STAFF 9/14/81

Mike Collins

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of)
)
METROPOLITAN EDISON COMPANY, ET AL.) Docket No. 50-289
) (Restart)
(Three Mile Island, Unit 1))

NRC STAFF'S RESPONSE TO LICENSING BOARD'S ORDER TO NRC STAFF OF AUGUST 25, 1981

On August 25, 1981 the Licensing Board issued its "Order to NRC Staff Regarding Board Notification of Unsatisfactory Test Results of Safety Valve." In that Order the Board indicated that it had become aware, via a board notification that was filed in another proceeding,1/ of some unsatisfactory test results for a safety valve of the type installed at TMI-1. Not having received such a notification in the captioned proceeding, the Board requested the Staff to inform it promptly whether notification of this matter by the Staff would have been appropriate in this proceeding, and if not why not.

Also, the board directed the Staff to explain the significance of the unsatisfactory safety valve test results in the context of the proposed findings and issues in this proceeding. The Board expressed a particular interest in the effect, if any, of these test results on the Staff's position that the PORV and associated block valve are not required to mitigate the consequences of any design basis accidents because the pressurizer safety valves provide the required protection.

1/ NRC Board Notification No. 81-20, dated August 11, 1980, that was filed in the McGuire proceeding.

DS07
50/1

DS07

The NRC Staff's response to the Licensing Board's Order is set forth in two documents:

1. The "NRC Staff's Report on Board's Comments Regarding Board Notification of Unsatisfactory Test Results of Safety Valves" that was prepared by John F. Stoltz and Dominic C. DiIanni.

2. The "NRC Staff's Report to the Board on Safety Aspects of EPRI Test Data on Relief and Safety Valves" that was prepared by Edgar G. Hemminger and Walton L. Jensen, Jr.

Copies of those documents and their attachments and copies of the affidavits of Messrs. Stoltz, DiIanni, Hemminger and Jensen are enclosed.

Respectfully submitted,

James M. Cutchin, IV
Counsel for NRC Staff

Dated at Bethesda Maryland
this 14th day of September, 1981

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
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(Three Mile Island, Unit 1)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of "NRC STAFF'S RESPONSE TO LICENSING BOARD'S ORDER TO NRC STAFF OF AUGUST 25, 1981" in the above-captioned proceeding has been served on the following by deposit in the United States mail, first class or, as indicated by an asterisk, by deposit in the Nuclear Regulatory Commission's internal mail system or, as indicated by a double asterisk, by hand-delivery, this 14th day of September, 1981:

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Counsel for NRC Staff

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the matter of	)	
	)	
METROPOLITAN EDISON CO., ET AL.	)	Docket No. 50-289
(Three Mile Island Nuclear	)	
Station, Unit 1)	)	(Restart)

NRC STAFF'S REPORT TO THE BOARD ON SAFETY ASPECTS OF EPRI TEST DATA ON RELIEF AND SAFETY VALVES

By order dated August 25, 1981, the Board directed the staff to explain the significance of unsatisfactory safety valve test results in the context of the proposed findings and issues in this proceeding. The Board is particularly interested in the effect of the test results on the staff's position regarding the PORV and associated block valve.

In a letter dated November 26, 1980 from R. H. Vollmer (NRR) to R. C. Youngdahl (EPRI), the Office of Nuclear Reactor Regulation (NRR) provided comments and requested additional information regarding EPRI's "Proposed Program Plan for the Performance Testing of PWR Safety and Relief Valves", Revision 1, dated July 1, 1980. In that letter, we requested that the PWR Owners make "provision for expeditious transmittal of test results from the PWR Owners to the NRC as individual valve tests are completed" so that we could continuously monitor the progress of the test program. The mechanism agreed to for regular transmittal of results is the EPRI Weekly Report. The report is usually issued on Friday and includes a summary of tests conducted at the various test facilities for the week from the previous Monday through the date of the report. One such report is the one dated June 26, 1981 referred to in the Board's August 25, 1981 Order to the Staff.

The utilities with assistance from the NSSS vendors have the primary responsibility for evaluating the safety significance of a given test result for their specific plant. They are responsible under the regulations to advise NRC if information obtained from the test program reveals an unreviewed safety question for their plant. NRR with assistance from RES and its consultant, EG&G, is reviewing and evaluating each reported test result for potential generic safety significance. The NRC and consultant personnel reviewing the test results are familiar with the basic valve types being tested, a general knowledge of valve and related piping installations in PWR plants and a knowledge of the conservatisms used to design PWR Overpressure Protection Systems. Actions to be taken based on a review of test results that fail a test screening criterion range from consideration of relevance and materiality for Board notification to shut down of plants. An example of a test result with obvious safety significance would be failure of a safety valve to open during a given test sequence. As stated in SECY-81-491 dated August 17, 1981 (attached) although some test screening criteria have not been met, the testing to date has not uncovered problems with safety or relief valves which are considered significant to the safety of operating plants. This same conclusion is applicable to the TMI-1 restart.

In response to the Board's August 25, 1981 Order, TMI-1 plant specific evaluation of the significance of the EPRI test results to date is as follows.

For Dresser relief valves (PORVs) of the type installed at TMI-1, the reported preliminary test results indicate that although the test acceptance criterion were not met for water seal type installations, the PORV's will function in the primary mode (pressure relief) as required. The test results to date indicate that the Dresser PORV's experienced a delay of as much as

70 seconds in closing time due to low or ambient water seal temperatures. The valves closed on their own, however, and on disassembly and inspection no damage was observed which might affect their ability to open or close on demand. These results do not indicate a safety concern with respect to TMI since the TMI plant specific piping does not contain water seals for the PORV's, and since all test results applicable to non-water seal piping configurations were satisfactory for the Dresser PORV.

For Dresser safety valves of the type installed at TMI-1, the preliminary test results indicate a need for additional information regarding the effects of inlet piping configuration, back pressure, and adjusting ring settings on safety valve operation. The test acceptance criteria with respect to flow capacity or stem position were not met for certain predetermined test conditions. Based on the worst case preliminary data point, a maximum stem position of 65% was observed for a high ramp rate, high back pressure steam test with the valve set to the original manufacturer recommended ring settings.

If it were assumed that the TMI-1 installed safety valves were limited to the worst case stem position of 65%, a conservative estimate of approximately 405,000 #/hr. relieving capacity would be available. This estimate is based on the conservative assumption that percent flow is approximately equal to the percent stem position. Sensitivity studies of the required safety valve flow capacities for design basis transients as described in topical report, BAW-10043, "Overpressure Protection for Babcock & Wilcox Pressurized Water Reactors", dated May 1972, indicate that a maximum total safety valve flow capacity of 345,000 #/hr. is required. We, therefore, conclude that sufficient safety valve relieving capacity is available at TMI-1, even based on the worst case preliminary EPRI test data and taking no credit for the 100,000 #/hr. relieving capacity available through the PORV. The staff testimony of Jensen and staff proposed findings on the PORV and block valve are, therefore, unchanged.

It should be noted that the EPRI test data as reported on a weekly basis is preliminary in nature. In general, no conclusions can be made on valve performance based on preliminary, individual test results. It is neither expected nor desirable for utilities to be making adjustments to their safety valves until all testing under all conditions has been completed with the results fully evaluated against plant specific configurations since all test results are not necessarily applicable to all reactor plants. The safety valve test data as reported to date includes only results of steam testing. The subcooled liquid and transition flow tests have not yet been performed.

August 17, 1981

SECY-81-491

For: The Commissioners

From: William J. Dircks
Executive Director for Operations

Subject: REVISED SCHEDULE FOR COMPLETION OF TMI ACTION PLAN ITEM II.D.1, RELIEF AND SAFETY VALVE TESTING

Purpose: To revise NUREG-0737 to extend the schedule for submittal of the subject PWR valve test program results from October 1, 1981 until July 1, 1982

Discussion: By letter dated December 17, 1979, Mr. William J. Cahill, Jr., then Chairman of the EPRI Safety and Analysis Task Force, submitted to the NRC "Program Plan for the Performance Verification of PWR Safety/Relief Valves and Systems". This proposed test program was in response to the requirements specified in NUREG 0578, "TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations", Item 2.1.2, "Performance Testing for BWR and PWR Relief and Safety Valves". Revision 1 of the program plan for PWR safety and relief valve tests was submitted by the industry to NRC on July 8, 1980, in response to NUREG 0737. In addition, there have been several meetings during this time between the PWR utility representatives, EPRI staff and their consultants and NRC staff, to provide additional clarification of the EPRI/PWR safety and relief valve test program. The staff reviewed both the initial and revised test descriptions and was in agreement that the technical requirements of NUREG 0578 and NUREG 0737 would be met on satisfactory completion of testing. However, the proposed test schedule was felt by the staff to be optimistic in that it provided no margin for contingencies.

By letter dated July 1, 1981, from R. C. Youngdahl to Harold R. Denton, enclosure 1, the PWR Owners Group reported on the status of the EPRI PWR safety

Contact:
E. Hemminger, DE, NRR
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and relief valve test program to date and requested an extension of the completion dates specified in NUREG-0737. The Owners Group stated their intention to develop an expanded test matrix in order to obtain more information with respect to the effects of inlet piping configurations and adjustments of ring settings on safety valve operation.

On July 17, 1981, the staff met with EPRI and the PWR Owners Group representatives to review the status of the safety and relief valve testing and to discuss the expanded test matrix. Although the exact number of additional tests will have to be determined as the program progresses, the test program managers estimated that it could take from four to eight months longer than the original test completion date of July 1, 1981, to complete the expanded test program.

Test Program and Status

The program plan developed by EPRI is an extensive testing and analysis effort costing in excess of $17 million. Three test facilities were designated for testing of ten relief valves and nine safety valves. The facilities are located at Marshall Steam Station (Duke Power Company), Wyle Laboratories (Norco, California), and Combustion Engineering (Windsor, Connecticut).

The test facilities at Marshall Steam Station and Wyle Laboratories have been in full operation since mid-1980 and have provided a substantial quantity of information on relief valve (PORV) performance. The PORV test results are summarized in section 4.0 of the "EPRI/PWR Safety and Relief Valve Test Program Interim Data Report", dated July 1, 1981, (enclosure 2). High pressure steam testing is reported as complete on all ten PORVs, and high pressure water, loop seal simulation, and transition steam to water tests are reported as complete on four of the ten PORVs.

The test results for each specific valve are forwarded to utilities that are known to have these valves installed or intended for use in their facilities for purposes of performing any required safety evaluation. In addition, NRR, with assistance from RES and our contractor, EG&G, has been evaluating the PORV test results on a weekly basis. The reported test results indicate that, while the initial screening criteria were not met in some cases, all PORVs tested will function in the primary mode (pressure relief) as required. Additional PORV tests are being planned to evaluate the effect of variable water seal temperature on valve closure times. The test results to date indicate that some valves experience a delay of as much as 70 seconds in closing time due to low or ambient water seal temperatures. However, the valves closed on their own and on disassembly and inspection no damage was observed which might affect their ability to open or close on demand. These results do not indicate a significant safety concern in the staff's view.

The testing of safety valves to meet the NRC requirements has necessitated the design and construction of a new facility at Combustion Engineering. This facility is the first of a kind with the capability to perform meaningful operability tests for large spring-loaded safety valves over a broad range of fluid inlet conditions. Although extraordinary effort, including three shift-work schedules, was devoted to this part of the testing program, delays in construction and shakedown testing resulted in significant delay in the safety valve test schedule. As a result, test results for only two of the nine safety valves to be tested are available (enclosure 2). These test results indicated a need for additional information regarding the effects of inlet piping configuration and adjusting ring settings on safety valve operation. Reporting of safety valve test results and review by affected utilities and the staff is on the same basis as for the PORV results.

Based on our review of the EPRI test program to date, we have concluded that the program represents a fully responsive effort to meet Commission requirements and that the additional testing proposed will provide needed information to assure that the technical requirements of item II.D.1 of NUREG-0737 will be met. Since testing to date has not uncovered problems with safety and relief valves which are considered significant to the safety of operating plants, we believe that good cause has been shown to extend the NUREG-0737 completion date for PORV and safety valve testing so that the extended EPRI program may be carried to completion on an orderly basis. The latest estimated test completion date is March 31, 1982.

A proposed general letter (enclosure 3) will advise all licensees, applicants, and construction permit holders of the revised schedule.

Recommendation: That the Commission approve a revised schedule for completion of the PWR (EPRI) valve test program. It should be noted that:

a. The BWR valve test program is not affected by the recommended change.

b. The change does not impose any additional reporting requirements.

Scheduling: For early consideration.

E Kevin Cornell / for
William J. Dircks
Executive Director for Operations

Enclosures:
1. Ltr. from R. Youngdahl to H. Denton dated July 1, 1981.
2. "EPRI/PWR Safety and Relief Valve Test Program Interim Data Report"
3. Proposed letter to all licensees

Commissioners' comments should be provided directly to the Office of the Secretary by c.o.b. Tuesday, September 1, 1981.

Commission Staff Office comments, if any, should be submitted to the Commissioners NLT August 25, 1981, with an information copy to the Office of the Secretary. If the paper is of such a nature that it requires additional time for analytical review and comment, the Commissioners and the Secretariat should be apprised of when comments may be expected.

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ENCLOSURE [illegible]

Enclosure 1

Consumers Power Company

COPY

General Offices: 212 West Michigan Avenue, Jackson, MI 49201 • (517) 788-0550

July 1, 1981

Mr Harold R Denton
Director, Nuclear Reactor Regulation
U S Nuclear Regulatory Commission
Washington, DC 20555

STATUS OF EPRI PWR SAFETY AND RELIEF VALVE TEST PROGRAM
NUREG-0737, ITEM II.D.1

In December, 1979 forty-one utilities* with planned or operating pressurized water reactors committed to be responsive to the recommendations of NUREG-0578, Section 2.1.2 and demonstrate the capability of safety and relief valves to operate satisfactorily under expected operating and accident conditions. By letter dated July 8, 1980 Revision 1 of the EPRI "Program Plan for the Performance Testing of PWR Safety Relief Valves" was submitted to the NRC. This revision addresses Item II.D.1.A of NUREG-0737, which provided NRC clarifications to the earlier NUREG recommendations.

The program plan developed by EPRI for the participating PWR utilities is an extensive testing and analysis effort which is utilizing three test facilities and will cost in excess of $20 million. The program has been "success" oriented with very little contingency time or funds to resolve potential problems. Although the program has been very successful and preliminary results-to-date indicate that the valves tested will perform their intended safety function, more information appears needed in selected areas. Additional tests, outside the July, 1980 Plan test matrix, are being performed. These additional tests of both safety and relief valves have been informally discussed with the NRC staff. The principal area requiring more testing and evaluation of relief valves is the impact of variable loop seal temperature on the valve operation. Revisions to the safety valve test matrix are necessary to obtain a better understanding of upstream pipe/valve interaction. The impact on the overall test schedule is provided in Attachment 1.

By previous agreement (R C Youngdahl letter to D G Eisenhut, dated December 15, 1980) the PWR utilities agreed to submit the attached Interim Data Report. This report provides all preliminary data collected through June 19, 1981. Additional quick look data reports and weekly activities reports will continue to be provided to the NRC staff until all testing is completed. The PWR utilities still intend to meet the commitment dates provided in the December 15, 1980 letter except that the <u>final</u> data report will not be provided by October 1, 1981.

*Six external organizations have since agreed to participate in the EPRI program (Combustion Engineering, Framatome, Central Nuclear de Almaraz, Furnas Electricas, Electronucleair and Swedish State Power Board).

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Separate from the safety and relief valve test program NUREG-0737, Item II.D.1.B requested that utilities provide verification of block valve functionability. During earlier meetings with the NRC staff, the utilities participating in the EPRI valve program concluded that emphasis must be placed on the demonstration of safety and relief valve operability but that EPRI would be requested to develop a block valve task action plan. The PWR utilities have reviewed a proposed action plan and are now prepared to discuss the need, depth and schedule of a possible block valve program.

While it is recognized that the schedules to satisfy the recommendations of NUREG-0737, Item II.D.1 are not totally consistent with the NRC's request, EPRI and the PWR utilities have instituted a program that is providing new scientific supportable data about valve operability which is not available from any other source.

The utility advisory groups coordinating the test program and EPRI are prepared to meet with the NRC staff to discuss the status of EPRI program in more detail. I propose to meet with you and your staff on July 16 or 17, 1981.

R C Youngdahl

R C Youngdahl
Chairman, EPRI
Research Advisory Committee

ENCLOSURE 2

Enclosure 2

COPY NUMBER 13

EPRI/PWR SAFETY AND RELIEF VALVE TEST PROGRAM

INTERIM DATA REPORT

JULY [illegible], 1981

3.0 SUMMARY OF SAFETY VALVE OPERABILITY DATA

A total of nine PWR pressurizer safety valve designs were tested under steam, water, steam to water (transition), and loop seal conditions.

The nine safety valves selected for testing in the EPRI Program, and the safety valves represented by the valves tested, are identified in Section 2.0 of this report.

The purpose of this section is to present the conditions tested and principal observations for the safety valves tested as of June 19, 1981. Appendix A of this report contains detailed data sheets for these tests. These data sheets are completed after each test and are designed to be self sufficient to allow timely dissemination of that safety valve test data deemed necessary to adequately evaluate valve performance. Key information included on these sheets are valve designation, tested conditions, valve opening and closing times, maximum stem position and valve flow rates.

3.1 DRESSER SAFETY VALVE MODEL 31709NA

3.1.1 Conditions Tested

Tests were performed on the Dresser safety valve model 31709NA at the EPRI/CE PWR Safety and Relief Valve Test Facility. Table 3.1.1 presents the matrix of conditions under which the Dresser valve was tested.

3.1.2 Summary of Principal Observations

A full pressure steam test (test No. 201) was performed on the Dresser safety valve, model 31709NA. The test was performed with the valve mounted on a loop seal configuration with the loop seal drained and the valve set point established at 2480 psig. The test was initiated with a high ramp rate transient from the pre-test pressure of 2315 psia. The safety valve opened at a valve inlet pressure of 2465 psia. The transient continued for a total time of 122 seconds. The valve chattered during most of the test duration. The valve reclosed at a pressure of 2000 psia. Several minutes after closure, the valve re-opened for a second time. The second opening pressure noted by the loop operator was approximately 2150 psia. The valve reclosed the second time at a slightly reduced pressure. The valve was open for about 10 seconds and chattered during this time.

After the test, a leak test was performed at an inlet pressure of about 2100 psia. The valve leakage measured was about 0.5 gpm. The valve was then disassembled and a preliminary inspection was performed. Galling of guiding surfaces was found; several internal parts were damaged.

Detailed data sheets are contained in Appendix A, Section A-1.

TABLE 3.1.1

"AS TESTED" COMBUSTION ENGINEERING TEST MATRIX FOR THE DRESSER SAFETY VALVE 31709NA

TEST. NO.	TEST TYPE	INLET PIPING CONFIGURATION	INITIAL CONDITIONS			TRANSIENT CONDITIONS				
			FLUID	TEMP °F	PRESS PSIA	PRESS RATE PSI/SEC	VALVE OPENING TANK PRESS PSIA	PEAK TANK PRESS PSIA	PEAK DOWN-STREAM PRESS PSIA	VALVE CLOSING TANK PRESS PSIA
201	Steam	Loop Seal (Drained)	Steam	Sat.	2315	340-425	2488	2680	(1)	2010

(1) Measurement was oscillatory

3.2 DRESSER SAFETY VALVE MODEL 31739A

3.2.1 Conditions Tested

Tests were performed on the Dresser safety valve model 31739A at the EPRI/CE PWR Safety and Relief Valve Test Facility. Table 3.2.1 presents the matrix of conditions under which the Dresser valve was tested.

3.2.2 Summary of Principal Observations

A full pressure, low ramp rate, low backpressure, steam test (test No. 302) was performed on the Dresser safety valve (31739A). The valve opened at a pressure within $\pm 3\%$ of the valve set point. A maximum stem position of 58% of rated lift was obtained at a pressure less than 6% above the valve set pressure. The valve reclosed at a pressure greater than 2250 psig.

Detailed data sheets are contained in Appendix A, Section A-2.

3.3 CROSBY HB-BP-86, 3K6 - Loop Seal Application

3.3.1 Conditions Tested

- later -

3.3.2 Summary of Principal Observations

- later -

3.4 CROSBY HB-BP-86, 6M6 - Loop Seal Application

3.4.1 Conditions Tested

- later -

3.4.2 Summary of Principal Observations

- later -

TABLE 3.2.1

"AS TESTED" COMBUSTION ENGINEERING TEST MATRIX FOR THE DRESSER SAFETY VALVE 31739A

TEST. NO.	TEST TYPE	INLET PIPING CONFIGURATION	INITIAL CONDITIONS FLUID	INITIAL CONDITIONS TEMP °F	INITIAL CONDITIONS PRESS PSIA	TRANSIENT CONDITIONS PRESS RATE PSI/SEC	TRANSIENT CONDITIONS VALVE OPENING TANK PRESS PSIA	TRANSIENT CONDITIONS PEAK TANK PRESS PSIA	TRANSIENT CONDITIONS PEAK DOWN-STREAM PRESS PSIA	TRANSIENT CONDITIONS VALVE CLOSING TANK PRESS PSIA
302	Steam	Straight	Steam	Sat.	2300	3.75	2483	2483	165	2336

3.5 CROSBY HB-BP-86, 6N8 - Loop Seal Application

3.5.1 Conditions Tested

- later -

3.5.2 Summary of Principal Observations

- later -

3.6 CROSBY HB-BP-86, 6N8 - Non-Loop Seal Application

3.6.1 Conditions Tested

- later -

3.6.2 Summary of Principal Observations

- later -

3.7 CROSBY HB-BP-86, 6M6 - Non-Loop Seal Application

3.7.1 Conditions Tested

- later -

3.7.2 Summary of Principal Observations

- later -

3.8 CROSBY HB-BP-86, 3K6 - Non-Loop Seal Application

3.8.1 Conditions Tested

- later -

3.8.2 Summary of Principal Observations

- later -

3.9 TARGET ROCK 69C

3.9.1 Conditions Tested

- later -

3.9.2 Summary of Principal Observations

- later -

End 2

4.0 SUMMARY OF RELIEF VALVE OPERABILITY DATA

The EPRI Program calls for the testing of ten PWR pressurizer relief valves under steam, water, steam to water (transition) and water seal simulation conditions.

The ten relief valves selected for testing in the EPRI Program, and the relief valves represented by the valves tested are identified in Section 2.0 of this report.

The purpose of this section is to present the test matrices and principal observations of the relief valves tested as of June 19, 1981. Appendix B of this report contains detailed data sheets for these tests. These data sheets are completed after each test and are designed to be self-sufficient to allow timely dissemination of that relief valve test data deemed necessary to adequately evaluate valve performance. Key information included on these sheets are valve designation, tested conditions, valve opening and closing times and valve flow rates.

4.1 DRESSER RELIEF VALVE

4.1.1 Conditions Tested

Tests were performed on the Dresser relief valve model at the Marshall Steam Station and during Phase II and Phase III of the Wyle Test Program. Tables 4.1.1a, b, and c present the matrix of conditions under which this valve model was tested at Marshall, Wyle (Phase II), and Wyle (Phase III), respectively.

4.1.2 Summary of Principal Observations

- Marshall Steam Station

  The valve fully opened on demand and fully closed on demand for each of the ten (10) evaluation test cycles. During the evaluation tests, steam leaked past the valve pilot stem. Upon valve disassembly, the bellows was found to have several partially failed welds. The valve was reassembled with a new bellows and cycled 16 more times with varying pilot back-pressures up to 900 psig. The valve fully opened and closed on demand for each cycle and the bellows did not leak. Upon disassembly, the bellows did not have any visible cracks. In all test cases, the valve fully opened on demand and closed on demand even though the bellows was damaged during some tests. Based on this input and the manufacturer's assessment of valve performance with the observed damage, the damage was determined to have no potential impact on valve operation.

  Detailed data sheets for the evaluation tests are contained in Appendix B, Section B-1a.

- Wyle Phase II

  The valve fully opened on demand and fully closed on demand for each of the five (5) test cycles.

  Detailed data sheets are contained in Appendix B, Section B-1b.

- Wyle Phase III

  The valve fully opened on demand and fully closed on demand for nine (9) test cycles. The valve fully opened on demand and did not close on demand during the three (3) water seal simulation tests; numbers 16-DR-6W, 22-DR-9W/W and 24-DR-6W. Each test was a 2500 psia pressure test with low temperature water just upstream of the valve followed by 650°F water.

  In test number 16-DR-6W, the low temperature water was at 103°F. During the test, the Dresser valve opened on demand. Upon de-energizing the valve for closure, the valve remained open until the valve was isolated from the test loop. Following test valve isolation, the valve closed. The valve was isolated approximately 40 seconds after it was signalled to close. The valve was removed from the test facility and disassembled by the Dresser representative. No damage was observed which might affect the ability of the valve to open/close on demand.

  In test number 22-DR-9W/W, the low temperature water was 321°F. During the test, the valve opened on demand. Upon de-energizing the valve for closure, the valve remained open for 2 seconds and then closed fully.

  Test number 24-DR-6W was a repeat of the test 16-DR-6W except that the test was run to maximize the time before the valve was isolated. The water temperature immediately upstream of the valve was 105°F. During the test, the valve opened on demand but failed to close immediately upon de-energizing the solenoid. The valve closed on its own approximately 70 seconds after the closure signal at an inlet pressure of approximately 2110 psia.

  After all tests were completed, the Dresser valve was removed, disassembled, and inspected. No damage was observed which might affect the ability of the valve to open/close on demand.

  Detailed data sheets are contained in Appendix B, Section B-1c.

- Opening Time

  The total valve opening time data for the Marshall Steam Station tests and the Wyle Phase II & III tests were obtained based on different types of inputs. As a result, the recorded Marshall opening times exceed the recorded Wyle times for similar steam test conditions. In addition, main disc opening times of the valve could not be accurately determined at Wyle. For that reason, the main disc opening time was not included on the Wyle Phase II & III data sheets.

TABLE 4.1.1a

"AS TESTED" MARSHALL TEST MATRIX FOR THE DRESSER RELIEF VALVE

TEST NO.	TEST TYPE	"NOMINAL" INITIAL CONDITIONS AT VALVE INLET			"NOMINAL" TRANSIENT CONDITIONS		
		FLUID	TEMP °F	PRESS PSIA	TEST DURATION (SEC)	VALVE CLOSURE PRESS (PSIA)	MAX DISCH. PIPE B.P. PSIA
1*	Steam	Steam	(Sat.)	2475	60	2315	415
2 - 5	Steam	Steam	(Sat.)	2475	15	2335	415
6*	Steam	Steam	(Sat.)	2455	60	2325	175
7 - 10	Steam	Steam	(Sat.)	2455	15	2320	175

*Tests 1 and 6 were extended duration flow measurement tests.

TABLE 4.1.1b

"AS TESTED" WYLE PHASE II TEST MATRIX FOR THE DRESSER RELIEF VALVE

TEST NO.	TEST TYPE	INITIAL CONDITIONS AT VALVE INLET			TRANSIENT CONDITIONS			
		FLUID	TEMP ^{o}F	PRESS PSIA	TEST DURATION (SEC)	VALVE CLOSURE PRESS. (PSIA)	MAX DISCH. PIPE B.P. PSIA	MAX PILOT LINE BP PSIA
DR-1-S	STEAM	STEAM	674	2490	~6	2215	~60	1040
DR-3-W	WATER	WATER	373	680	~6	510	155	213
DR-5-W	WATER	WATER	646	2500	~15	2300	380	680
DR-6-W	WATER	WATER	506	2500	~26	2120	330	380
DR-7W	WATER	WATER	447	2510	~16	2120	373	333

TABLE 4.1.1c

"AS TESTED" WYLE PHASE III TEST MATRIX FOR THE DRESSER RELIEF VALVE

TEST NO.	TEST TYPE	INITIAL CONDITIONS: AT VALVE INLET FLUID	TEMP °F	PRESS. PSIA	IN ACCUMULATOR FLUID	TEMP. °F	PRESS. PSIA	TRANSIENT CONDITIONS: TEST DURATION SEC.	VALVE CLOSURE PRESS PSIA	MAX DISCH. PIPE BP PSIA	MAX PILOT LINE BP PSIA	MAX (STATIC+DYNAMIC) BENDING MOMENT INDUCED IN-LB
10-DR-1S	STEAM	STEAM	668	2503	SAME AS VALVE INLET			~15	3035	755	830	N/A
11-DR-4W	WATER	WATER	647	2514				~15	2338	620	740	N/A
12- R-3W	WATER	WATER	450	699				~15	685	260	300	N/A
13-DR-7W	WATER	WATER	451	2492				~10	652	420	450	N/A
14-DR-2W	WATER	WATER	112	689				~10	2230	~2	~2	N/A
15-DR-5W	WATER (preload)	WATER	643	2504				~10	2360	640	750	35,600
16-DR-6W	WATER SEAL SIMULATION	WATER	103	2500	WATER	652	2500	~54	~14.7	292	513	~90,000
20-DR-1S	STEAM	STEAM	657	2505	SAME AS VALVE INLET			~10	2110	494	760	N/A
21-DR-8S/W	TRANSITION	STEAM	656	2496	WATER	641	2503	~10	2360	660	770	N/A
22-DR-9W/W	WATER SEAL SIMULATION	WATER	321	2490	WATER	647	2488	~17	2310	675	815	N/A
23-DR-1S	STEAM	STEAM	657	2505	SAME AS VALVE INLET			~11	2110	440	583	N/A
24-DR-6W	WATER SEAL SIMULATION	WATER	105	2505	WATER	650	2505	~85	2110	693	788	N/A

4.2 CROSBY RELIEF VALVE

4.2.1 Conditions Tested

Tests were performed on the Crosby relief valve model at the Marshall Steam Station, and during Phase II and Phase III of the Wyle Test Program. Tables 4.2.1a, b, and c present the matrix of conditions under which this valve model was tested at Marshall, Wyle (Phase II), and Wyle (Phase III), respectively.

4.2.2 Summary of Principal Observations

- Marshall Steam Station

  The valve fully opened on demand and fully closed on demand during each of the ten (10) evaluation tests.

  During valve cycling performed prior to the evaluation tests under full flow steam conditions, the pilot bellows leaked. When the valve was disassembled and inspected, one bellows weld fracture was found and a bellows assembly part was found to be improperly machined. The bellows was replaced, the bellows assembly was correctly machined and the valve was reassembled for further tests.

  The valve was subsequently cycled 44 times including the ten evaluation tests. The valve fully opened and closed on demand and no bellows leakage occurred during the tests.

  Detailed data sheets for the evaluation tests are contained in Appendix B, Section B-2a.

- Wyle Phase II

  The valve fully opened on demand and fully closed on demand for each of the six (6) test cycles. Upon disassembly after tests were completed, the pilot bellows was found to leak.

  Detailed data sheets are contained in Appendix B, Section B-2b.

- Wyle Phase III

  The valve fully opened on demand and fully closed on demand for each of the ten (10) test cycles. Upon disassembly after tests were completed, the pilot bellows was observed to be damaged.

  - Bellows Damage

    In all test cases, the valve fully opened on demand and closed on demand even though the bellows had been damaged. Based on this input and the manufacturer's assessment of valve performance with the observed damage, the damage was determined to have no potential impact on valve operation.

Opening Time

The total valve opening time data for the Marshall Steam Station tests and the Wyle Phase II & III tests were obtained based on different types of inputs. As a result, the recorded Marshall opening times exceed the recorded Wyle times for similar steam test conditions. In addition, main disc opening times of the valve could not be accurately determined at Wyle. For that reason, the main disc opening time was not included on the Wyle Phase II & III data sheets.

TABLE 4.2.1a

"AS TESTED" MARSHALL TEST MATRIX FOR THE CROSBY RELIEF VALVE

TEST NO.	TEST TYPE	"NOMINAL" INITIAL CONDITIONS AT VALVE INLET			"NOMINAL" TRANSIENT CONDITIONS		
		FLUID	TEMP °F	PRESS PSIA	TEST DURATION (SEC)	VALVE CLOSURE PRESS. (PSIA)	MAX DISCH. PIPE B.P. PSIA
1*	Steam	Steam	(Sat.)	2495	60	2350	385
2 - 5	Steam	Steam	(Sat.)	2495	15	2340	380
6*	Steam	Steam	(Sat.)	2495	60	2355	135
7 - 10	Steam	Steam	(Sat.)	2495	15	2335	120

* Tests 1 and 6 were extended duration flow measurement tests

TABLE 4.2.1b

"AS TESTED" WYLE PHASE II TEST MATRIX FOR THE CROSBY RELIEF VALVE

TEST NO.	TEST TYPE	INITIAL CONDITIONS AT VALVE INLET			TRANSIENT CONDITIONS			
		FLUID	TEMP °F	PRESS PSIA	TEST DURATION SEC	VALVE CLOSURE PSIA	MAX DISCHARGE PIPE B.P. PSIA	MAX PILOT LINE B.P. PSIA
CR-1-S	STEAM	STEAM	672	2510	~15	1920	142	945
CR-2-S	STEAM	STEAM	671	2495	~7	2140	560	(1) >1000
CR-3-W	WATER	WATER	376	680	~15	618	244	200
CR-5-W	WATER	WATER	634	2510	~15	2280	397	775
CR-6-W	WATER	WATER	505	2502	~18	2100	460	438
CR-7-W	WATER	WATER	446	2510	~19	2000	550	661

(1) The 1000 psia pressure sensor was over-ranged on this test.

TABLE 4.2.1c

"AS TESTED" WYLE PHASE III TEST MATRIX FOR THE CROSBY RELIEF VALVE

TEST NO.	TEST TYPE	INITIAL CONDITIONS: AT VALVE INLET FLUID	INITIAL CONDITIONS: AT VALVE INLET TEMP. °F	INITIAL CONDITIONS: AT VALVE INLET PRESS. PSIA	INITIAL CONDITIONS: IN ACCUMULATOR FLUID	INITIAL CONDITIONS: IN ACCUMULATOR TEMP °F	INITIAL CONDITIONS: IN ACCUMULATOR PRESS. PSIA	TEST DURATION (SEC)	TRANSIENT CONDITIONS: VALVE CLOSURE PRESS PSIA	TRANSIENT CONDITIONS: MAX DISCHARGE PIPE PRESS. PSIA	TRANSIENT CONDITIONS: MAX PILOT LINE BP PSIA	TRANSIENT CONDITIONS: MAXIMUM (STATIC+DYNAMI BENDING MOMENT INDUCED IN-LB
25-CR-1S	STEAM	STEAM	656	2505	SAME AS VALVE INLET			10	2050	NOT RECORDED	865	N/A
26-CR-6S	STEAM (PRELOAD)	STEAM	657	2505				10	2037	NOT RECORDED	868	38,400
27-CR-2W	WATER	WATER	104	694				10	620	1.0	518	N/A
28-CR-3W	WATER	WATER	437	695				10	655	160	540	N/A
29-CR-1S	STEAM	STEAM	656	2505				10	2050	740	865	N/A
30-CR-1S	STEAM	STEAM	656	2505				10	2060	370	780	N/A
31-CR-4S/W	TRANSITION	STEAM	656	2510	WATER	64[illegible]	2510	15	2313	NOT RECORDED	770	N/A
32-CR-5W/W	WATER SEAL SIMULATION	WATER	469	2505	WATER	646	2505	15	2290	560	740	N/A
33-CR-7W/W	WATER SEAL SIMULATION	WATER	294	2505	WATER	648	2505	15	2300	580	840	N/A
34-CR-8W/W	WATER SEAL SIMULATION	WATER	118	2500	WATER	645	2500	15	2290	570	700	N/A

4.3 TARGET ROCK RELIEF VALVE

4.3.1 Conditions Tested

Tests were performed on the Target Rock relief valve model at the Marshall Steam Station and during Phase III of the Wyle Test Program. Tables 4.3.1a and b present the matrix of conditions under which this valve model was tested at Marshall and Wyle (Phase III), respectively.

4.3.2 Summary of Principal Observations

- Marshall Steam Station

The valve fully opened on demand and fully closed on demand for each of the ten (10) evaluation test cycles.

Detailed data sheets are contained in Appendix B, Section B-3a.

- Wyle Phase III

The valve fully opened on demand and fully closed on demand in eleven (11) of the twelve (12) test cycles. The valve did not close on demand when the full pressure 2500 psi, water seal simulation test (test number 7-TR-7W) was performed. The water just upstream of the valve was 110°F water. For this test, the valve opened on demand. Upon de-energizing the valve for closure, the valve remained opened for approximately 12 seconds and then closed. The valve was removed from the test facility and disassembled by the Target Rock representative. No damage was observed which might affect the ability of the valve to open/close on demand.

Detailed data sheets are contained in Appendix B, Section B-3b.

- Opening Time

The total valve opening time data for the Marshall Steam Station tests and the Wyle Phase II & III tests were obtained based on different types of inputs. As a result, the recorded Marshall opening times exceed the recorded Wyle times for similar steam test conditions. In addition, main disc opening times of the valve could not be accurately determined at Wyle. For that reason, the main disc opening time was not included on the Wyle Phase II & III data sheets.

TABLE 4.3.1a

"AS TESTED" MARSHALL TEST MATRIX FOR THE TARGET ROCK RELIEF VALVE

TEST NO.	TEST TYPE	"NOMINAL" INITIAL CONDITIONS AT VALVE INLET			"NOMINAL" TRANSIENT CONDITIONS		
		FLUID	TEMP °F	PRESS PSIA	TEST DURATION (SEC)	VALVE CLOSURE PRESS. (PSIA)	MAX DISCH. PIPE B.P. PSIA
1*	Steam	Steam	(Sat.)	2435	60	2335	475
2 - 5	Steam	Steam	(Sat.)	2435	15	2300	475
6*	Steam	Steam	(Sat.)	2445	60	2315	165
7 - 10	Steam	Steam	(Sat.)	2445	15	2320	165

* Tests 1 and 6 were extended duration flow measurement tests

TABLE 4.3.1b

"AS TESTED" WYLE PHASE III TEST MATRIX FOR THE TARGET ROCK RELIEF VALVE

TEST NO.	TEST TYPE	INITIAL CONDITIONS							TRANSIENT CONDITIONS		
		AT VALVE INLET			IN ACCUMULATOR			TEST DURATION	VALVE CLOSURE	MAX DISCHARGE PIPE	MAXIMUM (STATIC+DYNAMIC) BENDING MOMENT INDUCED
		FLUID	TEMP. ^{o}F	PRESS. PSIA	FLUID	TEMP ^{o}F	PRESS. PSIA	(SEC)	PRESS. PSIA	PRESS. PSIA	IN-LB
1-TR-1S	STEAM	STEAM	660	2521	SAME AS VALVE INLET			~7	2132	320	N/A
2-TR-1S	STEAM	STEAM	669	2504				~7	2134	330	N/A
3-TR-3W	WATER	WATER	447	715				~70	639		N/A
4-TR-5W	WATER	WATER	645	2515				~15	2293	450	N/A
5-TR-2W	WATER	WATER	114	690				~10	616	~1	N/A
6-TR-4W	WATER	WATER	448	2545				~10	2196	395	N/A
7-TR-7W	WATER SEAL SIMULATION	WATER	113	2505	WATER	656	2506	~27	2172	520	N/A
8-TR-5W	WATER	WATER	648	2494	SAME AS VALVE INLET			~10	2320	430	N/A
9-TR-6W	WATER (PRELOAD)	WATER	645	2490				~10	2302	425	16,400
17-TR-1S	STEAM	STEAM	657	2510				~10	2028	325	N/A
18-TR-6S	STEAM (PRELOAD)	STEAM	658	2505				~10	2020	315	36,600
19-TR-9S/W	TRANSITION	STEAM	656	2500	WATER	642	2504	~10	2310	435	N/A

4.4 CONTROL COMPONENTS RELIEF VALVE

4.4.1 Conditions Tested

Tests were performed on the Control Components relief valve model at the Marshall Steam Station and during Phase III of the Wyle Test Program. Tables 4.4.1a and b present the matrix of conditions under which this valve model was tested at Marshall and Wyle (Phase III), respectively.

4.4.2 Summary of Principal Observations

- Marshall Steam Station

  The valve fully opened on demand and fully closed on demand for each of the tan (10) evaluation test cycles.

  Detailed data sheets are contained in Appendix B, Section B-4a.

- Wyle Phase III

  The valve fully opened on demand and fully closed on demand for each of the four (4) test cycles performed through June 19, 1981.

  Detailed data sheets are contained in Appendix B, Section B-4b.

TABLE 4.4.1a

"AS TESTED" MARSHALL TEST MATRIX FOR THE CONTROL COMPONENTS RELIEF VALVE

TEST NO.	TEST TYPE	"NOMINAL" INITIAL CONDITIONS AT VALVE INLET			"NOMINAL" TRANSIENT CONDITIONS		
		FLUID	TEMP °F	PRESS PSIA	TEST DURATION (SEC)	VALVE CLOSURE PRESS. (PSIA)	MAX DISCH. PIPE B.P. PSIA
1*	Steam	Steam	(Sat.)	2435	60	2175	615
2 - 5	Steam	Steam	(Sat.)	2435	15	2185	615
6*	Steam	Steam	(Sat.)	2435	60	2185	215
7 - 10	Steam	Steam	(Sat.)	2435	15	2185	215

* Tests 1 and 6 were extended duration flow measurement tests

TABLE 4.4.1b

"AS TESTED" WYLE PHASE III TEST MATRIX FOR THE CONTROL COMPONENTS RELIEF VALVE

TEST NO.	TEST TYPE	INITIAL CONDITIONS							TRANSIENT CONDITIONS		
		AT VALVE INLET			IN ACCUMULATOR						
		FLUID	TEMP. °F	PRESS. PSIA	FLUID	TEMP °F	PRESS. PSIA	TEST DURATION (SEC)	VALVE CLOSURE PRESS. PSIA	MAX DISCHARGE PIPE PRESS. PSIA	MAXIMUM (STATIC+DYNAMIC) BENDING MOMENT INDUCED IN-LB
35-CC-1S	Steam	Steam	683	2760	Same as Valve Inlet			5	2330	468	N/A
36-CC-2S	Steam (Failed Air)	Steam	683	2750				5	2280	416	N/A
37-CC-3S	Steam (Preload Failed Air)	Steam	670	2535				~2	2225	377	N/A
38-CC-5W	Water (Failed Air)	Water	440	2538				5	2180	400	N/A

- Balance of CCI Tests Completed After 6/19/81 -

Preliminary

4.5 MASONEILAN RELIEF VALVE

4.5.1 Conditions Tested

Tests were performed on the Masoneilan relief valve model at the Marshall Steam Station and during Phase III of the Wyle Test Program. Tables 4.5.1a and b present the matrix of conditions under which this valve model was tested at Marshall and Wyle (Phase III), respectively.

4.5.2 Summary of Principal Observations

- Marshall Steam Station

  The valve fully opened on demand and fully closed on demand for each of the ten (10) evaluation test cycles.

  Detailed data sheets are contained in Appendix B, Section B-5a.

- Wyle Phase III

  - later -

TABLE 4.5.1a

"AS TESTED" MARSHALL TEST MATRIX FOR THE MASONEILAN RELIEF VALVE

TEST NO.	TEST TYPE	"NOMINAL" INITIAL CONDITIONS AT VALVE INLET			"NOMINAL" TRANSIENT CONDITIONS		
		FLUID	TEMP °F	PRESS PSIA	TEST DURATION (SEC)	VALVE CLOSURE PRESS. (PSIA)	MAX DISCH. PIPE B.P. PSIA
1*	Steam	Steam	(Sat.)	2500	60	2235	535
2 - 5	Steam	Steam	(Sat.)	2500	15	2235	535
6*	Steam	Steam	(Sat.)	2500	60	2215	180
7 - 10	Steam	Steam	(Sat.)	2500	15	2230	180

* Tests 1 and 6 were extended duration flow measurement tests

TABLE 4.5.1b

"AS TESTED" WYLE PHASE III TEST MATRIX FOR THE MASONEILAN RELIEF VALVE

TEST NO.	TEST TYPE	INITIAL CONDITIONS							TRANSIENT CONDITIONS		
		AT VALVE INLET			IN ACCUMULATOR			TEST DURATION (SEC)	VALVE CLOSURE PRESS. PSIA	MAX DISCHARGE PIPE PRESS. PSIA	MAXIMUM (STATIC+DYNAMIC) BENDING MOMENT INDUCED IN-LB
		FLUID	TEMP. ^{o}F	PRESS. PSIA	FLUID	TEMP ^{o}F	PRESS. PSIA				

- LATER -

4.6 COPES-VULCAN RELIEF VALVE (316 w/stellite Plug and 17-4PH Cage)

4.6.1 Conditions Tested

Tests were performed on the Copes-Vulcan relief valve model (316 w/stellite Plug and 17-4PH Cage) at the Marshall Steam Station and during Phase III of the Wyle Test Program. Tables 4.6.1a and b present the matrix of conditions under which this valve model was tested at Marshall and Wyle (Phase III), respectively.

4.6.2 Summary of Principal Observations

- Marshall Steam Station

  The valve fully opened on demand and fully closed on demand for each of the ten (10) evaluation test cycles.

  Detailed data sheets are contained in Appendix B, Section B-6a.

- Wyle Phase III

  - later

TABLE 4.6.1a

"AS TESTED" MARSHALL TEST MATRIX FOR THE COPES VULCAN RELIEF VALVE
(316 w/stellite Plug and 17-4PH Cage)

TEST NO.	TEST TYPE	"NOMINAL" INITIAL CONDITIONS AT VALVE INLET			"NOMINAL" TRANSIENT CONDITIONS		
		FLUID	TEMP °F	PRESS PSIA	TEST DURATION (SEC)	VALVE CLOSURE PRESS. (PSIA)	MAX DISCH. PIPE B.P. PSIA
1*	Steam	Steam	(Sat.)	2435	60	2155	635
2 - 5	Steam	Steam	(Sat.)	2435	15	2165	635
6*	Steam	Steam	(Sat.)	2455	60	2145	205
7 - 10	Steam	Steam	(Sat.)	2455	15	2165	215

* Tests 1 and 6 were extended duration flow measurement tests

TABLE 4.6.1b

"AS TESTED" WYLE PHASE III TEST MATRIX FOR THE COPES-VULCAN RELIEF VALVE
(316 w/Stellite Plug and 17-4PH Cage)

TEST NO.	TEST TYPE	INITIAL CONDITIONS							TRANSIENT CONDITIONS		
		AT VALVE INLET			IN ACCUMULATOR			TEST DURATION (SEC)	VALVE CLOSURE PRESS. PSIA	MAX DISCHARGE PIPE PRESS. PSIA	MAXIMUM (STATIC+DYNAMIC) BENDING MOMENT INDUCED IN-LB
		FLUID	TEMP. ^{o}F	PRESS. PSIA	FLUID	TEMP ^{o}F	PRESS. PSIA				

- LATER -

4.7 COPES-VULCAN RELIEF VALVE (17-4PH Plug and Cage)

4.7.1 Conditions Tested

Tests were performed on the Copes-Vulcan relief valve model with the 17-4Ph Plug and Cage at the Marshall Steam Station and during Phase III of the Wyle Test Program. Tables 4.7.1a and b present the matrix of conditions under which this valve model was tested at Marshall and Wyle (Phase III), respectively.

4.7.2 Summary of Principal Observations

- Marshall Steam Station

  The valve fully opened on demand and closed on demand for each of the ten (10) evaluation test cycles.

  After these tests were completed, a new set of the same design cage and plug parts were installed and the valve was placed back in the test facility. The valve was cycled to investigate the cage to body gasket performance and to support other Marshall Steam Station test functions. The valve fully opened on demand and fully closed on demand for the first 13 full flow cycles. During the next seven cycles, the valve closed to within at least 88% of the full closed position. The valve did not fully close on demand. Disassembly showed galling of the cage and plug guiding surfaces.

  Detailed data sheets are contained in Appendix B, Section B-7a.

- Wyle Phase III

  \- later -

TABLE 4.7.1a

"AS TESTED" MARSHALL TEST MATRIX FOR THE COPES-VULCAN RELIEF VALVE
(17-4PH Plug and Cage)

TEST NO.	TEST TYPE	"NOMINAL" INITIAL CONDITIONS AT VALVE INLET			"NOMINAL" TRANSIENT CONDITIONS		
		FLUID	TEMP °F	PRESS PSIA	TEST DURATION (SEC)	VALVE CLOSURE PRESS. (PSIA)	MAX DISCH. PIPE B.P. PSIA
1*	Steam	Steam	(Sat.)	2445	60	2155	595
2 - 5	Steam	Steam	(Sat.)	2445	15	2200	610
6*	Steam	Steam	(Sat.)	2465	60	2155	195
7 - 10	Steam	Steam	(Sat.)	2465	15	2190	195

* Tests 1 and 6 were extended duration flow measurement tests

TABLE 4.7.1b

"AS TESTED" WYLE PHASE III TEST MATRIX FOR THE COPES-VULCAN RELIEF VALVE
(17-4PH Plug and Cage)

TEST NO.	TEST TYPE	INITIAL CONDITIONS							TRANSIENT CONDITIONS		
		AT VALVE INLET			IN ACCUMULATOR			TEST DURATION (SEC)	VALVE CLOSURE PRESS. PSIA	MAX DISCHARGE PIPE PRESS. PSIA	MAXIMUM (STATIC+DYNAMIC) BENDING MOMENT INDUCED IN-LB
		FLUID	TEMP. °F	PRESS. PSIA	FLUID	TEMP °F	PRESS. PSIA				

- LATER -

4.8 MUESCO CONTROLS RELIEF VALVE

4.8.1 Conditions Tested

Tests were performed on the MUESCO Controls relief valve model at the Marshall Steam Station and during Phase III of the Wyle Test Program. Tables 4.8.1a and b present the matrix of conditions under which this valve model was tested at Marshall and Wyle (Phase III), respectively.

4.8.2 Summary of Principal Observations

- Marshall Steam Station

  The valve fully opened on demand and fully closed on demand for each of the ten (10) evaluation test cycles.

  Further tests were performed on the valve with a replacement stem, plug and gaskets. These parts exhibited wear during the first set of tests and a second set of tests was recommended by MUESCO Controls for information purposes. The valve fully opened on demand and fully closed on demand for each of the evaluation test cycles. Similar wear patterns were found.

  Detailed data sheets are contained in Appendix B, Section B-8a.

- Wyle Phase III

  \- later -

TABLE 4.8.1a

"AS TESTED" MARSHALL TEST MATRIX FOR THE MUESCO RELIEF VALVE

TEST NO.	TEST TYPE	"NOMINAL" INITIAL CONDITIONS AT VALVE INLET			"NOMINAL" TRANSIENT CONDITIONS		
		FLUID	TEMP °F	PRESS PSIA	TEST DURATION (SEC)	VALVE CLOSURE PRESS. (PSIA)	MAX DISCH. PIPE B.P. PSIA
1*	Steam	Steam	(Sat.)	2435 (2485)**	60	2395 (2395)**	235 (255)**
2 - 5	Steam	Steam	(Sat.)	2435	15	2400 (2375)**	235 (255)**
6*	Steam	Steam	(Sat.)	2455 (2475)**	60	2395 (2415)**	80 (80)**
7 - 10	Steam	Steam	(Sat.)	2455	15	2380 (2470)**	80 (80)**

* Tests 1 and 6 were extended duration flow measurement tests

** (Second set of Tests)

TABLE 4.8.1a

"AS TESTED" MARSHALL TEST MATRIX FOR THE MUESCO RELIEF VALVE

TEST NO.	TEST TYPE	"NOMINAL" INITIAL CONDITIONS AT VALVE INLET			"NOMINAL" TRANSIENT CONDITIONS		
		FLUID	TEMP °F	PRESS PSIA	TEST DURATION (SEC)	VALVE CLOSURE PRESS. (PSIA)	MAX DISCH. PIPE B.P. PSIA
1*	Steam	Steam	(Sat.)	2435 (2485)**	60	2395 (2395)**	235 (255)**
2 - 5	Steam	Steam	(Sat.)	2435	15	2400 (2375)**	235 (255)**
6*	Steam	Steam	(Sat.)	2455 (2475)**	60	2395 (2415)**	80 (80)**
7 - 10	Steam	Steam	(Sat.)	2455	15	2380 (2470)**	80 (80)**

* Tests 1 and 6 were extended duration flow measurement tests

** (Second set of Tests)

TABLE 4.8.1b

"AS TESTED" WYLE PHASE III TEST MATRIX FOR THE MUESCO CONTROLS RELIEF VALVE

TEST NO.	TEST TYPE	INITIAL CONDITIONS							TRANSIENT CONDITIONS		
		AT VALVE INLET			IN ACCUMULATOR						
		FLUID	TEMP. ^{o}F	PRESS. PSIA	FLUID	TEMP ^{o}F	PRESS. PSIA	TEST DURATION (SEC)	VALVE CLOSURE PRESS. PSIA	MAX DISCHARGE PIPE PRESS. PSIA	MAXIMUM (STATIC+DYNAMIC) BENDING MOMENT INDUCED IN-LB

- LATER -

4.9 FISHER CONTROLS RELIEF VALVE

4.9.1 Conditions Tested

Tests were performed on the Fisher Controls relief valve model at the Marshall Steam Station and during Phase III of the Wyle Test Program. Tables 4.9.1a and b present the matrix of conditions under which this valve model was tested at Marshall and Wyle (Phase III), respectively.

4.9.2 Summary of Principal Observations

- Marshall Steam Station

The valve fully opened on demand and fully closed on demand for each of the ten (10) evaluation test cycles. At the conclusion of the test, the valve was disassembled and galling was observed on the plug and cage mating surfaces.

In addition to the evaluation tests, three other sets of cycles were performed on the valve. The first two sets of cycles were performed on a set of cage and plug parts which did not represent the correct Fisher Controls design for the PORV application. During the cycles, the valve closed on demand to within at least 96% of the full closed position on each cycle. After the cycles were completed, the valve was disassembled and galling was observed on the plug and cage mating surfaces. The galling was more severe than the evaluation test cycle galling pattern.

The evaluation test was then performed on a set of cage and plug parts with correct clearances. These are the tests discussed in the first paragraph of this section and they represent Fisher Controls PORVs supplied to PWR plants with the correct internals.

A fourth set of cycles were performed on a set of trim with the correct design clearances. The valve fully opened on demand and fully closed on demand for each cycle. A galling pattern similar to that observed in the evaluation test was observed. Again, it was less severe than the pattern observed when the valve did not fully close on demand.

Detailed data sheets are contained in Appendix B, Section B-9a.

- Wyle Phase III

- later -

TABLE 4.9.1a

"AS TESTED" MARSHALL TEST MATRIX FOR THE FISHER CONTROLS RELIEF VALVE

TEST NO.	TEST TYPE	"NOMINAL" INITIAL CONDITIONS AT VALVE INLET			"NOMINAL" TRANSIENT CONDITIONS		
		FLUID	TEMP °F	PRESS PSIA	TEST DURATION (SEC)	VALVE CLOSURE PRESS. (PSIA)	MAX DISCH. PIPE B.P. PSIA
1*	Steam	Steam	(Sat.)	2455	60	2255	485
2 - 5	Steam	Steam	(Sat.)	2455	15	2255	485
6*	Steam	Steam	(Sat.)	2415	60	2235	155
7 - 10	Steam	Steam	(Sat.)	2415	15	2255	155

* Tests 1 and 6 were extended duration flow measurement tests

TABLE 4.9.1b

"AS TESTED" WYLE PHASE III TEST MATRIX FOR THE FISHER CONTROLS RELIEF VALVE

TEST NO.	TEST TYPE	INITIAL CONDITIONS							TRANSIENT CONDITIONS		
		AT VALVE INLET			IN ACCUMULATOR						
		FLUID	TEMP. °F	PRESS. PSIA	FLUID	TEMP °F	PRESS. PSIA	TEST DURATION (SEC)	VALVE CLOSURE PRESS PSIA	MAX DISCHARGE PIPE PRESS. PSIA	MAXIMUM (STATIC+DYNAMIC) BENDING MOMENT INDUCED IN-LB

- LATER -

4.10 GARRETT RELIEF VALVE

4.10.1 Conditions Tested

Tests were performed on the Garrett relief valve model at the Marshall Steam Station and during Phase III of the Wyle Test Program. Tables 4.10.1a and b present the matrix of conditions under which this valve model was tested at Marshall and Wyle (Phase III), respectively.

4.10.2 Summary of Principal Observations

- Marshall Steam Station

  The valve fully opened on demand and fully closed on demand for each of the ten (10) evaluation test cycles.

  Additional cycles were performed on the valve. During these cycles, body to bonnet gasket leakage developed. In all cycles, the valve fully closed on demand. Disassembly showed wash-out of the cage to body gasket. As a result of the test observations, Garrett incorporated design modifications into the test valve for Wyle Phase III tests and into valves being supplied to PWR plants.

  Detailed data sheets are contained in Appendix B, Section B-10a.

- Wyle Phase III

  \- later -

TABLE 4.10.1a

"AS TESTED" MARSHALL TEST MATRIX FOR THE GARRETT RELIEF VALVE

TEST NO.	TEST TYPE	"NOMINAL" INITIAL CONDITIONS AT VALVE INLET			"NOMINAL" TRANSIENT CONDITIONS		
		FLUID	TEMP °F	PRESS PSIA	TEST DURATION (SEC)	VALVE CLOSURE PRESS. (PSIA)	MAX DISCH. PIPE B.P. PSIA
1*	Steam	Steam	(Sat.)	2445	60	2015	815
2 - 5	Steam	Steam	(Sat.)	2445	15	2045	815
6*	Steam	Steam	(Sat.)	2615	60	2035	335
7 - 10	Steam	Steam	(Sat.)	2615	15	2465	345

* Tests 1 and 6 were extended duration flow measurement tests

TABLE 4.10.1b

"AS TESTED" WYLE PHASE III TEST MATRIX FOR THE GARRETT RELIEF VALVE

TEST NO.	TEST TYPE	INITIAL CONDITIONS							TRANSIENT CONDITIONS		
		AT VALVE INLET			IN ACCUMULATOR						
		FLUID	TEMP. °F	PRESS. PSIA	FLUID	TEMP °F	PRESS. PSIA	TEST DURATION (SEC)	VALVE CLOSURE PRESS. PSIA	MAX DISCHARGE PIPE PRESS. PSIA	MAXIMUM (STATIC+DYNAMIC) BENDING MOMENT INDUCED IN-LB

\- LATER -

II.D.1 PERFORMANCE TESTING OF BOILING-WATER REACTOR AND PRESSURIZED-WATER REACTOR RELIEF AND SAFETY VALVES (NUREG-0578, SECTION 2.1.2)

Position

Pressurized-water reactor and boiling-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.

Changes to Previous Requirements and Guidance

A. Safety and Relief Valves and Piping--The types of documentation required for safety and relief valves and piping and the specific submittal dates are considered to be a clarification of item II.D.1 as described in NUREG-0660. The submittal of information was implied but not explicitly discussed in that report.

B. Block Valves--Qualification of PWR block valves is a new requirement. Since block valves must be qualified to ensure that a stuck-open relief valve can be isolated, thereby terminating a small loss-of-coolant accident due to a stuck-open relief valve. Isolation of a stuck-open power-operated relief valve (PORV) is not required to ensure safe plant shutdown. However isolation capability under all fluid conditions that could be experienced under operating and accident conditions will result in a reduction in the number of challenges to the emergency core-cooling system. Repeated unnecessary challenges to these system are undesirable.

C. ATWS Testing--Testing of anticipated transients without scram (ATWS) for later phases of the valve qualification program was noted in item II.D.1 of NUREG-0660. The clarification below provides updated information on PWR ATWS temperature and pressure conditions and clarifies that ATWS testing need not be accomplished by July 1981.

Clarification

Licensees and applicants shall determine the expected valve operating conditions through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Revision 2. The single failures applied to these analyses shall be chosen so that the dynamic forces on the safety and relief valves are maximized. Test pressures shall be the highest predicted by conventional safety analysis procedures. Reactor coolant system relief and safety valve qualification shall include qualification of associated control circuitry, piping, and supports, as well as the valves themselves.

ENCLOSURE 3

Enclosure 3

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

TO ALL LICENSEES OF OPERATING PLANTS AND APPLICANTS FOR OPERATING LICENSES AND HOLDERS OF CONSTRUCTION PERMITS

Gentlemen:

SUBJECT: REVISED SCHEDULE FOR COMPLETION OF TMI ACTION PLAN ITEM II.D.1, RELIEF AND SAFETY VALVE TESTING

On October 31, 1980 the NRC staff transmitted a Clarification of TMI Action Plan Requirements (NUREG-0737). Item II.D.1 of that document "Relief and Safety Valve Test Requirements" set forth implementation schedules of 7/1/81 for completion of the RV & SV test program and 10/1/81 for the submittal of plant specific reports.

We have completed our review of a request for schedule relief for completing that portion of the item related to the PWR (EPRI) testing program. The Commission has approved a revised schedule in response to this request. The revision, as indicated in the enclosed page changes to NUREG-0737, extends completion of the test program until April 1, 1982 and of the plant specific reports until July 1, 1982.

Sincerely,

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure:
NUREG-0737 Revised Pages
1-5, 2-6, 3-72, 3-74

Clarification Item	Shortened Title	Description	Implementation Schedule	Plant Applicability	Requirements Issued	Clarification Issued	Preimplementation Approval	Postimplementation Review	Tech Spec Req.	Licensee Submittal Req. by	Remarks
I.D.2	Plant-safety-parameter display console	1. Description	TBD	All	6/26/80	Encl 3				Later	Guidance per NUREG-0696; Rev. 2
		2. Installed	TBD	All	6/26/80	Encl 3					
		3. Fully implemented	TBD	All	6/26/80	Encl 3					
II.B.1	Reactor-coolant-system vents	1. Design vents	7/1/81	All	9/13/79	10/30/79	No	Yes	No	7/1/81	Complete
		2. Install vents (LL Cat B)	7/1/82	All	9/13/79	10/30/79	Yes	No	Yes	7/1/81	
		3. Procedures	1/1/82	All	9/13/79	Encl 3	Yes	No	Yes	1/1/81	
II.B.2	Plant Shielding	1. Review designs	1/1/80	All	9/13/79	10/30/79	No	Yes	No	1/1/80	Complete
		2. Plant modifications (LL Cat B)	1/1/82	All	9/13/79	10/30/79 Encl 3	No	Yes	No	1/1/82	
		3. Equipment qualification	6/30/82	All	CLI-80-21	Encl 3	No	Yes	No	11/1/80	
II.B.3	Postaccident sampling	1. Interim system	1/1/80	All	9/13/79	10/30/79	No	Yes	No	1/1/80	Complete
		2. Plant modifications (LL Cat B)	1/1/82	All	9/13/79	10/30/79 Encl 3	No	Yes	Yes	1/1/81 submittal if deviation from position	
II.B.4	Training for mitigating core damage	1. Develop training program	1/1/81	All	3/28/80	3/28/80 Encl 3	No	Yes	No	1/1/81	
		2. Implement program									
		a. Initial	4/1/81	All	3/28/80	Encl 3	No	Yes	No		
		b. Complete	10/1/81	All	3/28/80	Encl 3	No	Yes	No		
II.D.1	Relief & safety-valve test requirements	1. Submit program	1/1/80	All	9/13/79	10/30/79	No	No	No	1/1/80	Complete
		2. RV & SV testing (LL Cat B)									
		a. Complete testing	7/1/81	BWRs	9/13/79	10/30/79	No	No	No	7/1/81	
			4/1/82	PWRs	9/13/79	10/30/79	No	No	No	4/1/82	
		b. Plant-specific report	10/1/81	BWRs	9/13/79	10/30/79	Yes	Yes	TBD	1/1/82	
			7/1/82	PWRs	9/13/79	10/30/79	Yes	Yes	TBD	7/1/82	
		3. Block-valve testing	7/1/82	PWR		Encl 3	Yes	Yes	TBD	7/1/82	
II.D.3	Valve position indication	1. Install direct indications of valve position	1/1/80	All	9/13/79	10/30/79	No	Yes	Yes	1/1/80	Complete
		2. Tech Specs	12/15/80	All	7/2/79	7/2/80	Yes	No	Yes	9/1/80	

Clarification Item	Shortened Title	Description	Implementation Schedule	Plant Applicability	Requirements Issued	Clarification Issued	Preimplementation Approval	Postimplementation Review	Tech Spec Req.	Licensee Submittal Req. by	Remarks
II.D.1	Relief & safety-valve test requirements	1. Describe program & schedule	Fuel load	All	9/27/79	11/9/79			No		
		2. RV & SV tests	Fuel load	BWR	9/27/79	11/9/79			TBD		
			Fuel load or by 7/1/82 whichever is later	PWR							
		3. Block Valve Tests	Fuel load or by 7/1/82, whichever is later	PWR	*	11/9/79 Encl 3					
II.D.3	Valve position indication	Install in control room	Δ	All	9/27/79	11/9/79 Encl 3			Yes		
II.E.1.1	Auxiliary Feedwater system evaluation	1. Analysis	Full power	CE & W	3/10/80	None			No		See 3/10/80 and 4/24/80 letters
				B&W	4/24/80	None			No		
		2. Modification	Full power	PWR	4/24/80	None			As required		
II.E.1.2	Auxiliary feedwater system initiation and flow	1. Initiation									
		(a) Control grade	Fuel load	PWR	9/27/79	11/9/79			Yes		
		(b) Safety grade	Δ	PWR	9/27/79	11/9/79			Yes		
		2. Flow indication									
		(a) Control grade	Fuel load	PWR	9/27/79	11/9/79			Yes		
		(b) Safety grade	Δ	PWR	9/27/79	11/9/79			Yes		
II.E.3.1	Emergency power for pressurizer heaters	Installed capability	4 mos prior to issuance of SER	PWR	9/27/79	11/9/79 Encl 3			Yes		
II.E.4.1	Dedicated hydrogen penetrations	1. Design	Δ	All	9/27/79	11/9/79			No		
		2. Review & revise H_2 control proc	Fuel load	All	9/27/79	Encl 3			No		
		3. Install	7/1/81 or prior to issuance of OL	All	9/27/79	Encl 3			No		

*Requirement formally issued by this letter

ΔFour months before operating license is issued or 4 months before date indicated

II.D.1 PERFORMANCE TESTING OF BOILING-WATER REACTOR AND PRESSURIZED-WATER REACTOR RELIEF AND SAFETY VALVES (NUREG-0578, SECTION 2.1.2)

Position

Pressurized-water reactor and boiling-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.

Changes to Previous Requirements and Guidance

A. Safety and Relief Valves and Piping--The types of documentation required for safety and relief valves and piping and the specific submittal dates are considered to be a clarification of item II.D.1 as described in NUREG-0660. The submittal of information was implied but not explicitly discussed in that report.

B. Block Valves--Qualification of PWR block valves is a new requirement. Since block valves must be qualified to ensure that a stuck-open relief valve can be isolated, thereby terminating a small loss-of-coolant accident due to a stuck-open relief valve. Isolation of a stuck-open power-operated relief valve (PORV) is not required to ensure safe plant shutdown. However isolation capability under all fluid conditions that could be experienced under operating and accident conditions will result in a reduction in the number of challenges to the emergency core-cooling system. Repeated unnecessary challenges to these system are undesirable.

C. ATWS Testing--Testing of anticipated transients without scram (ATWS) for later phases of the valve qualification program was noted in item II.D.1 of NUREG-0660. The clarification below provides updated information on PWR ATWS temperature and pressure conditions and clarifies that ATWS testing need not be accomplished by July 1981.

Clarification

Licensees and applicants shall determine the expected valve operating conditions through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Revision 2. The single failures applied to these analyses shall be chosen so that the dynamic forces on the safety and relief valves are maximized. Test pressures shall be the highest predicted by conventional safety analysis procedures. Reactor coolant system relief and safety valve qualification shall include qualification of associated control circuitry, piping, and supports, as well as the valves themselves.

A. Performance Testing of Relief and Safety Valves--The following information must be provided in report form by October 1, 1981 for BWRs and July 1, 1982 for PWRs.

(1) Evidence supported by test of safety and relief valve functionability 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 the expected flow conditions.

Documentation Required

Preimplementation review will be based on EPRI, BWR, and applicant submittals with regard to the various test programs. These submittals should be made on a timely basis as noted below, to allow for adequate review and to ensure that the following valve qualification dates can be met:

Final PWR (EPRI) Test Program--July 1, 1980
Final BWR Test Program--October 1, 1980
Block Valve Qualification Program--January 1, 1981

Postimplementation review will be based on the applicants' plant-specific submittals for qualification of safety relief valves and block valves. To properly evaluate these plant-specific applications, the test data and results of the various programs will also be required by the following dates:

BWR Generic Test Program Results--July 1, 1981
PWR (EPRI) Generic Test Program Results--April 1, 1982
Plant-specific submittals confirming adequacy of safety and relief valves based on licensee/applicant preliminary review of generic test program results--July 1, 1981 for BWRs; April 1, 1982 for PWRs
Plant-specific reports for safety and relief valve qualification--October 1, 1981 for BWRs; July 1, 1982 for PWRs
Plant-specific submittals for piping and support evaluations--January 1, 1982 for BWRs; July 1, 1982 for PWRs
Plant-specific submittals for block valve qualification--July 1, 1982

Technical Specification Changes Required

No technical specification changes are required.

References

NUREG-0578

NUREG-0660, Item II.D.1

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of METROPOLITAN EDISON CO. ET AL. (Three Mile Island Nuclear Station, Unit 1)	)))))	Docket No. 50-289 (Restart)

JOINT AFFIDAVIT OF EDGAR G. HEMMINGER AND WALTON L. JENSEN, JR.

Edgar G. Hemminger and Walton L. Jensen, Jr., state under oath as follows:

1. I, Edgar G. Hemminger, am a Mechanical Engineer in the Division of Engineering, Mechanical Engineering Branch, Office of Nuclear Reactor Regulation of the United States Nuclear Regulatory Commission. I am responsible for review and evaluation of structural integrity, operability, and functional capability of safety related mechanical equipment, which includes evaluation of unsatisfactory safety and relief valve test results. A copy of my professional qualifications is attached.

2. I, Walton L. Jensen, Jr., am a senior engineer assigned to the Reactor Systems Branch, Office of Nuclear Reactor Regulation of the United States Nuclear Regulatory Commission. I am currently responsible for the branch review of TMI-1. A copy of my professional qualifications is attached.

3. The "NRC Staff's Report to the Board on Safety Aspects of EPRI Test Data on Relief and Safety Valves" was prepared by us and is true and correct to the best of our knowledge and belief.

Edgar G. Hemminger
Edgar G. Hemminger

Walton L. Jensen, Jr.
Walton L. Jensen, Jr.

Subscribed and sworn to before
me this 3rd day of September, 1981.

Linda M. Egler
Notary Public
My Commission Expires: July 1, 1982

EDGAR G. HEMMINGER

OFFICE OF NUCLEAR REACTOR REGULATION

U. S. NUCLEAR REGULATORY COMMISSION

PROFESSIONAL QUALIFICATIONS

I am a Mechanical Engineer in the Division of Engineering, Mechanical Engineering Branch, Office of Nuclear Reactor Regulation of the United States Nuclear Regulatory Commission. I am responsible for review and evaluation of the structural integrity, operability, and functional capability of safety related mechanical equipment and components.

I hold a Bachelor of Science Degree in Mechanical Engineering from Ohio University and a Master of Science Degree in Mechanical Engineering from Drexel University and am a licensed Professional Engineer in the State of New York.

From 1965 thru 1979, I was employed by the General Electric Company at the Knolls Atomic Power Laboratory in Schenectady, New York. My work experience was in the area of thermal and stress analysis of reactor plant components and equipment. I have specifically evaluated steam generators, reactor vessels, nozzles, closure heads, pumps and piping systems. Using finite element computer methods, I have modeled the vessel closure head and core barrel bolt up region to determine preload relaxation and lift off for various operating and accident conditions. I have also used results of the above type calculations in conjunction with fracture mechanics methods to establish safe heat up and cooldown pressure and temperature limits for normal plant operation.

In 1973, I completed a one year training program for test and start up of naval reactor plants aboard ship. From 1973 thru 1979, I contributed to the construction, start up and power range physics testing of eight reactor plants aboard ship. My primary duties were to review the test procedures and test data for acceptance testing of naval reactor plants aboard ship and to provide technical support to the shipyard in resolution of equipment problems dealing primarily with valves, pumps, and heat exchangers.

I joined the NRC in October, 1979.

WALTON L. JENSEN, JR.

PROFESSIONAL QUALIFICATIONS

I am a Senior Nuclear Engineer in the Reactor Systems Branch of the Nuclear Regulatory Commission. In this position I am responsible for the technical analysis and evaluation of the public health and safety aspects of reactor systems.

From June 1979 to December 1979, I was assigned to the Bulletins and Orders Task Force of the Nuclear Regulatory Commission. I participated in the preparation of NUREG-0565, "Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox Designed 177-FA Operating Plants."

From 1972 to 1976, I was assigned to the Containment Systems Branch of the NRC/AEC, and from 1976 to 1979, I was assigned to the Analysis Branch of the NRC. In these positions I was responsible for the development and evaluation of computer programs and techniques to calculate the reactor system and containment system response to postulated loss-of-coolant accidents.

From 1967 to 1972, I was employed by the Babcock and Wilcox Company at Lynchburg, Virginia. There I was lead engineer for the development of loss-of-coolant computer programs and the qualification of these programs by comparison with experimental data.

From 1963 to 1967, I was employed by the Atomic Energy Commission in the Division of Reactor Licensing. I assisted in the safety reviews of large power reactors, and I led the reviews of several small research reactors.

I received an M.S. degree in Nuclear Engineering at the Catholic University of America in 1968 and a B.S. degree in Nuclear Engineering at Mississippi State University in 1963.

I am a graduate of the Oak Ridge School for Reactor Technology, 1963-1964.

I am a member of the American Nuclear Society.

I am the author of three scientific papers dealing with the response of B&W reactors to Loss-of-Coolant Accidents and have authored one scientific paper dealing with containment analysis.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
METROPOLITAN EDISON CO. ET AL.	)	Docket No. 50-289
(Three Mile Island Nuclear	)	
Station, Unit 1)	)	(Restart)

NRC STAFF'S REPORT ON

BOARD'S COMMENTS REGARDING BOARD NOTIFICATION

OF UNSATISFACTORY TEST RESULTS OF SAFETY VALVES

In its Order dated August 25, 1981, the Atomic Safety and Licensing Board for the Three Mile Island Nuclear Station, Unit 1 (TMI-1) Restart Hearing noted that the NRC staff did not notify the Board on the TMI-1 proceeding regarding the unsatisfactory Electric Power Research Institute (EPRI) test results for the safety valve installed in TMI-1. A staff memorandum from J. P. Knight to R. L. Tedesco and T. M. Novak dated July 1, 1981 enclosed the EPRI memorandum of June 26, 1981 which reported on the tests. The TMI-1 Board became aware of the matter through NRC Board Notification No. 81-20, dated August 11, 1981, filed in the McGuire proceeding. The Board Order requested the staff, among other things, to inform the Board promptly whether notification of this matter by the staff would have been appropriate in this proceeding, and if why not.

The members of the staff that prepared this report discuss their reasoning herein as to why notification of the TMI-1 Board was not considered appropriate. However, The Director, Division of Licensing, was not provided the opportunity, in accordance with current guidelines in Nuclear Reactor Regulation (NRR) Office Letter No. 19 Rev. 1 dated December 9, 1980 (enclosed), to review the recommendations and the EPRI test results to make his determination as to whether the test results were material and relevant. In retrospect, The Director, Division of Licensing would have likely decided to notify the TMI-1 Board similar to the notification filed in the McGuire proceeding based on the Commission's policy cited in the NRR Office Letter No. 19. However, the staff discusses below why it believes that the unsatisfactory EPRI test results reported in the June 26, 1981 EPRI memorandum are not significant with respect to the issues in the TMI-1 proceeding.

The staff reviewed the unsatisfactory EPRI test results reported in the EPRI memorandum dated June 26, 1981 regarding their relevance and safety significance to the issues in the TMI-1 proceeding prior to considering notifying the TMI-1 Hearing Board. The basis for the unsatisfactory test report was that rated flow in accordance with the EPRI screening criteria was not met during a high back pressure steam test.

This test was only one part of the early phase of the EPRI test program and although some screening test criteria have not been met, the testing to date has not identified a safety problem with the safety or relief valves that would affect the staff's position on the TMI-1 hearing record. The "NRC Staff's Report to the Board on Safety Aspects of EPRI Test Data on Relief and Safety Valves" that was prepared by Edgar G. Hemminger and Walton L. Jensen, Jr. provides a more detailed discussion of the valve test results. The principal staff concern stated in the TMI-1 hearing record on this matter was the need to demonstrate that the safety and relief valves can withstand loadings from two-phase and solid flow; Zudans (UCS 6), ff. Tr 8824, at 5; and those EPRI tests on safety valves had not yet been conducted. Testing to date involving two-phase and solid flow for the Dresser type power operated relief valve as used on TMI-1 does not show unacceptable results. Therefore, for the reasons stated above we did not believe the failure of a safety valve to meet EPRI screening criteria during this steam test to be significant with respect to the issues in the TMI-1 proceeding.

In addition to the EPRI test report of June 26, 1981, the staff received other EPRI test reports on relief and safety valves of the TMI-1 type, some of which show test results that deviate from the EPRI screening criteria. In the cases discussed below, the staff also concluded that the results were not material to the TMI-1 hearing record issues:

1. EPRI test report dated May 15, 1981 (enclosed in Staff memorandum from J. P. Knight dated May 19, 1981 to Tedesco and Novak) noted unsatisfactory test results on a Dresser (power operated relief valve (PORV) of the type used at TMI-1. In that test, the unsatisfactory results were associated with the effects of an upstream simulated water seal. Since the PORV at TMI-1 does not have a water seal feature, the staff concluded that the water seal test effects should not be representative of TMI-1 valve behavior.

2. EPRI test report dated July 2, 1981 (enclosed in Staff memorandum from J. P. Knight dated July 16, 1981 to Tedesco and Novak).

3. EPRI test report dated July 10, 1980 (enclosed in Staff memorandum from J. P. Knight dated August 6, 1981 to Tedesco and Novak).

Reports 2 and 3 included results of tests on the Dresser safety valves of the type used at TMI-1. In those tests, rated flow was achieved but valve closing pressures were below the EPRI screening criteria for valve closure pressures. The staff does not believe that the valve closing pressure test results are material to the TMI-1 hearing, since the valve acceptably performed its minimum relief capacity function. Also the delayed closure is not an unreviewed safety concern, and further, does not correspond to a pressure level that would challenge the plant's engineered safety features. The test results would not affect the staff's position on the TMI-1 hearing record.

Copies are enclosed for the Board's information of the four memoranda cited in this report from J. P. Knight to R. L. Tedesco and T. M. Novak that enclosed the EPRI memoranda reporting on the valve tests.

closure pressures. The staff does not believe that the valve closing pressure test results are material to the TMI-1 hearing, since the valve acceptably performed its minimum relief capacity function. Also the delayed closure is not an unreviewed safety concern, and further, does not correspond to a pressure level that would challenge the plant's engineered safety features. The test results would not affect the staff's position on the TMI-1 hearing record.

Copies are enclosed for the Board's information of the four memoranda cited in this report from J. P. Knight to R. L. Tedesco and T. M. Novak that enclosed the EPRI memoranda reporting on the valve tests.

Notification of Unsatisfactory Test Results on Safety Valves

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

DECEMBER 9 1980

MEMORANDUM FOR: Darrell G. Eisenhut, Director, Division of Licensing
Richard H. Vollmer, Director, Division of Engineering
Stephen H. Hanauer, Director, Division of Human Factors Safety
Denwood F. Ross, Director, Division of Systems Integration
Thomas E. Murley, Director, Division of Safety Technology
Bernard J. Snyder, Program Director, TMI Program Office

FROM: Harold R. Denton, Director, Office of Nuclear Reactor Regulation

SUBJECT: NRR OFFICE LETTER NO. 19, REVISION 1
PROCEDURES FOR NOTIFICATION TO LICENSING BOARDS OF RELEVANT AND MATERIAL NEW INFORMATION

Effective immediately, all NRR personnel will use the following revised procedures for assuring prompt and appropriate action on notifying Licensing Boards, Appeal Panel and the Commission of new information which is considered by the staff to be relevant and material to one or more licensing proceedings. These revised procedures reflect the experience we have gained since issuing the original Office Letter No. 19 on July 6, 1978.

This Office Letter places an obligation on all NRR staff members to be alert to the significance of new information that is developed in the course of their review and to consider whether this information could reasonably be regarded as putting a new or different light upon an issue before Boards or as raising a new issue after publication of the staff's principal evidentiary documents. This is the central theme of the procedures and requires the exercise of good judgment to assure that Boards will not be burdened with material beyond that potentially significant to the individual licensing proceedings.

H R Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosure:
Board Notification
Procedure

cc: E. Christenbury, OELD
R. Rosenthal, ASLAP
R. Lazo, ASLBP

DUPE OF
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A. BACKGROUND

Following Commission approval of its Board Notification policy on May 4, 1978, the Office of Nuclear Reactor Regulation issued NRR Office Letter No. 19, dated July 6, 1978, which contained Board Notification procedures to be implemented by NRR. The term "Board Notification" refers to new information which is considered to be relevant and material to one or more licensing proceedings', i.e., material relating to an issue before a Licensing Board, Appeal Panel, or the Commission which can reasonably be regarded as putting a new or different light on that issue, or raising a new issue. (Note that the term "Board" will be used in this procedure to refer to Licensing Boards, Appeal Panel and Commission.)

In a memorandum dated May 10, 1978, the Commission requested that an evaluat of the Board Notification policy be prepared when approximately one year of experience was available. To this end, Commission Paper SECY-80-129, dated March 10, 1980, provided an assessment of then current procedures and proposed changes to those procedures to correct problems encountered in carrying out the Board Notification policy.

B. DISCUSSION

There were three significant changes to the Board Notification procedures recommenced in SECY-80-129 and approved by the Commission:

1. Change the time threshold for initiating the formal Board Notification procedures from the issuance of the ACRS Supplement and FES to 30 days before the start of the evidentiary hearing.
2. Eliminate the routine transmittal to the Boards of staff correspondence and notices to applicants and licensees. Staff correspondence and notices to applicants and licensees would be sent to the Board only if it is determined to meet the guidelines for Board Notification, i.e., new information considered material and relevant.
3. Incorporate the guidelines for staff appraisal and evaluation of Board Notification matte. set forth in ALAB-551, as follows:
   a. supply an exposition adequate to allow a ready appreciation of the precise nature of the Board Notification matter;
   b. supply an exposition adequate to allow a ready appreciation of the extent to what the Board Notification matter might have a bearing upon the particular facility before the board;
   c. in the event a conclusion with regard to the safety or environmenta significance of the Board Notification matter is presented, set for the reasoning underlying that conclusion sufficient to allow the board to make an informed judgment on the validity of the conclusic and

d. where the board has limited jurisdiction, spell out the possible relationship between the subject matter of the notification and one or more of the issues before the board.

C. DETERMINATION OF RECOMMENDATIONS FOR BOARD NOTIFICATION BY TECHNICAL REVIEW GROUPS AND PROJECT MANAGERS

The Board Notification policy is applicable to operating license proceedings as well as construction permit proceedings. In these proceedings the staff will send new information relevant and material to safety or environemntal issues to the Boards regardless of the specific issues which have been placed in controversy. This practice includes proceedings for the conversion of provisional to full-term operating licenses. In hearings concerning operating license amendments Board Notification is limited to the issues under consideration in the hearing. All staff members are responsible for reviewing all information received in the course of their assigned tasks, including reports identified by the Research and Standards Coordination Branch as being appropriate for consideration for Board Notification, to determine whether it may be related to licensing proceedings and may represent relevant and material new information which should be provided to appropriate Boards.

Information received from outside sources and considered to be suitable for Board Notification should be handled in an expeditious manner. Some examples of information from outside sources are: (1) the reporting of errors discovered in a vendors Emergency Core Cooling System (ECCS) models or codes which could result in changes to analyses previously evaluated and discussed in the SER, (2) the reporting of geological features which could result in significant changes to those previously reported by the applicant and evaluated by the staff as discussed in the SER, and (3) those reports identified by the Research and Standards Coordination Branch as being appropriate for consideration for Board Notification.

Internally generated information that could reasonably be regarded as putting a new or different light upon an issue before Boards should also be reported as expeditiously as practicable. However, the Commission's policy recognizes the difficulty of determining the point when an individual staff member's perceived concern has developed into a staff issue of sufficient importance that Boards are to be notified. In accordance with the Commission's policy, internally generated information should be provided to Boards at the point when the staff determines that it is necessary to get more information about a problem from a source external to the staff. That is, if such new information is determined to be of sufficient importance to seek further information, analyses, tests, etc., from licensees or vendors, NRC contracts, or others outside the NRC staff, then the issue has developed to the point where concerned Boards should be informed.

As for internally generated information, technical papers and journal articles should be provided to Boards at a point when the staff determines that (1) such information is of sufficient importance to call into question staff positions and criteria or (2) the staff has determined to seek further information, analyses, tests, etc., from licensees, vendors, NRC contractors or others outside the staff.

1. Staff members should provide promptly the following information, through their management, to the Director, Division of Licensing:

   a. The item recommended for notification of Boards.

   b. An exposition adequate to allow a ready appreciation of the precise nature of Board Notification matter.

   c. Considerations regarding relevancy and materiality; i.e., putting a new or different light upon an issue before the Board or raising a new issue.

   d. An exposition adequate to allow a ready appreciation of the extent to what the Board Notification matter might have a bearing upon the particular facility before the Board.

   e. A statement as to the perceived significance of the information as it may affect current staff positions. (A clear assessment of the significance is not required at this time and the recommendation should not be delayed in order to permit lengthy determinations. If a clear assessment and final resolution is available, it obviously provides for a clean Board submittal. For all recommendations which do not contain a final resolution followup action is required to inform the Boards as to the ultimate staff disposition.)

   f. In the event a conclusion with regard to the safety or environmental significance of the Board Notification matter is presented, set forth the reasoning underlying that conclusion sufficient to allow the Board to make an informed judgment on the validity of the conclusion.

   g. Where the Board has limited jurisdiction, spell out the possible relationship between the subject matter of the notification and one or more of the issues before the Board.

h. If the information relates to a specific docket, a statement as to possible applicability to other dockets.

2. NRR also has a responsibility for identifying information potentially relevant and material to Boards considering facilities licensed under Part 70 and under the cognizance of the Office of Nuclear Material Safety and Safeguards (NMSS). Staff members should make any such recommendations through their management to the Director, Division of Licensing. The information provided should, to the extent possible, conform to that listed in Item 1. above. The Director, Division of Licensing, will forward the Board Notification material to the Director, Office of Nuclear Material Safety and Safeguards.

3. Recommendations may be judged by the Director, Division of Licensing, not to be material and relevant and a memorandum to that effect will be provided to the originator. If the originator still feels that the information should be provided to Boards, he or she should so state in a followup recommendation. Such a followup recommendation will be processed through the normal Board Notification channels. Although comments may be added indicating disagreement by those who judged the information not to be relevant and material, it will be forwarded to the Board.

4. Board Notifications on differing professional opinions will follow the procedures of NRC Manual Chapter 4125, "Differing Professional Opinions."

D. PROCESSING OF BOARD NOTIFICATION RECOMMENDATIONS

1. The key to commencement of Board Notifications on a specific case is the establishment of the date for the beginning of evidentiary hearing and issuance of related notice by the Board. Prior to 30 days before the hearing, new material which is considered material and relevant to a proceeding is presented to the Boards via SER supplement or other documents. However, if there are items that have not been appropriately disposed of, a summary list is to be provided by the project manager to the Board 30 days before the start of the hearing. For cases within 30 days of (or during) the evidentiary hearing new material found material and relevant shall be forwarded promptly to the Board according to these procedures.

2. OELD will provide DL with periodic updates of a list of current proceedings for facilities under the cognizance of DL, indicating whether the Licensing Board, Appeal Board or Commission has jurisdiction over proceedings.

3. The Office of the Director, DL, will establish and maintain the record-keeping system related to all Board Notification matters. This will include a log of current proceedings and a detailed list of issues under consideration.

4. The Director, Division of Licensing, shall review all recommendations and determine whether they are relevant and material (5 working days from logging). Recommendations containing information considered to be directly related to a specific case are also reviewed for applicability to other cases. If it is determined that a recommendation is not considered to be relevant and material, a memorandum to that effect is sent to the recommending parties. If the information and accompanying recommendation are not clear enough for a determination to be made, the Director will request clarifying information from the originator.

5. For instances prior to 30 days of the evidentiary hearing, the Director, Division of Licensing, shall forward a memorandum to the cognizant DL Assistant Director(s) advising them that the item be brought to the attention of the Board through incorporation in the SER or as supplemental staff testimony. A copy of the memorandum will be sent to the originator. The project manager is responsible for seeing that the item is covered in evidentiary documents unless it has been determined that the item has been resolved and that Board Notification is not required. Final disposition shall be reported to the Office of the Director, DL (Board Notification Coordinator).

6. For instances within 30 days of (or during) the evidentiary hearing, the Director, Division of Licensing, shall forward a memoranda to the cognizant DL Assistant Director advising them that the item must be brought promptly to the attention of the appropriate Boards. The cognizant DL Assistant Director shall assure that the item is brought promptly to the attention of the Boards (5 working days from receipt of the Director's memorandum). Copies of the Board Notification shall be sent to the originator, technical review group, Office of the Director, DL (Board Notification Coordinator) and OELD (Hearing Division Director and Chief Counsel).

7. A finding by the Director, Division of Licensing, with regard to Board recommendations shall be reviewed by the DL Assistant Directors for applicability to proceedings related to applications for construction permits, post-CP proceedings, applications for operating licenses, as well as proceedings relating to issuance of license amendments. Proceedings related to research and test facilities licensed under Part 50 are to be taken into consideration also.

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

MAY 1 9 1981

MEMORANDUM FOR: Robert L. Tedesco, Assistant Director for Licensing, DL

Thomas H. Novak, Assistant Director for Operating Reactors, DL

FROM: James P. Knight, Assistant Director for Components & Structures Engineering, DE

SUBJECT: REPORTING OF UNSATISFACTORY EPRI/PWR TEST RESULTS FOR POWER OPERATED RELIEF VALVES

References: (a) EPRI memorandum, dated 5/1/81
(b) EPRI memorandum, dated 5/15/81

As described in the referenced memorandums, the Dresser PORV model no. 31533VX-30 and the Target Rock PORV model no. 80X-006-1 failed the initial loop seal simulation tests at Wyle. The valves opened on 110° F water at full pressure 2500 psi, but failed to close as water temperature was ramped up to 650°F, a condition similar to that experienced in plants with loop seals upstream of the PORV's.

The Dresser PORV in question is believed to be installed in CE and B&W PWR's only and Fort Calhoun specifically is known to have loop seals upstream of Dresser PORV's. The Target Rock valve is reportedly not used on any operating plants but is planned for use on some plants presently under construction.

It is requested that operating PWR's and NTOL's be contacted to determine what corrective action, if any, is being taken by the licensees and NTOLs for which the above test results are applicable. In addition, this information may also be relevant for licensing board notification. It is further noted that the Target Rock and Dresser PORV's in question were disassembled and inspected and no visible damage was observed which would affect future operation or testing. The Mechanical Engineering Branch will forward the results of future testing of these valves as they become available.

James P. Knight, Assistant Director for
Components & Structures Engineering
Division of Engineering

cc: R. Vollmer, DE
R. Bosnak, DE
F. Cherny, DE
E. Hemminger, DE
Z. Rosztoczy, DE
R. Woods, IE
E. Jordan, IE

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Hemming

Memorandum

EPRI

May 15, 1981

U.S. NUCLEAR REGULATORY COMMISSION
1981 MAY 18 PM 12 32
TELECOM-BR-OFOS

TO: DISTRIBUTION

FROM: JOHN J. CAREY John J. Carey

SUBJECT: S/RV TEST ACTIVITY

The EPRI/PWR Safety and Relief Valve Test Program testing activities for the period of May 11 - May 15 were as follows:

WYLE

The full pressure preload test and the 110°F/650° water seal test on the Dresser relief valve were performed as scheduled last Friday, May 8. The screening criteria was met for the preload test. The Dresser valve performance for the water seal simulation test was similar to the Target Rock valve performance for the same test condition. The Dresser valve opened as expected. Upon de-energizing the valve for closure, the valve remained open until the valve was isolated from the test loop. Following test valve isolation, the valve closed. The valve did not pass the screening criteria (failure to close on demand). The valve was removed from the test facility and disassembled by the Dresser Representative. No damage was observed that would affect future testing.

During this water seal simulation test larger than expected bending moments were measured in the upstream and downstream piping. It has been speculated that this resulted from the uneven exhausting of the 110°F water through the downstream ramshead. To eliminate the re-occurence of this during future testing the ramshead has been removed.

The Target Rock valve was reinstalled in the test loop. A full pressure steam test was performed Wednesday, May 13. The preload test originally scheduled for Monday, May 11, was performed Thursday, May 14. In addition a steam to 650°F water transition test was performed. For the above tests on the Target Rock valve the screening criteria were met.

Resumption of testing on the Dresser relief valve is scheduled for Monday, May 18. Present plans call for retesting the Dresser valve for the water seal simulation test condition.

(continued)

Fresser
B&W & C&E

Target Rock Valve
on on New Plants

Memorandum

EPRI

May 1, 1981

Jim Brenner

TO: DISTRIBUTION

FROM: JOHN J. CAREY John J Carey

SUBJECT: S/RV TEST ACTIVITY

The EPRI/PWR Safety and Relief Valve Test Program testing activities for the period of April 27 - May 1 were as follows:

WYLE

Testing on the Target Rock valve resumed this week. On Monday, the low pressure 665 psi, 100^{o}F, water test was performed. On Tuesday, the full pressure 2500 psi, 450^{o}F, water test was performed. For these tests, as well as the previous tests, the Target Rock valve opened and closed as expected. The valve passed the screening criteria. On Wednesday, the full pressure 2500 psi, loop seal simulation test was performed. The water just upstream of the valve was 110^{o}F followed by 650^{o}F water. For this test, the valve opened as expected. Upon de-energizing the valve for closure, the valve remained opened for approximately 12 seconds and then closed. The valve did not pass the screening criteria for this condition (failure to close upon demand). The valve was removed from the test facility and disassembled by the Target Rock Representative. No damage was observed that would affect future testing. The valve was re-installed in the test facility. The full pressure 2500 psi, 650^{o}F, water preload test is scheduled for today.

COMBUSTION ENGINEERING

All work on facility construction was completed this week. Pre-test adjustments will continue through the weekend. The first full pressure 2500 psi, steam shakedown test is scheduled for Monday, May 4, 1981.

JJC/WJB/ad

DISTRIBUTION:
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Ed Hemminge
29481

NUCLEAR REGULA

WASHINGTON, D. C. 20555

JUL 1 1981

MEMORANDUM FOR: [illegible] Tedesco, Assistant Director for Licensing, DL

Thomas H. Novak, Assistant Director for Operating Reactors, DL

FROM: James P. Knight, Assistant Director for Components & Structures Engineering, DE

SUBJECT: REPORTING OF UNSATISFACTORY EPRI/PWR TEST RESULTS FOR CONTROL COMPONENTS, INC. POWER OPERATED RELIEF VALVE AND DRESSER MODEL 31739A SAFETY VALVE

The attached memorandum from EPRI for the week of June 26, 1981 discusses the results of both steam and water tests performed at Wyle-Norco on the Control Components, Inc. PORV and the results of steam tests at the CE-Windsor facility on the Dresser 31739A Safety Valve. Note that this is not the same Dresser safety valve discussed in our June 16, 1981 memorandum. As described in the EPRI memorandum, each of these valves failed an EPRI "screening criterion" in one or more of these tests. For each "failure", the applicable criterion is as stated in the memorandum.

It is our understanding that the Licensees and Construction Permit Holders that utilize or plan to utilize one or both of these valves and the NSSS vendors have been notified of these tests results and have the responsibility for assessing the safety significance of the observed valve behavior for their plants.

Our information from EPRI indicates that the Control Components, Inc. PORV is being used or will be used on the following plants:

McGuire 1 and 2
Catawba 1 and 2

The Dresser 31739A Safety Valve is being used or will be used on the following plants:

Calvert Cliffs 1 and 2
Palisades
Midland 1 and 2
Oconee 1, 2 and 3
Crystal River 3
TMI-1
Millstone 2

DUP of
8107160593-XA

Although the specific safety significance of these test results is still being evaluated, this information may be relevant for licensing board notification.

James P. Knight, Assistant Director for
Components & Structures Engineering
Division of Engineering

cc: R. Vollmer
H. Levin
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E. Hemminger
H. Gregg
M. Stolzenberg
Z. Rosztoczy
R. Kiessel
E. Jordan
E. Brown
D. Chaney
R. Clark
S. Varga
W. Johnston
R. Bosnak

EPRI

Memorandum

June 26, 1981

TO: DISTRIBUTION (Attached)

FROM: JOHN J. CAREY John J. Carey

SUBJECT: S/RV TEST ACTIVITIES

The EPRI/PWR Safety and Relief Valve Test Program testing activities for the period of June 22 - 26 were as follows:

WYLE

On Friday, June 19, a 2500 psia, 450°F water test was performed on the Control Components relief valve, utilizing the operator spring force only for closure. The valve opened and closed on demand.

On Saturday, June 20, three additional tests were performed, again utilizing the operator spring force only for closure. The first test was a 500 psia, 100°F water test. The second test was a 500 psia, 450°F water test. For both tests, the valve opened and closed on demand. The third test was a 2500 psia, 650°F water test. The valve opened on demand. Upon signalling the valve for closure, the valve remained opened for approximately 20 seconds. Valve closure occurred at a valve inlet pressure of 2195 psia. EPRI screening criteria requiring valve closure on demand was not met. The valve was disassembled and inspected by the CCI valve Representatives. No damage was observed that would effect future valve performance. The valve was reassembled and the system readied for testing.

On Wednesday, June 24, a 2750 psia, steam test was performed utilizing air pressure to open and close the valve. The valve opened and closed on demand.

On Thursday, two 2500 psia, 650°F water tests were performed. The first utilized air pressure to open and close the valve. The valve opened and closed on demand. The second test utilized operator spring force only for closure. During this test the valve opened on demand. Upon signalling the valve for closure, the valve remained open for approximately 40 seconds. Valve closure occurred at valve inlet pressure of 2040 psia. EPRI screening criteria requiring valve closure on demand was not met. The next test on the CCI valve is scheduled for Monday, June 29.

COMBUSTION ENGINEERING

On Friday, June 19, a high ramp rate, low back pressure, steam test was performed on the Dresser safety valve (31739A). The valve opened at a pressure within ± 3% of the valve design set pressure. A maximum stem position of 60% of rated lift was obtained at a pressure less than 6% above the valve design set pressure. The valve reclosed at a pressure greater than 2250 psig. (During test preparation for the above test, the tank pressure upstream of the valve inadvertently increased resulting in a short duration actuation of the test valve.)

On Monday, June 22, a second (slightly higher) high ramp rate, low back pressure, steam test was performed. The valve opened at a pressure within ±3% of the valve design set pressure. A maximum stem position of 73% of rated lift was obtained at a pressure greater than 6% of the valve design set pressure. Rated flow was achieved.* The valve closed at a pressure greater than 2250 psig.

On Tuesday, a high ramp rate, medium back pressure test was performed. The valve opened within ±3% of the valve design set pressure. A maximum stem position of 67% of rated lift was obtained at a pressure greater than 6% of the valve design set pressure. Rated flow was achieved.* The valve closed at a pressure greater than 2250 psig. Peak back pressure obtained was approximately 420 psig (target steady state back pressure was 435 psig).

On Wednesday, a high ramp rate, low back pressure, steam test with modified valve ring settings was performed. The valve opened at a pressure within ±3% of the valve design set pressure. A maximum stem position of 76% of rated lift was obtained at a pressure greater than 6% of the valve design set pressure. Rated flow was achieved.* The valve closed at a pressure greater than 2250 psgi.

On Thursday, a high ramp rate, high back pressure, steam test, with original ring settings was performed. The valve opened at a pressure within ±3% of the valve design set pressure. A maximum stem position of 65% of rated lift was achieved at a pressure greater than 6% of the valve design set pressure. Rated flow was not achieved.* The EPRI screening criteria was not met. Peak back pressure obtained was approximately 465 psig (target steady state backpressure was 635 psig).

The next steam test on the Dresser safety valve is scheduled for today.

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* Based on preliminary venturi flow data.

JJC/ad

(Distribution attached)