RELATED CORRESPONDENCE

LIC 9/15/80

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

LICENSEE'S TESTIMONY OF

JAMES H. CORREA, GARY T. URQUHART AND ROBERT C. JONES, JR.

IN RESPONSE TO UCS CONTENTIONS 5 AND 6

(VALVES AND VALVE TESTING)

OUTLINE

The purposes and objectives of this testimony are to respond to UCS Contentions 5 and 6, which assert that proper operation of the pressurizer power operated relief valve (PORV) is necessary to mitigate the consequences of accidents, that the failure of the PORV can create or aggravate a loss of coolant accident (LOCA) and that appropriate qualification testing has not been performed to verify the capability of the reactor coolant relief and safety valves. The testimony discusses that the PORV was not designed to fulfill a safety function and is not required for mitigation of design basis LOCA's. It is explained that while the PORV can be actuated and potentially remain open, creating or aggravating a LOCA, analyses have been performed to demonstrate that these transients can be safely mitigated. Changes to minimize the possibility of such an occurrence are also addressed. The testimony continues with a discussion of the original design and testing applied to the pressurizer relief and safety valves. Recent experience at Crystal River 3 during which a safety valve flowed steam, two-phase fluid and water is addressed. Modifications being made to the PORV, and the EPRI valve testing program are described.

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INTRODUCTION

This testimony, by Mr. James H. Correa, Engineer, Mechanical Components, GPU, Mr. Gary T. Urquhart, Unit Manager, Auxiliary Equipment Unit, Babcock & Wilcox Company, and Mr. Robert C. Jones, Jr., Supervisory Engineer, ECCS Analysis Unit, Babcock & Wilcox Company, is addressed to the following contentions:

UCS CONTENTION NO. 5

> Proper operation of power operated relief valves (PORV's), associated block valves and the instruments and controls for these valves is essential to mitigate the consequences of accidents. In addition, their failure can cause or aggravate a LOCA. Therefore, these valves must be classified as components important to safety and required to meet all safety-grade design criteria.

UCS CONTENTION NO. 6

> Reactor coolant system relief and safety valves form part of the reactor coolant system pressure boundary. Appropriate qualification testing has not been done to verify the capability of these valves to function during normal, transient and accident conditions. In the absence of such testing and verification, compliance with GDC 1, 14, 15 and 30 cannot be found and public health and safety is endangered.

UCS withdrew its sponsorship of its Contention No. 6, which has been adopted as a Board Question (<u>See</u> Board Memorandum and

Order of Prehearing Conference of August 12-13, 1980, dated August 20, 1980).

RESPONSE TO UCS CONTENTION NO. 5

BY WITNESS JONES:

UCS Contention 5 states that proper operation of the pressurizer power operated relief valve (PORV) is necessary to mitigate the consequences of accidents and that the failure of the PORV can create or aggravate a loss of coolant accident (LOCA). Contrary to the contention, the PORV is not required for mitigation of design basis LOCA's and, while a LOCA would result if the PORV did not close after being actuated, such as occurred at TMI-2, the safety-grade Emergency Core Cooling System (ECCS) is designed to mitigate the event and to assure adequate core cooling.

The original design function of the PORV was to provide a pressure relief capability which, in conjunction with plant control system actions to reduce reactor power and/or adjust steam generator feedwater flow, would prevent a reactor trip on high primary system pressure during various operational transients. In this manner, unit availability would be enhanced. The relief capability of the PORV was not designed to fulfill a safety function. The high pressure trip function of the Reactor Protection System (RPS) and the pressurizer safety valves provide the required overpressure protection for the Reactor Coolant System. The RPS and the pressurizer safety valves are safety-grade equipment and comply with applicable criteria.

Since the TMI-2 accident the setpoints for the PORV and the high pressure reactor trip setpoint have been inverted. In the original design and operation of TMI-1, the opening pressure for the PORV was 2255 psig and the high pressure reactor trip setpoint was 2355 psig. These setpoints are now 2450 psig and 2300 psig, respectively. As a result, actuation of the PORV is not now expected during operational transients provided that feedwater is delivered to the steam generators in a timely manner. Thus, the frequency of PORV actuation has been reduced.

However, there are still circumstances where the PORV can be actuated and potentially remain open, creating or aggravating a LOCA. Analyses have been performed to demonstrate that these transients can be safely mitigated (as defined by 10 CFR Part 50, Paragraph 50.46(b)) by the ECCS. These analyses included both a stuck-open PORV case (i.e., the PORV causes a LOCA), and a scenario in which a small-break LOCA occurs simultaneously with a loss of all feedwater and results in a subsequent stuck-open PORV (i.e., the PORV aggravates a LOCA) - see Licensee's testimony on Additional LOCA Analysis in response to UCS Contention 8. Additionally, there have been several changes made to enhance the operator's ability to recognize and terminate a transient caused by a stuck-open PORV. Specifically, an accelerometer which senses discharge line flow and discharge line flow measurement instrumentation are being provided. These, along with PORV position demand indication and PORV discharge line temperature measurement, will provide additional assurance that PORV position will be recognized. Also, the PORV and block valve have power supplied by the emergency power system. This provides the capability for closing the block valve upstream of the PORV, in the event of a stuck-open PORV and a loss-of-offsite power.

In summary, and contrary to the above contention, proper operation of the PORV and associated block valve and the instruments and controls for these valves is not essential to mitigate the consequences of design basis LOCA's and, although the failure of the PORV can create or aggravate a LOCA, the consequences of such an accident can be safely mitigated by safety-grade equipment.

RESPONSE TO UCS CONTENTION NO. 6

BY WITNESS URQUHART:

UCS Contention 6 asserts that appropriate qualification testing has not been performed to verify the capability of the reactor coolant relief and safety valves. Contrary to this assertion, these valves - the pressurizer power operated relief valve (PORV) and safety valves have - been properly designed and tested pursuant to applicable criteria.

The pressurizer safety valves are components important to safety in that they are both part of the reactor coolant pressure boundary and functionally provide overpressure protection for the Reactor Coolant System (RCS). The valves were designed for and protect the integrity of the RCS at the design conditions of the primary system - 2500 psig and 670°F. Reference 1 describes in detail the pressure relief criteria for the valves, the method of analysis to develop the criteria, and the results and conclusions of the analysis. As is shown in the referenced document, the RCS is adequately protected by either of the two safety valves since each is capable of relieving the required capacity.

The relief capacity of the safety valves was established consistent with the applicable edition and addenda of Section 9 of Section III of the ASME Boiler and Pressure Vessel Code. This included certification by the valve manufacturer of the capacity of the valves utilizing prototypical testing to establish discharge factors and analytical verification of the ability of the valves to withstand design and operating pressures.

The safety valves were also designed in accordance with the requirements of Section III of the ASME Code to assure reactor coolant pressure boundary integrity. Testing and examination of the valves during and following manufacturing and testing included the following:

(a) Chemical and mechanical testing of the materials.

(b) Volumetric examination of the materials.

(c) Surface examination of the materials.

(d) Hydrostatic pressure testing of the completed valves at the manufacturer and after installation.

(e) Verification of set pressure.

(f) Seat leakage testing following opening and closing.

Also of significance with regard to the capability of the pressurizer safety valves is the transient which occurred February 26, 1980, at the Crystal River nuclear unit, a plant with a B&W nuclear steam system and components similar to TMI-1. During the transient, one of the two safety valves lifted at approximately 2400 psig and flowed saturated steam, two-phase fluid and liquid water. The water flow rate was up to 700 gpm and the valve reseated at approximately 2300 psig, a blowdown of about 4% below the opening pressure.

Subsequent to the transient, the affected valve was subjected to detailed laboratory inspection and testing to determine if any damage had been sustained. The set pressure of the valve was checked three times and determined to be approximately the 2400 psig experienced during the transient. Leakage was measured at about 1.1 gpm. Disassembly and inspection identified steam cutting of the valve disc and a damaged bellows assembly. The steam cutting was most likely caused by leakage that was present prior to the transient. The damage to the bellows did not appear to be due to the February 26, 1980 transient. Neither the steam cutting of the disc nor the damaged bellows impaired the intended pressure relief function of the valve. In summary, no damage detrimental to the proper operation of the valve was discovered even though it had experienced flow conditions other than saturated steam.

The pressurizer PORV was designed for the same system conditions as the safety valves - 2500 psig and 670°F. The valve design was governed by the same ASME Code requirements as the safety valves as it related to pressure boundary integrity, and the valve was tested and examined in a manner similar to the safety valves. Because the PORV is power operated in response to an independent pressure signal, verification of set pressure was not applicable. Verification of valve opening and closing was performed however, prior to shipment and following installation. Also, as discussed in the testimony above in response to UCS Contention 5, the PORV does not serve a pressure relief safety function. Therefore, certification of relief capacity was not required nor was such considered necessary, and an upstream isolation/block valve is allowed by design criteria and is provided. Relief capacity was established by design analysis. The General Design Criteria are applicable to the PORV only to the extent that it forms part of the reactor coolant pressure boundary.

BY WITNESS CORREA:

The PORV which will be installed in TMI-1 prior to restart is the TMI-1 spare PORV. This valve was ordered per the original PORV requirements, was manufactured in 1978, was "N" stamped per Code Case 1581, and in general satisfies the 1977 Edition with the Winter 1979 Addendum of Section III of the ASME B&PV Code for fabrication requirements.

The valve is being modified per the manufacturer's latest design features to improve seat tightness. The modification is being performed per the latest ASME B&PV Code, Section III, requirements. As part of the modification effort, the valve will be disassembled and all critical dimensions will be recorded and checked against drawing requirements. In addition, all moving parts will be inspected for surface finish and signs of wear caused by the original testing of the valve prior to its shipment in 1978. This inspection of the valve internals will ensure that the valve parts meet all requirements. After reassembly of the valve, it will be seat leak tested and opened at its set point. This will ensure that the valve will function properly.

Prior to being installed in TMI-1 the valve will again be seat leak tested. During hot functional testing the valve also will be actuated to ensure its functional ability and to test all downstream instrumentation.

A valve testing program is also in progress. This program is being conducted by the Electric Power Research Institute (EPRI). The purpose of the program is stated in the EPRI Program Plan for the Performance Testing of PWR Safety and Relief Valves, Revision 1, dated July 1, 1980 and is as follows:

> The primary objective of these tests is to evaluate the performance of each of the various types of reactor coolant system safety and relief valves in pressurized water reactor plant service for the range of fluid conditions under which they may be required to operate.[1] The requirements are that:
>
> 1. The safety and relief valves open and close on command, when subjected to simulated plant operational conditions calculated to result in valve actuation.
>
> 2. The flow capacity of the valves be established.
>
> The second objective of the program is to obtain sufficient piping thermal hydraulic and support reaction load data to permit confirmation of analytical models utilized for plant unique analysis of safety and relief valve discharge piping systems.

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1 These conditions will be defined based on an evaluation of the transients specified in Regulatory Guide 1.70, Revision 2.

The program plan to be followed in evaluating the performance of PWR safety and relief valves includes a number of elements which are described in the following:

° A test program will be performend in which selected, actual safety and relief valves are tested under fluid conditions which are calculated to occur during anticipated operational transients and postulated accident sequences in PWR plants. These fluid conditions include steam, water and transition from steam to water. The primary purpose of these tests is to demonstrate that the valves will open and close as required when subjected to simulated transient conditions and that the flow capacity of the valves can be correctly predicted.... It is expected that all testing will be complete by July, 1981.

° A combined test and analysis program will be performed to evaluate the adequacy of analytical methods utilized for PWR safety and relief valve discharge piping response. First, the main valve test facility at Combustion Engineering will include prototypical upstream piping, including water seals, and a simplified discharge piping arrangement which simulates significant features of plant discharge piping systems. These systems will be instrumented to measure dynamic load, piping response and fluid conditions. In parallel with this effort, engineering evaluations are being performed to assess the adequacy of available methods for prediction of safety and relief vavle discharge piping loads. A key part of this effort is the analysis of a number of sample problems using state-of-the-art methods. These problems will include the upstream and discharge piping configurations and ranges of fluid conditions selected for use in the valve performance tests. In addition, analysis of piping configurations representative of actual PWR discharge piping installations has been initiated to demonstrate that the test configuration adequately represents all significant features important to safety and relief valve operation. The combined results of these analytical test programs will provide the data needed to confirm the analytical methods used for piping and support analysis. This information will then be available to utilities for use on a plant-specific basis for evaluation of installed discharge piping systems....

- An evaluation will be performed of available data and experience obtained in foreign valve test facilities, and any domestic test programs that may be applicable. Utilization of other related test experience is considered desirable in order to identify and minimize potential problem areas which might otherwise have an impact on the EPRI test program schedule....

- Effort is underway to evaluate the effects of postulated valve failure modes (e.g., excessive leakage, excessive blowdown, reduced flow capacity, etc.,) on reactor system performance in order to establish preliminary acceptance criteria and guidelines for evaluation of the significance of the valve test results.

- Evaluations of the Crystal River 3 safety and relief valves and piping will be performed. This will be a co-operative effort among EPRI, Florida Power Corporation and Babcock and Wilcox to examine the valves and piping at Crystal River 3 which were subjected to water discharge conditions in February 1980. This evaluation is expected to provide early information on the performance of the affected valves and discharge piping. It may also provide useful information on the effect of service history and aging on valve performance.

(See Mr. Urquhart's testimony above on the Crystal River inspection.)

Met-Ed has submitted its plant specific data (valve drawings and inlet and discharge piping drawings) to EPRI for inclusion in the testing program. One of the relief valve types chosen to be tested is the same model as the TMI-1 relief valve, Dresser model no. 31533VX-30. Also, one of the safety valves types chosen to be tested is the same model as the TMI-1 safety valve, Dresser model no. 31739A.

B&W has supplied operational transient and postulated accident sequence data to EPRI for 177-fuel-assembly reactors (TMI-1 type). This data is being used in defining test parameters for the EPRI test matrix. Therefore the EPRI test results can be directly applied to TMI-1.

As stated in the Restart SER, the EPRI test program is responsive to NRC short term recommendation 2.1.2 of NUREG-0578.

BY WITNESSES CORREA AND URQUHART

In summary, contrary to the above contention, the TMI-1 pressurizer relief and safety valves have been appropriately designed and tested. In addition, actions are being taken to provide further assurance that the valves will function properly and reliably.

REFERENCE

1. Topical Report BAW-10043, "Overpressure Protection for Babcock & Wilcox Pressurized Water Reactors," May, 1972.

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Business Address:	GPU Service Corporation 100 Interpace Parkway Parsippany, New Jersey 07054
Education:	B.S., Mechanical Engineering, Rensselaer Polytechnic Institute, 1969.
Experience:	Mechanical Engineer III, GPU Service Corporation, 1978 to present. Responsible for providing technical engineering on valves for GPU system nuclear power plants; providing technical support to resolve field problems, including repair recommendations and field technical guidance; providing technical support for plant modifications, including writing technical specifications for valves and modification documentation packages. Other responsibilities have included reviewing flow diagrams for proper valve selection; reviewing architect-engineer technical specifications for technical content including referencing the proper codes and standards and valve design features. Mechanical Design Engineer, Foster Wheeler Corporation, 1972 to 1978. Performed engineering work on primary sodium valves for the fast flux test facility and steam generators for a high temperature gas cooled reactor. Responsibilities included preparing material and sub-contracted machining requisition packages; vendor surveillance; preparing and issuing shop fabrication releases which include drawings and shop procedures; and the resolution of vendor material and machining problems and shop fabrication problems in the areas of manufacturing, materials and quality control. Cognizant Engineer, Machinery Apparatus Operation, General Electric Company, 1970 to 1972. Performed technical engineering

work on Naval Nuclear Heat Exchangers and Pressurizers, including definition of specifications, vendor selection, design review and analysis, fabrication surveillance, and the resolution of installation problems. Engineering work included the solving of technical problems in a number of technical disciplines such as mechanical analysis, heat transfer, quality control, materials and welding, and manufacturing.

Engineer, Mechanical Facilities Planning, Missile and Space Division and Re-entry and Environmental Systems Division, General Electric Company, 1969 to 1970. Performed design and cost estimates for specific projects such as ventilation systems and piping systems. Provided design direction for construction and renovation projects.

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Education: B.S., Mechanical Engineering, State University of New York at Buffalo, 1970. M.B.A., Lynchburg College, 1979.

Experience: Unit Manager, Auxiliary Equipment Unit, Equipment Engineering Section, Babcock & Wilcox Co., 1980 to present. Responsible for preparation of equipment specifications for equipment such as valves, heat exchangers, small pumps and tanks, evaluation of vendors' designs, review and approval of vendor submitted documentation, and resolution of field problems.

Senior Engineer and Supervisory Engineer, RCS Mechanical Design Unit, Component Engineering Section, Babcock & Wilcox Co., 1976 to 1980. Responsible for detail design and analysis, manufacturing liaison and resolution of shop and field problems for the reactor internals (core support assembly).

Various assignments in Quality Control (Assurance) and Materials Engineering for the Fossil Power Generation Division, Nuclear Equipment Division and Nuclear Power Generation Division, Babcock & Wilcox Co., 1970 to 1976. Responsibilities included preparation of manufacturing procedures such as non-destructive examination and welding, material selection, evaluation and analysis for fossil boilers and the performance of internal and vendor quality audits.

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Education: B.S., Nuclear Engineering, Pennsylvania State University, 1971. Post Graduate Courses in Physics, Lynchburg College.

Experience: June 1971-June 1975: Engineer, ECCS Analysis Unit, B&W. Performed both large and small break ECCS analyses under both the Interim Acceptance Criteria and the present Acceptance Criteria of 10 CFR 50.46 and Appendix K.

June 1975-Present: Acting Supervisory Engineer and Supervisory Engineer, ECCS Analysis Unit, B&W. Responsible for calculation of large and small break ECCS evaluations, evaluations of mass and energy releases to the containment during a LOCA, and performance of best estimate pretest predictions of LOCA experiments as part of the NRC Standard Problem Program. Involved in the preparation of operator guidelines for small-break LOCA's and inadequate core cooling mitigation.