PSNH

SEABROOK STATION
Engineering Office

Public Service of New Hampshire

New Hampshire Yankee Division

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B7.1.3

United States Nuclear Regulatory Commission
Washington, DC 20555

Attention: Mr. Vincent S. Noonan, Project Director
PWR Project Directorate No. 5

References: (a) Construction Permits CPPR-135 and CPPR-136, Docket Nos. 50-443 and 50-444
(b) PSNH Letter, J. DeVincentis to G. W. Knighton, "Compliance with NUREG-0737: Clarification of TMI Action Plan Requirements," dated October 10, 1985

Subject: NUREG-0737 Task II.D.1, "Performance Testing of Boiling Water Reactor and Pressurized Water Reactor Relief and Safety Valves"

Dear Sir:

In Reference (b), we indicated that relief and safety valves representative of Seabrook's valves were being tested in the EPRI PWR Safety and Relief Valve Test Program. Furthermore, we indicated that Seabrook would submit evaluations and other plant-specific data as EPRI's program progressed and information became available.

Enclosed herewith, please find Seabrook's response to Task II.D.1 (Attachment 2) and marked-up FSAR Page 1.9-12 (Attachment 1) which indicates Seabrook's compliance with NUREG-0737, "Clarification of TMI Action Plan Requirements." The marked-up FSAR page will be incorporated into the FSAR by a future amendment.

We have provided responses to the NUREG-0737 Task II.D.1 positions by referencing applicable EPRI reports and recalling plant-specific data. Our results show that Seabrook Station's design is conservatively enveloped by the EPRI test results.

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Should you or your staff have any questions, please do not hesitate to contact us. We do request that the acceptability of this item be reflected in the next supplement to Seabrook Station's SER.

Very truly yours,

John DeVincentis

John DeVincentis, Director
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Attachments

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Task II.D.1 Performance Testing of Boiling Water Reactor and Pressurized Water Reactor Relief and Safety Valves (NUREG-0737)

Position:

Pressurized water reactor licensees and applicants shall conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents.

Response: Seabrook complies with Task II.D.1. Refer to SBN- , dated February , 1986, for a discussion of testing to qualify Seabrook's Reactor Coolant System relief and safety valves.

By letter dated July 1, 1981, R. C. Youngdahl (Consumers Power) transmitted the Interim Data Report for the EPRI PWR Safety and Relief Valve Test Program. This report summarizes the test data collected to date on relief and safety valves. The Seabrook Station units each have two Garrett Model Number 3750014 relief valves and three Crosby Model Number DS-C-56964 safety valves. Relief and safety valves representative of the above valves are being tested in the EPRI Program. Seabrook will submit evaluations and other plant specific data on a schedule consistent with the R. C. Youngdahl letter of December 15, 1980, and modified on July 1, 1981.

Task II.D.3 Direct Indication of Relief and Safety Valve Position (NUREG-0737)

Position:

Reactor coolant system relief and safety valves shall be provided with a positive indication in the Control Room derived from a reliable valve position detection device or a reliable indication of flow in the discharge pipe.

Response:

See FSAR Sections 5.2.2 and 7.5.

Task II.E.1.1 Auxiliary Feedwater System Evaluation (NUREG-0737)

Position:

The Office of Nuclear Reactor Regulation is requiring re-evaluation of the Auxiliary Feedwater (AFW) systems for all PWR operating plant licensees and operating license applications. This action includes:

(1) Perform a simplified AFW system reliability analysis that uses event-tree and fault-tree logic techniques to determine the potential for AFW system failure under various loss-of-main-feedwater transient conditions. Particular emphasis is given to determining potential failures that could result from human errors,

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ATTACHMENT 2

NUREG-0737, Task II.D.1, "Performance Testing of Boiling Water Reactor and Pressurized Water Reactor Relief and Safety Valves"

BACKGROUND

NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations," generated a call for full-scale testing of relief and safety valves utilized in the Reactor Coolant System (Section 2.1.2). NUREG-0660, Item 2.D, "Reactor Coolant System Relief and Safety Valves," listed the initial testing requirements. Finally, NUREG-0737, Item II.D.1, "Performance Testing of Boiling Water Reactor and Pressurized Water Reactor Relief and Safety Valves," determined the parameters for testing, and provided clarification to the action plant under NUREG-0660.

DISCUSSION

I. Description of Seabrook Station Safety and Relief Valves and Piping

The Seabrook Station design employs three Crosby Model Number HB-BP-86, Size 6M6, Safety Relief Valves; two Garret Model Number 3750014 Power-Operated Relief Valves; and two Westinghouse Model Number 03003GM99FNH00G ("99 series") Block Valves. The inlet piping configurations are of the "short" length type and do not include loop seals.

The above piping configurations were not the original plant designs, but rather are the result of an extensive redesign of the inlet and discharge piping and an installation by retrofit of the above mentioned valves. These changes are:

a. Block Valves

The original block valves (RC-V-122, 124) were 3" Copes-Vulcan D-100-160 Globe Valves. These valves required replacement due to the lack of a qualified air supply. The replacement was the Westinghouse 3" Gate Valve, 03003GM99FNH00G, with a Limitorque SMB-000-10 Operator. This operator was found to be undersized and would not stroke fully closed under operating conditions (per the EPRI Test Report, discussed in Section III.B of this report). The valve was modified by an operator replacement with an SB-00-15 Model which was proven capable of full function under operating conditions.

b. Power-Operated Relief Valves

The original relief valves (RC-PCV-456A, 456B) were 3" Copes-Vulcan D-100-160 Globe Valves. Seabrook now employs Garret Power-Operated Relief Valves (PORV), Part No. 3750014. This valve is typical for Westinghouse installations. It is a 3" x 6" (inlet x discharge) "Y"-Pattern valve and is classified as the "straight through" design as opposed to the right-angle model used by C-E plants.

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c. Safety Valves

The safety valves (RC-V115, 116, 117) are Crosby HB-BP-86, Size 6M6. The valve employs a size "M" orifice. The original design was for loop-seal internals, using stellite to seal against water. The valves were modified when the inlet pipe loop-seals were deleted (Paragraph d below) and the internals were substituted with steam internals composed of ASTM-A637.

d. Inlet Piping to Safeties

The original piping configurations from the pressurizer to the three safeties were "long" runs, ranging from 13' to 15' each, with several elbows and a loop seal. The purpose of the loop seal was to collect condensed steam at the valve inlet. This allowed the use of water internals in the valve which have better sealing and maintenance properties than steam internals. However, the slug of water that collects in the loop can cause downstream dynamic loads ten times greater than steam loads upon valve cycling. Additionally, the long inlet piping proves to be problematic during two-phase flow and water flow, causing inlet piping pressure oscillations, and more importantly, valve chatter.

By modifying the inlet piping to approximately 2' of nearly vertical run, water slugs are eliminated and water/steam transition flow is much smoother by reducing two-phase flow.

e. Discharge Piping from Safety Relief Valves

Two Barco ball joints were added to each discharge line downstream of the safeties. These joints were added to reduce nozzle loads in lieu of thermal expansion loops which could not be physically run in the crowded pressurizer cubicle.

f. Discharge Piping from the Power-Operated Relief Valve Header

The two relief valves discharge to a common header. A thermal expansion loop was added downstream of the header to reduce the transient loads. Ball joints were not used for this application due to the availability of space in the pressurizer cubicle for this one expansion loop.

II. The EPRI Test Report

The EPRI Test Report is the basis of our conclusion that the valves and piping in the Seabrook Overpressure Protection System will operate under accident conditions. We believe that the operating conditions, piping configurations, and range of fluid qualities used in the test runs envelope all the expected operating conditions for Seabrook's Westinghouse NSSS.

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The EPRI Test Report was transmitted to the NRC by a letter, dated April 1, 1982, from D. P. Hoffman, Chairman of the PWR Safety and Relief Valve Test Program Subcommittee.

III. NUREG-0737, Item II.D.1 - Position/Response

A. "Performance Testing of Relief and Safety Valves"

(1) Position: "Evidence supported by test of safety and relief valve functionality for expected operating and accident (non-ATWS) conditions must be provided to NRC. The testing should demonstrate that the valves will open and reclose under expected flow conditions."

Response: The results of testing of Crosby Safety Relief Valve Model HB-BP-86, Size 6M6, are found in Reference (k). This test used loop-seal internals and a long inlet (Seabrook uses steam internals and a short inlet). While the difference in internals is mostly a materials factor and affects only the leakage properties of the valve, the additional References (j) (short inlet) and (l) (steam internals) are applied here to demonstrate performance of the existing Seabrook hardware. Seventeen full scale tests were performed with a nominal set pressure of 2,500 psia and a long inlet (Reference (k)). For steam and transition tests with the loop-seal drained (most closely corresponding to Seabrook's configuration), the valves performed satisfactorily on the manufacturer's recommended ring settings. For saturated water tests (650°F), the valve operated with stable performance in the first test, and with chatter developing into stable performance on the second test. For subcooled water tests (550°F), the valve chattered necessitating termination of the test. This chatter on passing subcooled water with high back-pressure is characteristic of Crosby Safety Relief Valves (References (j), (k), and (l)). However, it is our engineering judgement that this does not pose a significant threat to maintaining proper RCS pressure and inventory as subcooled water can best be passed by the PORVs without challenging the safeties. Therefore, we conclude that the Crosby Safety Relief Valves will open and reclose under the expected flow conditions.

The results of testing of the Garret Model 3224718-2 PORV, which is representative of Seabrook's Model 3750014, are presented in Reference (m). The similarity justifications are presented in Reference (n), Appendix B6. A total of ten tests were performed on the Garret PORV. The tests were performed under steam, water, preload, transition, and water seal conditions. The valve fully opened and fully closed during all ten tests.

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(2) Position: "Since it is not planned to test all valves on all plants, each licensee must submit to NRC a correlation or other evidence to substantiate that the valves tested in the EPRI or other generic test programs demonstrate the functionability of as-installed primary relief and safety valves. This correlation must show that the test conditions used are equivalent to expected operating and accident conditions as prescribed in the Final Safety Analysis Report (FSAR). The affect of as-built relief and safety valve discharge piping on valve operability must be accounted for, if it is different from the generic test loop piping."

Response: The valves and piping configurations of the Seabrook designs are typical of the installations at Westinghouse's 37 4-loop, 17 x 17 array, 3,423 MWt plants (Seabrook 1 and 2 are in this group). Therefore, it is no coincidence that the valves and piping at Seabrook are applicable to the EPRI Test Report, as EPRI chose to select hardware that adequately represented the range commonly found in most PWRs.

The justifications for the valves that EPRI chose to test are found in Reference (n).

Comparison Table

Safety Valves	PORV	Block Valve
Seabrook:		
Crosby HB-BP-86, 6M6 Short Inlet Steam Internals	Garret 3750014	Westinghouse 03003GM99FNH00G "99 Series"
EPRI:		
1. Crosby HB-BP-86, 6M6 Long Inlet Water Internals	Garret 3224718-2	Westinghouse 03001GM99FNH02000 "99 Series"
2. Crosby HB-BP-86, 3K6 Short Inlet Steam/Water Internals		
3. Crosby HB-BP-86, 6N8 Long Inlet Steam Internals		

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The limiting plant transients were chosen in Reference (p) for Westinghouse plants. These transients represent expected plant operations (Condition II - Faults of Moderate Frequency) and postulated incidents (Condition IV - Limiting Faults). With the parameters of RC pressure and rate of pressure increase chosen, sensitivity studies were performed in which safety valve setpoints and valve flow rates were varied, and a number of safety valves were assumed stuck open (Reference (p)).

In those events in which high safety valve setpoints and low flow rates were assumed, it was found that the Westinghouse NSSS (Reference 4-loop plant identical to Seabrook) did not overpressurize. In those events in which one or more safety valves were assumed stuck open, the core remained covered.

Table 4-7 from Reference (q) presents the fluid conditions used in testing the Crosby 6M6 (loop-seal internals). This test data is applied to Seabrook's 6M6 with steam internals (see previous comment concerning this applicability in the response to A(1) and also in Reference (q), Section 4.6). These tests were performed with a long inlet, which is conservative to our analysis. Section 4.7 of Reference (q) justifies the adequacy of the range of tests to be representative of FSAR events.

It is exceedingly unlikely that piping configurations used at steam facilities such as Wyle's or Marshall's would be identical for any plant's particular configuration, let alone all PWRs. Therefore, instead of a comparison of the physical pipe-runs (i.e. length, diameter, number of elbows, etc.), we will examine the affect of as-built piping from the standpoints of (nozzle loads) backpressure and transient loads.

The safety relief valve discharge lines leading to the pressurizer relief tank header are equipped with ball joints. The PORV discharge lines and PORV header are provided with expansion loops to accommodate thermal displacements. Two thermal conditions have been analyzed (Reference (r)):

1. Normal Operating Condition -

    - o 659^{o}F upstream of valves
    - o ambient downstream of valves

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2. Safety Relief Valve/PORV Discharge Conditions -

- 673°F for all lines
- Steam Generator Nozzle displacements are:

  2.551" vertical
  0.156" radial

- Steam Generator Lug (At Springline) displacement:

  2.278" vertical

The breaking resistance of the ball joints during a transient is 2,000 ft.-lbs. Piping stresses in the safety relief valve discharge lines due to this resiting moment are on the order of 2,000 psi (approximately 10% of the allowable). The effect of this resisting moment has been considered in the evaluation of the safety relief valve discharge flange loading. During steady state, bending resistance is negligible.

The PORVs discharge to a bullhead tee header which is supported by the pressurizer support structure. The thermal expansion loop is downstream of the header; therefore, resisting moment due to the thermal expansin loop is not transmitted back up the line past the header.

The backpressures listed in the testing of the Crosby 6M6 given by Table 4-7 of Reference (q) range from 245 psia to 725 psia. However, those tests with high backpressures (>700 psia) can be attributed to the clearing of a loop-seal, which is not applicable to Seabrook. Test No. 932 lists a backpressure of 650 psia; however, this was a test for long-term safety injection and the valve was passing 463°F water. Since the pressurizer relief tank is equipped with 75 lb. rupture disks, such a high backpressure should not be achieved at Seabrook. Therefore, considering the similarity in sizes of the test piping (Reference (k), Pages 2-6) to the installed piping (6" line discharging into a 12" line) yet without the backpressure orifice which is employed in the test line, but not the Seabrook installed line, we believe that the backpressures in Table 4-7 are conservative and that the discharge piping at Seabrook will not adversely affect the performance of the Crosby Safety Relief Valves.

The backpressures listed in the testing of the Garret 3224718-2 PORV given by Table 3-31 of Reference (q) range from 25 psia to 875 psia. However, those tests with high backpressures (>800 psia) can be attributed to the clearing of a loop-seal, which is not applicable to Seabrook. Therefore, if we assume that the greatest contributor to backpressure in the test line is the backpressure orifice (Reference (s), Pages 2-11) which the installed line does not employ, then the backpressures in Table 3-31 of Reference (q) are conservative and that the discharge piping at Seabrook will not adversely affect the performance of the Garret PORV.

Transient loads in the discharge piping for Seabrook Station are not considered applicable to the measure of scrutiny necessary at other PWRs due to the fact that we employ no loop seals. Loop seals cause a high discharge pressure peak due to the passing of the loop seal water slug. Transient loads in the discharge piping at Seabrook are 10 times less than if we had employed loop seals.

(3) Position: "Test data including criteria for success and failure of the valves tested must be provided for NRC Staff review and evaluation. These test data should include data that would permit plant-specific evaluation of discharge piping and supports that are not directly tested."

Response: Reference (p) establishes an acceptable safety valve performance which will not pose a direct challenge to the safe operation of a PWR power plant based on conventional safety analysis methodology. Reference (k) and Reference (m) give the test data for the safety relief valve and the PORV, respectively.

Plant-specific evaluation of the discharge piping was given in Reference (r); the computer output and stress summaries are available for further review.

B. "Qualification of PWR Block Valves"

"Position: Qualification of PWR Block Valves - Although not specifically listed as a short-term lessons-learned requirement in NUREG-0578, qualification of PWR block valves is required by the NRC Task Action Plan NUREG-0660 under Task Item II.D.1. It is the understanding of the NRC that testing of several commonly used block valve designs is already included in the ge eric EPRI PWR Safety and Relief Valve Testing Program to be completed by July 1, 1981. By means of this letter, NRC is establishing July 1, 1982 as the date for verification of block valve functionability. By July 1, 1982, each PWR licensee, for plants so equipped, should provide evidence supported by test that the block or isolation valves between the pressurizer and each power-operated relief valve can be operated, closed, and opened for all fluid conditions expected under operating and accident conditions."

Response: The results of testing of the Westinghouse Block Valve, Model No. 03001GM99FNH02000, are found in Reference (o). The original test valve had a Limitorque SMB-000-10 operator which would not fully close against full flow during preliminary tests. Westinghouse replaced the original Limitorque operator with an SB-00-15 operator which allowed the valve to function acceptably for the remainder of the test cycles.

Seabrook's Block Valves had the same modification of the Limitorque operators at the request of Westinghouse. Therefore, we conclude that the modified Seabrook block valves will operate properly under operating and plant conditions.

C. "ATWS Testing"

Position: "Although ATWS testing need not be completed by July 1, 1981, the test facility should be designed to accommodate ATWS conditions of approximately 3,200 to 3,500 (Service Level C pressure limit) psi and 700^{o}F with sufficient capacity to enable testing of relief and safety valves of the size and type used on operating pressurized-water reactors."

Response: No ATWS testing was performed for Seabrook Station.

CONCLUSION

Complete testing of PWR Safety Relief, PORV, and Block Valves has been concluded by EPRI. The operating conditions expected by a Westinghouse NSSS, namely our reference 4-loop plant, had been introduced into the test program, and the valves performed their intended safety operations.

We have provided responses to the NUREG-0737 - II.D.1 positions by referencing applicable EPRI Reports and recalling plant-specific data. Our results show that Seabrook Station's design is conservatively enveloped by the EPRI test results.

NUREG-0737, Task II.D.1, "Performance Testing of Boiling Water Reactor and Pressurized Water Reactor Relief and Safety Valves"

REFERENCES

(a) NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations," July 1979
(b) NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," May 1980
(c) NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980
(d) UE&C Drawing No. 9763-F-805007, Revision 13, "Reactor Coolant System - Pressurizer - P&I Diagram"
(e) UE&C Drawing No. 9763-F-805135, Revision 9, "Containment Structure - Piping in Zone 58E - Plants and Sections"
(f) UE&C FP 220-50445, Crosby Drawing No. DS-C-56964, Revision C, "Nozzle Type Safety Valve"
(g) UE&C FP 220-97657, Crosby Drawing No. DS-C-56964-8, Revision 0, "Nozzle Type Safety Valve"
(h) UE&C FP 220-56786, Westinghouse Drawing No. 1D99836, "Motor-Operated Gate Valve"
(i) EPRI/CE PWR Safety Valve Test Report, Volume 1: Summary, January 1983
(j) EPRI/CE PWR Safety Valve Test Report, Volume 5: Test Results for Crosby Safety Valve (HB-BP-86 3K6), March 1983
(k) EPRI/CE PWR Safety Valve Test Report, Volume 6: Test Results for Crosby Safety Valve (HB-BP-86 6M6), March 1983
(l) EPRI/CE PWR Safety Valve Test Report, Volume 7: Test Results for Crosby Safety Valve (HB-BP-86 6N8), March 1983
(m) EPRI/WYLE Power-Operated Relief Valve Phase III Report, Volume 11: Summary of Phase III Testing of the Garret Relief Valve, October 1982
(n) EPRI PWR Safety and Relief Valve Test Program: Valve Selection/ Justification Report, March 1982
(o) EPRI - Marshall Electric Motor-Operated Valve (Block Valve) Interim Test Data Report, July 1982
(p) EPRI - Safety Valve Contingency Analysis in Support of the EPRI S/RV Test Program, Volume 3: Westinghouse Systems, October 1981
(q) EPRI PWR Safety and Relief Valve Test Program: Test Condition Justification Report, June 1982
(r) SBU-76579, "Piping Analysis for Seabrook Pressurizer Safety and Relief Valve Piping," dated August 5, 1983
(s) EPRI/Wyle Power-Operated Relief Valve Phase III Test Report, Volume 1: Test System Description, October 1982