June 19, 1984

Docket No. 50-219
LS05-84-06-031

Mr. P. B. Fiedler
Vice President & Director
Oyster Creek Nuclear Generating Station
Post Office Box 388
Forked River, New Jersey 08731

Dear Mr. Fiedler:

SUBJECT: SAFETY AND RELIEF VALVE TESTING, NUREG-0737, ITEM II.D.1

Re: Oyster Creek Nuclear Generating Station

We have completed our review of information submitted concerning testing of safety and relief valves for Oyster Creek. We find the information submitted demonstrated the ability of the reactor coolant system relief and safety valves to function under expected operating conditions for design-base transients and accidents as defined under II.D.1. and is therefore acceptable. No further action is anticipated and the staff considers Item II.D.1 to be complete.

Sincerely,
Original signed by Thomas Wambach
for

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosure:
Safety Evaluation Report

cc w/enclosure
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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Enclosure:
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Mr. P. B. Fiedler

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SAFETY EVALUATION REPORT

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

RELIEF AND SAFETY VALVE TESTING

FOR

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

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 Oyster Creek Station there are no dual function SRVs, but there are five PARVs and 16 single function SVs. Nearly all of the problems with these valves have been with the dual function or power actuated valves whose function is to limit anticipated operational transients and prevent the safety valves from relieving into the dry well. The single function safety valves, designed and set to comply with the over pressure protection requirements of the ASME Boiler and Pressure Vessel Code have been essentially failure free. The safety valves used in the early BWRs were of the same size and configuration of those used for many years in fossil fuel plants and therefore backed by many years of experience. Because of this, direct acting single function safety valves were not included in the test program. The valves included in the test program were direct acting dual function safety/relief valves, power actuated relief valves and two and three stage pilot operated safety/relief valves.

The qualification of the SRVs and PARVs 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 or PARV. 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 SRV or PARV as a return flow path for <u>low pressure liquid</u> 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 or PARV, 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 or PARVs 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 or PARV 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 or PARV 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

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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.

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.

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 a BWR/5 show that the Reactor Core Isolation Cooling (RCIC) system can automatically provide sufficient inventory to keep the core covered. This 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 or PARV, 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 valve discharge line with a stuck open SRV or PARV because the assumed break area is larger.

In summary, based on the BWR operating history of inadvertent SRV or PARV 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 Oyster Creek Nuclear Generating Station response to the six plant specific questions, was submitted for review on April 27, 1983.

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 Oyster Creek Nuclear Generating Station 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 Dresser Electromatic 6 x 8 Relief Valve, Model 1525 VX. The in-plant valves, like the test valve, are vented to the atmosphere.

Thus, the tested valve was considered to be applicable to the in-plant valves at the Oyster Creek Nuclear Generating Station.

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 PARV 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 PARVs. The first segment of test piping downstream of the PARV is longer than the comparable in-plant segment (12 ft vs 2 ft) which would result in a higher moment at the test valve. Discharge from the "y" quencher at the end of the Oyster Creek PARV discharge line cannot transmit loads to the valve as the test system could because the in-plant line contains an anchor and bellows between the quencher and the valve. 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 PARV and the shorter valve opening time. Additionally, the test facility discharge line submergence is greater and the total discharge line volume is less than the Oyster Creek discharge lines so that the test facility had a smaller air volume and hence a larger back pressure.

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 Oyster Creek Nuclear Generating Station discharge lines indicating that spring hangers do exist at Oyster Creek 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 PARV 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 by the Wyle Laboratory test log sheet for the Dresser 6 x 8, Model 1525VX Electromatic Relief Valve. 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 Oyster Creek 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 PARVs and maximize the dynamic forces on the valves. 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 safety valve inlet prior to valve opening. The Licensee indicates that the alternate cooling mode of operation at Oyster Creek will result in a relief valve liquid discharge that would be approximately 50°F subcooled at 150 psig. Therefore, the test conditions envelope the expected conditions for this event should it occur at the Oyster Creek Nuclear Generating Station. 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 PARV 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 PARVs 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 Oyster Creek Nuclear Generating Station 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 PARV 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. Further, as part of the initial review, the loads on the in-plant piping and supports due to steam discharge were found to be acceptable by the Staff. It should also be mentioned that the Staff's on-going review of the Mark-I Containment Long Term Program includes a review of the methods of analysis, computer code adequacy and design criteria for PARV or SRV discharge piping and supports for high pressure steam discharge conditions.

5. EVALUATION SUMMARY

The Licensee for the Oyster Creek Nuclear Generating Station 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, D. M. Crutchfield (USNRC) to GPU Nuclear on NUREG-0737, Item II.D-1, Testing of Relief and Safety Valves, December 17, 1982.

11. Letter, Peter B. Fiedler to D. M. Crutchfield (USNRC), "Oyster Creek Nuclear Generating Station, NUREG-0737 Item II.D.1 Safety/Relief Valve Testing," April 27, 1983.

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.