 415 to 2350 psia.

4.2.2 FSAR Water Discharge 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) provided WBN, Unit 1, inlet conditions for the FWLB transient. These conditions also apply to WBN, Unit 2 (Reference 16). As discussed above, however, TVA reanalyzed the FWLB accident. In Reference 17, TVA stated the new FWLB inlet conditions are bounded by those from the Westinghouse valve inlet conditions report (Reference 9). From Reference 9 assuming the PORVs do not open, the safety valve inlet conditions include maximum pressurizer pressure, 2575 psia; maximum liquid surge rate, 430.4 gpm; and maximum pressurization rate, 1.6 psi/s. The liquid temperature range was taken from the 1991 analysis as provided by TVA in Reference 16; the liquid temperatures range from 607.2 to 629.1°F.

If the PORVs open, they will see conditions similar to the safety valves as discussed above. The PORV valve inlet conditions include maximum pressurizer pressure, 2575 psia; maximum liquid surge rate, 430.4 gpm; maximum

pressurization rate, 1.6 psi/s; and liquid temperatures ranging from 603.7 to 624.6°F. The liquid temperature range was provided by TVA in Reference 16.

With the Crosby 6M6 test valve, EPRI performed one transition test (931a) with ring settings applicable to those at the plant. This test included a loop seal upstream of the valve. However, with respect to valve operability, Lockheed Idaho can use this test 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 repressurized, and the valve cycled on 635°F water (Test 931b). In addition, one water test (932) was run with ring settings applicable to the plant valves. The peak pressure was 2520 psia, and the pressurization rate was 3.0 psi/s. The water temperature at the valve inlet at the start of the test was 463°F, and the tank water temperature was 515°F. These conditions bound those at the plant.

For the Target Rock PORV, EPRI completed one transition test and four high pressure water tests. In the transition test, the peak pressure was 2500 psia, and the water temperature was 656°F. In the water tests, the pressure ranged from 2490 to 2536 psia, and water temperatures ranged from 451 to 648°F. The peak pressures in the tests discussed above are lower than the 2575 psia expected in a FWLB. However, Reference 6 stated the inlet pressure will affect PORV performance only during valve opening and closing. Reference 6 also noted the Target Rock PORV opens quickly (less than 0.7 s). For the FWLB at WBN, Units 1 and 2, the calculated pressurization rate is 1.6 psi/s. At this pressurization rate, the plant PORVs will be fully open before the inlet pressure exceeds the test pressure. Therefore, Lockheed Idaho considers testing of the PORV at 2490 to 2536 psia adequate. Lockheed Idaho also considers all the above conditions representative of those expected for the plant PORVs.

4.2.3 Extended High Pressure Injection Events

The limiting extended High Pressure Injection (HPI) event is a spurious activation of the safety injection system at power. For four-loop plants, this event challenges both the safety valves and PORVs. The PORVs and safety valves open on steam, but liquid discharge would not occur until the

pressurizer became water solid. According to References 9 and 12, this would not occur for at least 20 minutes into the event, and this allows ample time for operator action. Thus, Lockheed Idaho disregarded the potential for liquid discharge in extended HPI events.

4.2.4 Low Temperature Overpressurization Events

At WBN, Units 1 and 2, TVA uses the PORVs for protection from low temperature overpressurization (LTOP) events. The fluid conditions for these events can vary between steam and subcooled water because of administrative requirements for maintaining a pressurizer steam bubble during low temperature operations. In Reference 16, TVA provided the plant specific range of potential fluid conditions for LTOP events. The current LTOP control system varies the PORV setpoint from 485 to 2365 psia as the RCS temperature ranges from 70 to 450°F. In Reference 16, TVA stated a new LTOP control system is being implemented that will slightly increase the PORV setpoints so that they vary from 500 to 2365 psia as the RCS temperature ranges from 70 to 450°F.

In addition to the various high pressure tests previously mentioned, EPRI performed two low pressure water tests on the Target Rock PORV. The test pressures were 690 and 715 psia, while the valve inlet temperatures were 114 and 447°F. These test conditions, together with the high pressure test conditions, adequately represent the expected LTOP inlet conditions at WBN, Units 1 and 2.

4.2.5 Block Valve Inlet Conditions

The block valves operate over a range of fluid conditions (steam, steam-to-water, and water) similar to those of the relief valves. However, EPRI tested the block valves only under full pressure steam conditions (to 2420 psia). For Westinghouse manufactured valves, WEMD performed additional water flow tests. The WEMD test conditions ranged from 60 to 600 gpm water flow and 1500 to 2600 psi differential pressure. Based on Reference 8, Westinghouse found four things concerning block valves with similar internal materials. Under full pressure steam conditions, Westinghouse found the required torque to open or close the valve:

1. Depends almost entirely on the differential pressure across the valve disc.
2. Is rather insensitive to momentum loading.
3. Is nearly the same for water or steam.
4. Is nearly independent of the flow.

Thus, full pressure steam tests are adequate to show valve operability for steam and water conditions.

4.2.6 Other Transients

Two transient conditions not part of the design basis are feed and bleed decay heat removal and anticipated transients without scram. This review did not consider the response of the overpressure protection system to these two transient conditions. Neither the Applicant nor the NRC have evaluated the performance of the system for these events.

4.2.7 Inlet Conditions Summary

The presentation above demonstrates that the test conditions bounded the conditions for the plant valves, and it verifies TVA met Items 2 and 4 of Section 1.2 in this report. That is, TVA determined the conditions for the operational occurrences and chose the highest predicted pressures for the tests. They also met the portion of Item 7 that requires showing test conditions are equivalent to those prescribed in the FSAR.

4.3 Operability

The safety valves and PORVs operate over a range of full pressure steam, steam-to-water transition, and subcooled water fluid conditions, and EPRI tested the valves for the required range of conditions. The block valves also operate over a range of steam and liquid flow conditions. The Electric Power Research Institute tested the block valves with full pressure steam; those test results also apply to liquid flow.

4.3.1 Safety Valves

In the one applicable steam test (1411), the safety valve opened at 2410 psia (-3.6% of the setpoint) and operated stably. The valve achieved 107% of rated steam flow at 3% accumulation and 92% of rated lift, and it closed with 8.2% blowdown. Test 929 was the loop seal test used to bound valve performance with high backpressures. In the test, the valve was stable on steam and achieved 111% 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 test (931b and 932) bound the expected behavior of the plant valves. In Test 931a, the valve opened at 2570 psia (+2.8% of the setpoint), fluttered or chattered during loop seal discharge, stabilized during steam and water discharge, and closed with 12.7% blowdown. 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 within -1% of the setpoint, chattered during opening, stabilized, and closed with 4.8% blowdown. The operators did not record the liquid flow rate in Test 931b. In Test 932, the valve opened and immediately began to chatter. The operators manually terminated the test by opening the valve to stop the chatter. 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 WBN, Units 1 and 2, (607 to 629°F) fall between the available test data at 635 and 463/515°F. However, based on engineering judgement, the WBN, Unit 1 and 2, FWLB temperatures are close enough to the hot water EPRI tests to conclude the plant safety valves will operate satisfactorily during a FWLB.

The largest bending moment EPRI induced on the discharge flange of the Crosby 6M6 test valve was 298,750 in-lb (Test 908). Application of this bending moment did not affect valve performance. This applied moment exceeds the maximum estimated bending moment of 90,984 in-lb for WBN, Unit 1, valves. The plant value is based on the algebraic sum of the maximum, faulted, local

M_y and M_z moments provided by TVA in Reference 16. In Reference 16, TVA committed to calculating a similar bending moment for Unit 2 during the reanalysis of the Unit 2 pipe support loads prior to Unit 2 fuel load. Therefore, the bending moments imposed during discharge transients will not affect Unit 1 valve performance.

As stated earlier, the maximum observed blowdown in the applicable EPRI tests was 12.7%, and this exceeds the design value of 5%. Thus, TVA must demonstrate that extended blowdown will not impact plant safety and valve operability. They provided this information in Reference 16. Tennessee Valley Authority stated Westinghouse evaluated the effect of 13% safety valve blowdown on various accident analyses. The Westinghouse evaluation found:

1. Extended safety valve blowdown of up to 13% will not cause the pressurizer to fill in any licensing basis event where the pressurizer does not already become water solid.

2. Extended safety valve blowdown of up to 13% will not challenge any safety systems that were not previously challenged in the licensing basis safety analyses.

3. Extended safety valve blowdown of up to 13% will not cause voiding of the primary system in any licensing basis event.

Therefore, the extended blowdown observed in the EPRI tests does not impact plant safety or valve operability.

For the steam test to adequately demonstrate safety valve stability, the test inlet piping pressure differences should exceed the plant pressure differences. In Reference 16, TVA provided the WBN, Units 1 and 2, calculated values. The pressure differences calculated for the plant safety valves were 256.51 and 152.73 psid for opening and closing, respectively. The corresponding differences for the test valve on the loop seal configuration were 263 psid on opening and 181 psid on closing. Therefore, the plant valves should be as stable as the test valves.

4.3.2 Power-Operated Relief Valves

For all applicable tests on the Target Rock PORV (non-loop seal tests), the valve opened and closed on demand. Total valve opening times were less than 0.66 s and closing times were less than 0.69 s. The Electric Power Research Institute inspected the valve after the completion of testing. Based on the limited number of cycles in the test program, EPRI observed no damage that would affect future valve performance.

Reference 6 indicated the Target Rock PORV is susceptible to backpressure effects because it is a pilot valve design. In Reference 16, TVA argued the pilot valve design of the Target Rock PORV is different from other pilot valve operated PORVs because the pilot valve is internal to the valve. The main and pilot discs are mechanically linked together so the solenoid force applied to the pilot disc assists in opening the main disc. At zero differential pressure across the PORV, the solenoid force alone is sufficient to lift the main disc. Further, TVA stated Target Rock personnel indicated increased backpressure makes it easier to open or close the valve, but does not affect the ability of the valve to remain open. Therefore, they concluded the backpressure would only affect flow through the valve and the valve opening and closing times. Based on the information from TVA, Lockheed Idaho agrees with this conclusion.

In Reference 16, TVA also discussed specific test data to show the Target Rock PORV is not affected by backpressure. The EPRI test valve was subjected to a 520 psia backpressure in one test, and this backpressure is close to the 550 psia backpressure expected at the plant. The valve fully opened and closed on demand in this test. In addition, TVA stated they contracted with Target Rock to conduct additional tests on one of the PORVs from WBN, Units 1 and 2. Backpressures in these tests ranged from 415 to 2350 psia, and the PORV operated normally.

During EPRI testing, the operators induced a bending moment of 32,900 in-lb on the Target Rock test PORV. In Reference 17, TVA discussed the results of an evaluation performed by Target Rock on the differences between the EPRI test valve and the Target Rock PORVs installed at the plant. The evaluation concluded that, because of the differences in valve design, the

potential for valve binding was greater in the EPRI test valve relative to the valves installed at WBN, Units 1 and 2. Application of the test bending moment did not affect test valve performance. The maximum calculated bending moment for the WBN, Unit 1, valves is 28,596 in-lb, and this is less than the bending moment applied to the test valve. The plant value is based on the algebraic sum of the maximum, faulted, local M_y and M_z moments provided by TVA in Reference 16. In Reference 16, TVA committed to calculating a similar bending moment for Unit 2 during the reanalysis of the Unit 2 pipe support loads prior to Unit 2 fuel load. Based on the above, Lockheed Idaho expects the Unit 1 Target Rock PORVs to operate with the maximum expected bending moment at the plant.

Based on valve performance during testing and other information provided, the Applicant verified the PORVs operated properly under expected fluid transient conditions.

4.3.3 PORV Control Circuit Qualification

NUREG-0737, Item II.D.1, requires the qualification of the PORVs and their associated control circuitry for design basis accidents and transients. The EPRI test program included the PORV control circuitry attached directly to the valve (Reference 19). It did not include the circuits away from the valve such as 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 NRC concluded Applicants could meet the NUREG environmental qualification requirement for those circuits by including them in their 10 CFR 50.49 program. However, TVA stated (Reference 16) the PORV control circuitry at WBN, Units 1 and 2, contains non-environmentally qualified inputs.

In question 12, Reference 14, the NRC gave TVA several alternatives if the PORV control circuitry is not included in the 10 CFR 50.49 program. In Reference 16, TVA discussed how they met one of those alternatives. The NRC alternative stated:

> The PORVs are not required to perform a safety function to mitigate the effects of any design basis event in a harsh environment and failure in a harsh environment will not adversely

impact safety functions or mislead the operators (PORVs will not experience any spurious actuations and, if emergency operating procedures do not specifically prohibit use of PORVs in accident mitigation, it must be ascertained the operators can close the PORVs under harsh environment conditions).

In Reference 16, TVA stated they do not require the PORVs to perform a safety function to mitigate the effects of a design basis event in a harsh environment and failure in a harsh environment will not adversely affect safety functions or mislead the operators. They noted no credit is taken for PORV operation to mitigate the effects of an accident, except for high-point venting of the RCS. This venting is accomplished by remote-manual control of the environmentally qualified PORVs using only environmentally qualified portions of the control circuitry.

Lockheed Idaho notes the NRC alternative requires that the PORVs not experience any spurious actuations. In Reference 16, TVA did not state the PORVs will not experience any spurious actuations. However, TVA did say that, if the PORV opens spuriously due to an environmentally induced failure of one of the non-qualified inputs to the PORV control circuitry, the operators can still close the PORV by remote-manual control using the qualified control circuits discussed above. If a postulated single-failure is assumed to prevent remote-manual closure, TVA stated the operators can use the environmentally qualified block valve and environmentally qualified block valve control circuitry to isolate the PORV. The plant post-accident monitoring system provided positive indication of both PORV and block valve position. Although this does not meet the NRC requirement regarding spurious PORV activation directly, Lockheed Idaho considers that TVA meets the intent of the NRC requirement by having a single-failure proof means of closing and/or isolating the PORVs in a harsh environment should a spurious activation occur.

Therefore, Lockheed Idaho concludes TVA meets the environmental qualification requirements for the control circuitry.

With respect to the qualification of the control circuits during normal operation, TVA stated they submitted Unit 1 technical specifications to the NRC that include surveillance requirements to ensure the operability of the PORVs, block valves, and their associated control circuits. In Reference 16,

TVA also stated they had committed to follow the recommendations in Generic Letter (GL) 89-10 for safety-related motor-operated valve testing and surveillance and to implement the improvements identified in GL 90-06 for PORV and block valve reliability (References 20 and 21). This meets the requirements to qualify the control circuits for normal operation.

4.3.4 PORV Block Valves

The PORV block valve must close over a range of steam and water conditions. As described in Section 4.2 of this report, high pressure steam tests adequately bound operation over the full range of inlet conditions. As described in Section 4.1 of this report, the tests conducted with the 3GM88 valve and SB-00-15 operator demonstrate plant valve operability. This is because TVA modified the Unit 1 block valve operator to close on limit, and, in this mode of operation, the torque provided by the operator is greater than that used in the EPRI tests. In Reference 17, TVA noted the Unit 2 block valve operator will be modified in the same way as the Unit 1 operator prior to fuel load. The valve tested opened and closed successfully with the test valve operator set to produce 182 ft-lb of torque (Reference 7). Therefore, the tests demonstrated acceptable valve operation. In addition, TVA stated (Reference 17) the block valve operating requirements and capabilities are validated by dynamic testing that is part of the WBN, Units 1 and 2, GL 89-10 test program. Including the PORV block valves in the GL 89-10 test program provides additional assurance the block valves will operate acceptably.

4.3.5 Operability Summary

The facts presented above demonstrate TVA met Item 1 (conducting tests to qualify the valve) and met Item 7 (considering the affects of discharge piping on operability) of Section 1.2 in this report. Meeting the NRC alternative to qualifying the control circuits under 10 CFR 50.49 and committing to meet the requirements of GL 90-06 adequately satisfies 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 extending from the pressurizer nozzle to the pressurizer relief tank. The Applicant designed the piping for deadweight, internal pressure, thermal expansion, earthquake, and safety and relief valve discharge conditions. This section discusses the calculation of the hydraulic force time histories due to valve discharge, the structural analysis methods, and the load combinations and stress evaluation. This evaluation is for WBN, Unit 1. In References 13 and 16, TVA stated they plan to redo the thermal hydraulic and structural analyses for the piping and supports for Unit 2 as part of a hanger and analysis update program. This will be completed prior to fuel load for Unit 2.

4.4.1 Thermal Hydraulic Analysis

The TVA used pressurizer fluid conditions 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.

They analyzed the safety valve and PORV discharge transients in six separate cases (Reference 16). These cases included: (a) the three safety valves open/close and the relief valves remain closed, (b) the two relief valves open/close and the safety valves remain closed, and (c) the two relief valves open/close during the LTOP mode of operation. Lockheed Idaho considers this approach acceptable because the safety valves and PORVs have different setpoints. Therefore, they will not lift simultaneously.

A valve operating condition that is more likely to occur would be a PORV discharge followed by a safety valve discharge. Because the PORVs have a lower setpoint, they would open first. In this case the PORV piping loads would be the same as those calculated from case b above. This scenario, however, reduces the safety valve piping loads due to the backpressure buildup in the discharge piping resulting from the PORV discharge; therefore, TVA need not analyze this condition.

Because there are no water seals upstream of the safety valves, the steam discharge condition would generate the highest loads on the safety valve piping system. The safety valve steam discharge cases analyzed adequately represent the conditions expected for the safety valve piping system as discussed below. Similarly, the PORV discharge cases adequately represent the conditions expected for the PORV piping system. Also, TVA stated in Reference 18 that valve opening on water is not calculated for FSAR transients. Lockheed Idaho notes that valve inlet conditions for NUREG-0737, Item II.D.1, were to be based on FSAR transients. Therefore, Lockheed Idaho considers the selection of these cases adequate to represent the limiting conditions for the piping load evaluation.

For the safety valve opening case, TVA assumed the safety valves opened at 2575 psia, passed saturated steam at 673°F, and had a pressurization rate of 54 psi/s. The maximum pressure was 2748 psia. The safety closing case assumed the safety valves closed at 2374.7 psia on saturated steam at 673°F. When the PORVs passed steam, TVA assumed they opened at 2420 psia, passed steam at 663°F, and had a pressurization rate of 54 psi/s. The maximum pressure was 2525 psia. The PORV closing case assumed the PORVs closed at 2400 psia on saturated steam at 663°F. For PORV water discharge, TVA assumed LTOP type conditions. For PORV closing, TVA assumed a pressure of 850 psi and a water temperature of 380°F. For PORV opening, the assumed conditions were pressure of 605 psia and 70°F water.

The pressurization rate used in the thermal-hydraulic analyses, 54 psi/s, is less than the 144 psi/s discussed in the Westinghouse valve inlet conditions report. In Reference 17, TVA responded to a question on the adequacy of the pressurization rate used for WBN, Units 1 and 2. TVA noted the peak loads were calculated to occur within 0.28 s of the valve opening. Therefore, use of a 144 psi/s pressurization rate would result in a maximum pressure increase of 25 psi. This is 1% of the pressurizer safety valve opening pressure of 2500 psia. For the other cases analyzed, the percent increase in pressure was less than 1% of the valve opening or closing pressure for the particular case analyzed because of shorter times to the peak calculated loads. Therefore, Lockheed Idaho considers use of the 54 psi/s pressurization rate adequate for the thermal-hydraulic analysis.

The TVA performed the thermal hydraulic analysis using the WATHAM and STEHAM computer programs. In Reference 16, TVA stated Stone & Webster Engineering Corporation (SWEC) used these programs to analyze steam discharge (STEHAM) and water discharge (WATHAM). Lockheed Idaho reviewed STEHAM in other utilities' submittals (Reference 22) and determined it was adequately qualified by SWEC for valve discharge thermal hydraulic analyses. In Reference 17, TVA provided information on the qualification of WATHAM that showed the program was also adequate to calculate valve thermal hydraulic analyses.

Lockheed Idaho reviewed the key input parameters and assumptions made in the thermal hydraulic analysis, such as the valve opening time, time step size, valve flow rates, etc. The valve opening time for the safety valves was 0.008 s on steam. This time adequately represents that measured in the EPRI tests for steam inlet conditions (valve opening time in the applicable steam test was 0.026 s). The valve flow area used in the safety valve discharge analysis produced a flow corresponding to greater than 111% of the rated flow. This is adequate for the Crosby valves used at the plant which passed 111% or less of rated flow in the EPRI tests. The TVA assumed the PORVs opened in 0.006 s for steam and water discharge. The measured opening times for the Target Rock PORV in the EPRI tests were 0.2 s on water and 0.66 s on steam. The use of faster times is conservative due to the larger acoustic wave generated by the faster valve opening time. The flow rate used in the analysis for the PORVs was 111% of the valve rated flow (233,333 lbm/hr). In Reference 18, TVA noted that adjusting the EPRI measured flow for the EPRI test valve to the larger orifice used in the WBN, Unit 1 and 2, PORVs results in a flow range of 226,930 to 240,720 lbm/hr. Therefore, Lockheed Idaho considers the analysis value representative of that expected at the plant. The time step control resulted in time steps between 0.0006 to 0.001 s, and this is adequate based on the code verification problems. Therefore, Lockheed Idaho considers the thermal hydraulic analysis adequate for predicting the safety valve and PORV discharge loads.

4.4.2 Stress Analysis

The TVA calculated the dynamic structural responses of the piping system to safety valve/PORV discharge transients using the TPIPE computer program.

Based on Reference 16, TVA described TPIPE and its qualification against other computer programs used throughout the industry in Section 3.9.1.2.1 of the FSAR. Lockheed Idaho concluded NRC acceptance of TPIPE in the FSAR indicates the adequacy of the program.

The TVA calculated the piping system response using the direct integration method. In Reference 16, TVA supplied information on the important structural analysis parameters of time step, cutoff frequency, damping, and mass point spacing. The time step chosen for the structural analysis was 0.00025 s. This time step is small enough to accurately represent the wave functions for the cutoff frequency selected, 500 Hz. The TVA modeled damping at or below the values given in Regulatory Guide 1.61 for the frequencies to be analyzed. Mass point spacing for the major pipe sizes was less than 3.5, 5.0, and 8.0 ft for 3, 6, and 8 inch piping. Lockheed Idaho considers these structural analysis parameters adequate.

The TVA took the load combinations for the piping and supports from TVA Design Criteria WB-DC-40-31.7. In Reference 16, TVA stated they based the load combinations on FSAR commitments, and this is adequate. For the stress limits, TVA based the upstream piping stress limits on ASME Class 1 requirements and the downstream piping stress limits on ASME Class 2 requirements (Reference 17). The ASME code version was the 1971 Edition with Addenda through Summer 1973. They took the allowable stress limits for the upstream and downstream piping supports from TVA Design Criteria WB-DC-40-31.9, and this is consistent with the plant FSAR (Reference 17). The requirements of the AISC code, 7th Edition, with Supplements 1, 2, and 3, and/or later editions and manufacturer allowable loads were used to develop TVA Design Criteria WB-DC-40-31.9.

The piping stress summaries provided by TVA (Reference 16) compare the highest stresses in the piping with the applicable stress limits in the form of ratios (calculated over allowable). Lockheed Idaho reviewed the piping stress results and found all stresses within the applicable stress limits. Lockheed Idaho reviewed a similar comparison in Reference 16 for the pipe supports, and all supports met the applicable requirements. In Reference 16, TVA stated they calculated the stresses and loads on the pressurizer safety valve and PORV nozzles. Westinghouse and TVA personnel reviewed these

stresses and loads and found them acceptable. Finally, TVA indicated they had to modify the piping and support configuration during the course of the analysis to meet the piping design criteria. They committed to complete the Unit 1 modifications needed to bring the actual plant configuration into agreement with the final system analyzed and the stress analysis results discussed above prior to Unit 1 fuel load (Reference 16).

4.4.3 Piping and Support Summary

The applicant met Item 3 of Section 1.2 in this report by selecting bounding cases for the piping evaluation. Based on the piping and support stress analysis TVA provided, they also met Item 8.

5. EVALUATION SUMMARY

The Applicant for WBN, Units 1 and 2, provided an acceptable response to the requirements of NUREG-0737, Item II.D.1. Therefore, the Applicant reconfirmed 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 discussion below provides the rationale for this conclusion.

The Applicant participated in the development and execution of an acceptable test program. The program would qualify the operability of prototypical valves and demonstrate their operation would not invalidate the integrity of the associated equipment and piping. The Electric Power Research Institute successfully completed the subsequent tests under operating conditions that by analysis bounded the most probable maximum forces expected from anticipated operational occurrences 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 WBN, Units 1 and 2. Also, the pressure boundary component design criteria were not exceeded. Analysis and review of the test results and the Applicant's justifications indicated direct applicability of the prototypical valve and valve performance to the in-plant valves and systems covered by the generic test program. The Applicant's analysis showed the plant specific piping was acceptable.

Thus, the Applicant met the requirements of Item II.D.1 of NUREG-0737 (Items 1-8 of Section 1.2 in this report). Therefore, the Applicant 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). December 26, 1994, and ending January 29, 1995.

Lockheed Idaho performed this review for both WBN, Units 1 and 2. However, the applicability of this review to Unit 2 depends on the Applicant verifying that the Unit 2 as-built system conforms to the Unit 1 design reviewed in this report.

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NRC FORM 335 (2-89) NRCM 1102, 3201, 3202

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

INEL-94/0277

2. TITLE AND SUBTITLE

TECHNICAL EVALUATION REPORT
TMI ACTION--NUREG 0737 (II.D.1)
RELIEF AND SAFETY VALVE TESTING
WATTS BAR NUCLEAR PLANT - UNITS 1 AND 2
DOCKET NOs. 50-390 AND 50-391

3. DATE REPORT PUBLISHED

MONTH	YEAR
February	1995

4. FIN OR GRANT NUMBER

L1354

5. AUTHOR(S)

C. P. FINEMAN

6. TYPE OF REPORT

Technical Evaluation Report

7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION – NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Idaho National Engineering Laboratory
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9. SPONSORING ORGANIZATION – NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, DC 20555

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

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 development and completion of programs to do two things. First, they should reevaluate the functional performance capabilities of pressurized water reactor safety, relief, and block valves. Second, they should 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 Lockheed Idaho Technologies Company. Specifically, this report documents the review of the Watts Bar Nuclear Plant, Units 1 and 2, Applicant response to the requirements of NUREG-0578 and NUREG-0737. This review found the Applicant provided an acceptable response reconfirming they met General Design Criteria 14, 15, and 30 of Appendix A to 10 CFR 50 for the subject equipment. It should also be noted Lockheed Idaho performed this review for both Units 1 and 2. However, the applicability of this review to Unit 2 depends on verifying that the Unit 2 as-built system conforms to the Unit 1 design reviewed in this report.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

13. AVAILABILITY STATEMENT

Unlimited

14. SECURITY CLASSIFICATION

(This Page)

Unclassified

(This Report)

Unclassified

15. NUMBER OF PAGES

16. PRICE

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13. **AVAILABILITY STATEMENT.** Denote public releasability, for example "unlimited", or limitation for reasons other than security.

14. **SECURITY CLASSIFICATION.** Enter U.S. Security Classification in accordance with U.S. Security Regulations (i.e., unclassified).

15. **NUMBER OF PAGES.** Leave blank. (Added by NTIS)

16. **PRICE.** Leave blank. (Added by NTIS)

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January 25, 1995

50-390/391

Mr. Lawrence K. Cohen
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Technical Assistance Management Section
Planning, Program, and Management Support Branch
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Larry:

SUBJECT: WATTS BARR EVACUATION TIME ESTIMATE REVIEW

Enclosed is PNL's review submittal of the revised report on "Evacuation Time Estimates within the Plume Exposure Pathway Emergency Planning Zone" for the Watts Bar Plant dated March 3, 1994. This review was performed against NUREG-0654, FEMA-REP-1, Rev. 1.

If you have any questions or problems regarding the above, please do not hesitate to give me a call at (509) 375-2065. Thank you.

Sincerely,

Eva Eckert Hickey, Technical Group Leader
Operational Health Physics
Health Protection Department

EEH:cps
Attachment
cc: Amira M. Gill, NRC

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Ed Fox, Jr.
Emergency Preparedness Branch
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Fox:

I have reviewed the revised report "Evacuation Time Estimates within the Plume Exposure Pathway Emergency Planning Zone" for the Watts Bar Plant dated March 3, 1994. The review is based on the guidance of NUREG-0654, FEMA-REP-1, Rev.1. The report is consistent with the guidance. The report adequately addresses the concerned expressed in my September 8, 1993 review of the March 1993 report.

If you have any specific questions or need further elaboration, please let me know.

Sincerely yours,

Thomas Urbanik II

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