Docket No. 50-397

JUL 30 1985

Mr. G. C. Sorensen, Manager
Regulatory Programs
Washington Public Power Supply Systems
P.O. Box 968
3000 George Washington Way
Richland, Washington 99352

Dear Mr. Sorensen:

SUBJECT: WNP-2 TMI ITEM II.D.1, SAFETY AND RELIEF VALVE TESTING

The staff, in conjunction with its review contractor, Idaho National Engineering Laboratory (EG&G), has completed its review of the Safety and Relief Valve Testing Program for WNP-2. The staff finds that the Safety and Relief Valve Testing Program and Procedures for WNP-2 are acceptable.

This completes the staff's action for TMI Item II.D.1 and closes the Post Trip Review issue for WNP-2. A copy of the related Safety Evaluation is enclosed.

Sincerely,

Walter R. Butler, Chief
Licensing Branch, No. 2
Division of Licensing

Enclosure: As stated

cc: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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SAFETY EVALUATION REPORT
TMI ACTION--NUREG-0737 (II.D.1)
RELIEF AND SAFETY VALVE TESTING
FOR
WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PLANT NO. 2
DOCKET NO. 50-397

1. INTRODUCTION

1.1 Background

Light water reactor experience has included a number of instances of improper performance of relief and safety valves installed in the primary coolant systems. There have been instances of valves opening below set pressure, valves opening above set pressure and valves failing to open or reseat. From these past instances of improper valve performance, it is not known whether they occurred because of a limited qualification of the valve or because of a basic unreliability of the valve design. It is known that the failure of a power-operated relief valve to reseat was a significant contributor to the TMI-2 sequence of events; however, such an event in a Boiling Water Reactor (BWR) would not have the same severe consequences. Nevertheless, these facts led the task force which prepared NUREG-0578[1] to recommend that programs be developed and executed which would reexamine the performance capabilities of BWR safety and relief valves for unusual but credible events. These programs were deemed necessary to reconfirm that the General Design Criteria 14, 15 and 30 of Appendix A to Part 50 of the Code of Federal Regulations, 10 CFR are indeed satisfied.

1.2 General Design Criteria and NUREG Requirements

General Design Criteria 14, 15, and 30 require that (1) the reactor primary coolant pressure boundary be designed, fabricated and tested so as to have an extremely low probability of abnormal leakage, (2) the reactor coolant system and associated auxiliary, control and protection systems be designed with sufficient margin to assure that the design conditions are not exceeded during normal operation or anticipated transient events and (3) the components which are part of the reactor coolant pressure boundary shall be constructed to the highest quality standards practical.

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

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

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

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

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

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

6. Test data including criteria for success or failure of valves tested must be provided for Nuclear Regulatory Commission (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.

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

2. BWR OWNERS' GROUP RELIEF AND SAFETY VALVE PROGRAM

To respond to the NUREG requirements listed above, the BWR Owners' Group contracted the General Electric Company (GE) to design and conduct a Safety/Relief Valve Test Program.[3] The program describes the safety/relief valves to be tested, the test facility requirements, the test sequence, the valve acceptance criteria and the procedure for obtaining, analyzing and reporting the test data. Prior to its acceptance, the test program received extensive NRC review and comment followed by responses from the GE/BWR Owners' Group. Six NRC questions and Owners' Group responses dealing with justification of the applicability of test results to the in-plant safety/relief valves are contained in the enclosure to Reference 4. The NRC review of the response to these questions is contained in Reference 5. Based on this review, the concerns expressed in the questions were appropriately resolved.

The early BWRs contain a combination of dual function safety/relief valves (SRV), power actuated relief valves (PARV) and single function safety valves (SV). At the Washington Nuclear Plant No. 2 (WNP-2), there are 18 dual function SRV's. There are no PARV's or SV's at the WNP-2.

The qualification of the SRVs for steam discharge under expected operating and accident conditions has been demonstrated by vendor production tests and is confirmed routinely by in-plant startup and operability tests. Based on this, it was agreed that the valves should be tested for those events that result in liquid or two-phase flow at the SRV.

The test sequence and conditions established in the test program were based on an evaluation of expected operating conditions determined through the use of analyses of accident and anticipated operational occurrences referenced in Regulatory Guide 1.70, Rev. 2. Enclosure 2 to Reference 3 provides this evaluation which indicated that there is one event which is significantly likely to occur and can lead to the discharge of liquid or

two-phase flow from the SRVs. This event combined with the single failure requirement of NUREG 0737 results in the conclusion that a test should be performed simulating the alternate shutdown cooling mode which utilizes the SRVs as a return flow path for low pressure liquid to the suppression pool.

At a meeting on March 10, 1981,[6] the BWR Owners' Group presented results of a study by Science Applications, Inc. (SAI) which showed that the probability of getting liquid to the steamline, and hence to the SRV's, is approximately 10^{-2} per reactor year. However, even if the water level increases to the mid-plane of the steam line nozzle on the vessel, which is not likely,[a] the fluid quality at the valve was calculated by GE to be greater than 20%.[3] Because the steam lines typically drop about 45 feet vertically from the vessel nozzles to the horizontal runs on which the SRVs are mounted, much of the liquid which gets to the steam lines would be entrained as droplets. Therefore, the two-phase mixture upstream of the SRVs, should liquid reach the level of the steam lines, would exist as a froth, droplet, annular or stratified flow regime, and slug flow or subcooled liquid flow would be unlikely.

Even if two-phase discharge through a SRV should result in a stuck open valve, the results of the blowdown are not severe. As discussed in Reference 7, historically there have been a total of 53 inadvertent blowdown events due to pressure relief system valve malfunctions from 1969 through April 1978. These events varied in consequences from a short duration pressure transient to a rapid depressurization and cooldown of the primary coolant system from approximately 1100 psig to a few hundred psig. No fuel failures due to these transients have been reported.

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a. Feedwater pumps would be tripped prior to the water level reaching the mid-plane by the L8 high level trip, turbine vibration trip, or by operator action.

In Reference 8, the BWR Owners' Group discusses the consequences of the worst case transient for maintaining the core covered (loss of feedwater) combined with the worst single failure (failure of the high pressure injection system) and one stuck open relief valve. Reference plant analyses for a BWR/4 and BWR/5 show that the Reactor Core Isolation Cooling (RCIC) system can automatically provide sufficient inventory to keep the core covered. The capability is not a design basis for the RCIC system and not all plants have been analyzed to demonstrate this capability. If a plant should not have this capability, manual depressurization to low pressure core cooling systems will avoid core uncovery for the case of loss of feedwater plus worst single failure plus a stuck open relief valve. Therefore, even for the loss of feedwater transient with the worst single failure, a stuck open relief valve does not uncover fuel.

At the March 10, 1981 meeting,[6] the BWR Owners' Group presented an analysis that showed that even if a slug of subcooled water exists upstream of the SRVs, the probability of rupturing the discharge line is 7×10^{-4} per event. The Staff has not reviewed the supporting analysis for this value; however, even if the failure probability is as high as 10^{-2} per event, the combined probability is no greater than for a steam line break inside containment. GE states that the steam line break, which has been analyzed and found to be acceptable, would be more severe (effects on the core and containment) than a break in a SRV discharge line with a stuck open SRV because the assumed Break area is larger.

In summary, based on the BWR operating history of inadvertent SRV blowdowns, the low likelihood of severe consequences, and the bounding design basis steam line break, the staff decided not to require high pressure testing with saturated liquid or subcooled water.

Based on the above, the Licensee has complied with NUREG Requirements 1-4 (Paragraph 1.2 above). That is, an acceptable test program was established which adhered to the Staff guidelines on the

selection of test conditions and the maximization of system loads. That portion of Item 5 dealing with the qualification of the associated control circuitry is considered to be satisfied as a result of the anticipated licensing action for compliance with 10 CFR, Part 50.49.

3. BWR OWNERS' GROUP TEST RESULT AND ANALYSIS

In October 1981, the BWR Owners' Group published a technical report[9] documenting the results of the prototypical safety/relief valve tests conducted in accordance with the accepted Test Program.[3] The tests were performed by the General Electric Company for the BWR Owners' Group at the Wyle Laboratory in Huntsville, Alabama. The test report, which was reviewed by the Staff, describes the test facility, the basis for the test conditions and valve selection, the instrumentation and its accuracy, and analyzes the results with respect to valve operability, piping and support loads and the applicability of the test results to the in-plant safety and relief valves.

With the completion of the testing and the submittal of the test report, the Licensees complied with NUREG Requirement No. 6 listed in 1.2 above. However, the subsequent Staff review of the test results generated six plant specific questions stated in Reference 10 which required resolution. Reference 11, representing the WNP-2 response to the six plant specific questions, was submitted for review on January 18, 1985.

4. REVIEW AND EVALUATION

4.1 Review of Test Results and Analysis

An extensive review[12,13] of the test results[9] was conducted by NRC consultants (EG&G Idaho, Inc.) at the Idaho National Engineering Laboratory. The review addressed not only the test results but also the applicability of the test results and equipment to the WNP-2 safety-relief valve systems. The six plant specific questions generated by the review and the Licensee responses to those questions are discussed in Paragraph 4.4 below.

4.2 Valves Tested

The generic test program required the testing of six different safety/relief valves. Included was a Crosby (6 x R x 10) Safety Relief Valve, Style HB-65-BP. This valve is a direct acting, dual function, spring loaded SRV with no material, dimensional or operational differences compared to the in-plant valves. Thus, the test results are directly applicable to the in-plant valves at the WNP-2.

4.3 Test Conditions

As discussed in Section 2.0 herein, test conditions to envelop the expected BWR Safety/Relief Valve events were developed in accordance with NRC guidelines. They were accepted and are presented in Reference 3. The review of the test results indicates that the actual test conditions were in accordance with the established test program.

4.4 Evaluation of Responses to Plant Specific Questions

The response to Question No. 1 indicates that there are valve discharge line differences between the test configuration and the in-plant configuration. However, it is pointed out that these differences result in bounding loads on the SRV's. The first segment of test piping downstream of the test valve is longer than the comparable in-plant segment (12 ft

vs. 8 ft) which would result in a higher moment at the test valve. Discharge from the tee quencher at the end of the WNP-2 SRV discharge line cannot transmit loads to the valve as the test system could because the in-plant line is anchored between the quencher and the valve. In addition the quencher in each line is anchored. Thus, this portion of the response is considered to be acceptable. The second part of the response addressed the back pressure (dynamic, hydraulic) loads on the test and in-plant valves. The Licensee addressed both transient and steady state back-pressure loads. The steady state back pressure for the test valve was forced to be greater than that expected in-plant by installing a predetermined orifice plate in the discharge line before the ram's head and above the water line. The response also indicated that the high pressure steam test preceding the low pressure water test would produce the greater transient back pressures between the two tests. This would be true due to the higher pressure upstream of the SRV and the shorter valve opening time. Although the submergence length of the in-plant discharge line is greater than at the test facility (17.4 ft vs 13 ft), this increasing back pressure effect will be offset by the longer in-plant discharge line compared to the test facility (135 ft vs 112 ft).

Based on the above discussion, the response to the first question is considered by the Staff to be acceptable.

The response to the second question described the support system components in the WNP-2 discharge lines indicating that spring hangers do exist at the WNP-2 whereas the test facility piping did not include spring hangers. The basic argument defending the adequacy of the spring hangers (in fact, all supports) is that they were designed for the much larger, high steam pressure relief valve opening loads. In this case, therefore, sufficient margin is available in the in-plant spring hangers to account for the additional load due to the dead weight in the water-filled, low pressure event. The test results indicated significantly lower dynamic loads during the water discharge event than during the high pressure steam

discharge case and the point made in this response (as well as in the response to Question No.1) is that the test program was designed primarily to demonstrate valve and system adequacy under the prototypical water discharge events (i.e., the alternate shutdown cooling mode).

Thus, with the in-plant safety/relief valve discharge piping and support system designed for the high pressure steam discharge event and with the satisfactory response of the test valves, the discharge piping and support system to the low pressure water blowdown, the reply to the second question is considered by the Staff to be acceptable.

The third question inferred that, during testing, there may have been valve functional deficiencies or anomalies encountered that invalidated test runs and were not reported in the test results because subsequent valid test runs were obtained. The response to this question states, "All the valves subjected to test runs, valid or invalid, opened and closed without loss of pressure integrity or damage." This statement is supported with the submittal of the Wyle Laboratory test log sheets for the Crosby Valve tests. Thus, the Staff finds the response to Question No. 3 to be acceptable.

Question No. 4 asked the Licensee to describe and compare expected events at WNP-2 with the test conditions of the generic test program. The Licensee summarizes the analysis procedure(3) using Regulatory Guide 1.70 which arrived at 13 events that would result in liquid or two-phase flow through the SRV's and maximize the dynamic forces on the valve. As indicated in Section 2.0 herein, this analysis concluded that the alternate shutdown cooling mode is the only expected event which will result in liquid at the valve inlet. To simulate this event the test program(3) used a 15-50°F subcooled liquid at 20-250 psig at the SRV inlet prior to valve opening. The Licensee indicates that the fluid/flow conditions tested conservatively bound the WNP-2 conditions expected for the alternate shutdown cooling mode of operation. The Licensee's response to the fourth question is acceptable to the Staff.

The fifth question addresses the effect on valve performance of steam flow cycling of the valves prior to the low pressure liquid flow event. The sequence to arrive at the alternate shutdown cooling mode is described in the response. It indicates that the safety/relief valve would be cycled under steam conditions to maintain a 100°F cooldown rate. The test program and, of course, the actual tests included only one steam cycle, the purpose of which was to bring the valve up to the proper service temperature prior to the low pressure liquid test. Thus, any adverse effect of several high pressure steam cycles on valve performance during the liquid test was not included. The response indicates that the valve vendors subject their valves to steam flow cycling and that no loss of valve performance has been noted. The response to this question is acceptable to the Staff.

The response to the sixth question addresses the determination and future use of the valve flow coefficient, C_v. The response indicates that the value of the liquid flow coefficient, in itself, is not of direct interest. The flow capacity of the valves as measured during the tests is the data of interest. The flow capacity of the system SRV's is larger than the capacity of the coolant source pump of the residual heat removal (RHR) system and therefore sufficient to remove decay heat. The answer to this question is considered to be acceptable to the Staff.

Considering the above evaluations, the Staff finds that the Licensee for the Washington Nuclear Plant No. 2 has provided an acceptable response to NUREG Item 7 and to the piping and support concerns of NUREG Item 5 (Paragraph 1.2 herein).

4.5 Supporting Information-High Pressure Steam Flow/Discharge Piping Response

The applicability of the response of the safety-relief valve discharge piping system to the response of the in-plant piping system has been accepted above. In the test report,[9] it is indicated that, (1) the analytically predicted response of the test piping and supports was comparable to the measured values, and (2) the maximum test piping response to liquid flow was

generally less than 30% of that due to test steam flow conditions. The adequacy of the WNP-2 SRV discharge piping under high pressure steam discharge loads was investigated as part of the staff's normal licensing review.

5. EVALUATION SUMMARY

The Licensee for the Washington Nuclear Plant No. 2 has provided an acceptable response to the requirements of NUREG-0737, and thereby, reconfirmed that the General Design Criteria 14, 15 and 30 of Appendix A to 10 CFR-50 have been met. The rationale for this conclusion is given below.

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

Thus, the requirements of Item II.D.1 of NUREG 0737 have been met (Items 1-7 in Paragraph 1.2) and, thereby, assure that the reactor primary coolant pressure boundary will have, by testing, a low probability of abnormal leakage (General Design Criterion No. 14) and that the reactor primary coolant pressure boundary and its associated components (piping, valves and supports) have been designed with sufficient margin such that design conditions are not exceeded during relief/safety valve events (General Design Criterion No. 15).

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

REFERENCES

1. TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations, NUREG-0578, July 1979.

2. Clarification of TMI Action Plan Requirements, NUREG-0737, November 1980.

3. Letter, D. B. Waters (BWR Owners' Group), to Richard H. Vollmer (NRC) "NUREG-0578 Requirement 2.1.2 - Performance Testing of BWR and PWR Relief and Safety Valves," September 17, 1980.

4. Letter, D. B. Waters, Chairman BWR Owners' Group to D. G. Eisenhut, Director, Division of Licensing, NRR, USNRC, "Responses to NRC Questions on the BWR S/RV Test Program," BWR06-8135, March 31, 1981.

5. Letter, B. F. Saffell to R. E. Tiller, Comments on BWR Owners' Group Responses to NRC Questions on Safety/Relief Valve Low Pressure Program - Saff-95-81, April 23, 1981.

6. Memorandum to Themis P. Speis from Wayne Hodges, "Summary of March 10 Meeting with General Electric to Discuss BWR Liquid Overfill Events," May 1981.

7. Technical Report on Operating Experience with BWR Pressure Relief Systems, NUREG-0462, July 1978.

8. Letter to Darrell G. Eisenhut (NRC) from David B. Waters (BWR Owners' Group), BWROG-80-12, "BWR Owners' Group Evaluation of NUREG-0737 Requirements," December 29, 1980.

9. Analysis of Generic BWR Safety Relief Valve Operability Test Results, General Electric NEDE-2 4988-P, October 1981.

10. Letter from A. Schewencer, USNRC to G. C. Sorensen, Washington Public Power Supply System on NUREG-0737, Item II.D-1, Testing of Relief and Safety Valves, September 14, 1984.

11. Letter, G. C. Sorensen, Washington Public Power Supply System to A. Schwencer (NRR-USNRC), "Nuclear Plant No. 2 Operating License NPF-21, NRC Request for Additional Information Regarding BWROG's Response to NUREG-0737, Item II.D.1," January 18, 1985.

12. Letter, B. F. Saffell to R. E. Tiller, "Review of BWR/GE Safety Relief Valve Test Report (A6356)" Saff-14-82, January 13, 1982.

13. Letter, B. F. Saffell to D. E. Solecki, "Open Questions-BWR/GE Safety/Relief Valve Test Report, BWR Owners' Safety/Relief Submittals (A6356)"-Saff-178-82, May 4, 1982.

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