TESTIMONY OF

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ON BEHALF OF PACIFIC GAS AND ELECTRIC COMPANY

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CONTENTION 24

The Electric Power Research Institute (EPRI) conducts major research programs relating to the safety of nuclear power plants and other related energy research. In August, 1979, the TMI Ad Hoc Nuclear Oversight Committee requested that EPRI develop a generic valve testing program responsive to the recommendation contained in NUREG 0578, Section 2.1.2, "Performance Testing for BWR and PWR Relief and Safety Valves." This program is administered by EPRI, subject to the normal utility advisory review process. In addition, the EPRI-RAC-NSAC Subcommittee provides overall utility industry review of the program. A special subcommittee of the Safety and Analysis Department Task Force, headed by Mr. David Hoffman (Consumers Power Company) was established to provide direct utility review and guidance.

A program plan was developed by EPRI staff in late 1979, reviewed by the PWR utilities and submitted to the NRC on December 17, 1979 by Mr. William J. Cahill, Jr., then

Vice President of Consolidated Edison Company of New York, Inc., and Chairman of the EPRI Safety and Analysis Task Force. As the total program scope was developed and refined during the first few months of 1980, the program plan was revised and modified slightly. The revised program plan, entitled "Program Plan for the Performance Testing of PWR Safety and Relief Valves," July 1, 1980, was submitted by the PWR utilities to the NRC on July 8, 1980.

The total program cost is approximately $18,000,000, and is supported by contributions from 41 electric utilities with pressurized water reactors. Pacific Gas and Electric Company is a participant in this program. The major program cost is associated with the development of relief and safety valve test facilities with the capability to perform all of the tests required by NUREG 0737, Item II.D.1A. Such test facilities did not exist prior to the EPRI program.

The overall objective of the EPRI PWR Safety and Relief Valve Test Program is to obtain full scale data on the operational performance of pressurized water reactor primary system relief and safety valves under expected operating conditions for design basis transients and accidents by July 1, 1981. It is expected that PWR utilities will utilize this data to support plant specific submittals in response to safety and relief valve test

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requirements, first identified in NUREG 0578 and subsequently clarified in NUREG 0737, Item II.D.1A.

The EPRI PWR Safety and Relief Valve Test Program will have four principal program outputs:

1. Relief and safety valve test reports.

2. A report documenting the basis for selection of the relief and safety valves to be tested.

3. A report documenting the basis for the set of fluid test conditions.

4. A report documenting a code for computing hydrodynamic loads for relief and safety valve discharge piping under steam and water discharge.

None of the above reports is presently available. These reports will be submitted to the NRC in preliminary form starting July 1, 1981.

Ten (10) power operated relief valves (PORVs) and nine (9) safety valves have been selected for testing. The valves selected for testing are identified in Table 1. The PWR Valve Test Program was developed so that the valves selected for testing are representative of relief and safety valve designs in use or planned for use in PWR's.

A Masoneilan valve, Model No. 20000 series will be tested. In order to meet the required test completion date of July 1, 1981, this valve was obtained from TVA's Sequoyah Nuclear Power Plant. The Masoneilan Model No. 20000 series valve is believed to be fully representative of the valves utilized as PORVs in the Diablo Canyon Nuclear Power Plant.

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Three sizes of Crosby safety valve Model HB-BP-86 with and without loop seal internal materials will be tested. In particular, the 6M6 size Crosby safety valve Model HB-BP-86 with loop seal internal materials will be tested. This Crosby safety valve is believed to be fully representative of the safety valves utilized in the Diablo Canyon Nuclear Power Plant.

The conditions under which the relief and safety valves are being tested envelope the expected operating and accident conditions as prescribed in the final safety analysis reports (FSARs) for pressurized water reactors. In addition, the conditions resulting from cold pressurization transients and transients resulting from the extended operation of the high pressure liquid injection system will be enveloped.

The test conditions for safety and relief valves include steam, subcooled water, water seal, and steam to water transition discharge conditions.

In order to complete all of the required tests by July 1, 1981, three testing facilities are being utilized. These test facilities are identified below:

1. Marshall Relief Valve Test Facility (Marshall)
   Marshall Steam Station (Duke Power Company)
   Terrell, North Carolina

2. EPRI/Wyle Relief Valve Test Facility (Wyle)
   Wyle Laboratories
   Norco, California

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3. EPRI/CE PWR Safety/Relief Valve Test Facility (CE)
Combustion Engineering
Windsor, Connecticut

The Marshall Facility is being used for relief valve tests under steam conditions. The Wyle Facility is being utilized primarily for relief valve tests under water and water seal conditions. The Combustion Engineering Facility will be used for safety valve tests under steam, subcooled water, water seal, and steam to water transition tests.

Performance screening criteria have been developed for all relief and safety valve tests conducted by EPRI. The performance screening criteria for relief valves are identified below:

1. Valve opens and remains fully open on demand.

2. Valve closes (closure is defined as the valve disk physically returning to its full closed position and does not imply zero seat leakage) and remains fully closed on demand.

3. Valve sustains no external or internal damage having the potential for adversely affecting its normal operations as defined in 1 and 2 above.

Similarly, proposed performance screening criteria for safety valves have been developed and are listed below:

1. Criteria for performance on steam conditions

   a. The valve opens when the inlet pressure is within ± 3% of the valve design set pressure.

   b. The minimum valve open position when the inlet pressure is 6% above valve design set pressure shall be the valve rated lift and/or flow position.

c. The valve fully closes (closure is defined as the valve disk physically returning to its full closed position and does not imply zero seat leakage) when the inlet pressure is less than the pressure at which the valve opened and greater than 2250 psig.

2. Criteria for performance on water/transition conditions

a. The valve opens when the inlet pressure is within ± 3% of the valve design set pressure.

b. The valve fully closes (closure is defined as the valve disk physically returning to its full closed position and does not imply zero seat leakage) when the inlet pressure is less than the pressure at which the valve opened and greater than 2250 psig.

Modifications to the relief valve test loop at the Marshall Steam Station were completed in July, 1980. Testing was initiated in August, 1980 and was completed in January, 1981. In particular, the Masoneilan PORV performed successfully in the Marshall tests and passed all of the performance screening criteria.

Preliminary test data were transmitted by EPRI to PWR utilities with Masoneilan relief valves, including Pacific Gas and Electric Co., on January 9, 1981.

Modifications of the Wyle Test Facility were completed the week of April 19, 1981, and testing was commenced. Testing is scheduled to be completed July 1, 1981. In particular, the Masoneilan relief valve is scheduled for testing at the end of May, 1981.

Test Facility construction at Combustion Engineering is complete and shakedown tests have begun. The current

schedule for the CE safety valve tests calls for completion of all tests by July 1, 1981. In particular, the Crosby safety valve Model HP-BP-86, size 6M6 with loop seal internal materials, is scheduled for testing the week of June 8, 1981.

A formal program for testing PWR pressurizer PORV block valves (hereinafter referred to as block valves), responsive to NUREG 0737, Item II.D.1B, is not part of the EPRI test program scope. However, in addition to the ten (10) relief valves tested at Marshall, seven (7) block valves were also tested. The block valves tested are identified in Table 2.

Two block valves manufactured by Velan were tested at Marshall and performed satisfactorily. Of these two, the Velan valve Model B10-3054B-13MS, drawing no. 88425/B, which is believed to be fully representative of the block valve model utilized in the Diablo Canyon Nuclear Power Plant, was cycled in excess of 21 times, and satisfactorily fully opened and fully closed each cycle. A preliminary draft of the test report for this block valve will be furnished to Pacific Gas and Electric Company in May, 1981.

The results of the seven block valve tests were reviewed with the NRC Staff at a meeting on March 20, 1981 in Bethesda, Maryland. In particular, in preliminary tests of the block valves, three block valve models did not fully close on demand. All three block valves that did not close

initially were subsequently retested with an increased closing thrust and fully closed on demand. The valve seat was also reworked on one block valve prior to retesting. These three block valve models are not utilized as block valves in the Diablo Canyon Nuclear Power Plant.

TABLE 1

PORVs AND SAFETY VALVES SELECTED FOR TEST

PORVs

MANUFACTURER	MODEL #	SIZE
Crosby	HPV-SN	2½" x 4"
Dresser	31533VX-30	2½" x 4"
Target Rock	80X-006	2½" x 4"
Garrett (AiResearch)	3 x 6 Straight Through	3" x 6"
Control Components	3" Drag	3"
Copes Vulcan 17-4PH Plug and Cage	Globe, D-100-160	3"
Copes Vulcan 316 W/Stellite Plug and 17-4 PH Cage	Globe, D-100-160	3"
Fisher Controls	SS-103-SS-95	3"
Masoneilan	20000 Series	2"
Muesco Controls (BS&B)	70-18-9DRTX	2"

SAFETY VALVES

MANUFACTURER	MODEL #	SIZE (SEE NOTE)
Crosby	HP-BP-86	3K6
Crosby	HP-BP-86	3K6 (LS)
Crosby	HP-BP-86	6M6
Crosby	HP-BP-86	6M6 (LS)
Crosby	HP-BP-86	6N8
Crosby	HP-BP-86	6N8 (LS)
Dresser	31739A	2½" x 6"
Dresser	31709NA	6" x 8"
Target Rock	69C	6" x 6"

NOTE: The size of the Crosby safety valve covers the valve inlet size (3", 6"), orifice size (K, M, N), outlet size (6", 8") and whether or not the valve is designed for a loop seal application and test (LS for loop seal, blank for non-loop seal).

The "LS" designation is an EPRI identifier and not a Crosby and/or NSSS vendor identifier.

The difference between loop seal and non-loop seal Crosby valves is internal materials.

TABLE 2

BLOCK VALVES TESTED AT MARSHALL

Anchor Darling

Borg Warner

Rockwell

Velan, Model B10-3054B-13MS, drawing no. 88425/B

Velan, Model B10-3054B-13MS, drawing no. GBH-0300-13MS-MO

Westinghouse, Model 3GM88

Westinghouse, Model 3GM99

PROFESSIONAL QUALIFICATIONS OF

JOHN J. CAREY

My name is John Joseph Carey.

My educational background is as follows:

Illinois Institute of Technology - BS in Mechanical Engineering, 1962

Illinois Institute of Technology - MS in Mechanics, 1966

Illinois Institute of Technology - PhD in Mechanics, 1968

From June, 1957, to January, 1962, I was an Undergraduate Cooperative Education Student at the Illinois Institute of Technology and its Research Institute.

From January to September, 1962, I was an Assistant Research Engineer at the Illinois Institute of Technology Research Institute. My duties included the application of experimental stress techniques to the study of underground structures.

From May, 1965 to January, 1966, I was a Research Engineer for the American Dental Association, responsible for the development and application of Standards for Dental Materials and Devices.

From April, 1968 to March, 1976, I as an Associate Mechanical Engineer in the Reactor Analysis and Safety Group at Argonne National Laboratory. I was responsible for Treat Converter fuel element behavior studies, "In-Pile" and

"Out-of-Pile" experiments to determine the adequacy of candidate Treat Converter fuel element materials with respect to cyclic-thermal shock and fracture resistance; supervision of structural modelling of Treat Converter core (utilizing the MARC-CDC non-linear finite element analysis program); development of structural analysis methods for elastic and inelastic behavior of LMFBR components; development and application of bounding methods for dynamic elastic and creep deformation of structural members, and I directed the ANL assistance effort for analysis of FFTF components (utilizing the ANSYS finite element program).

From March, 1976 to April, 1978, I was a Project Manager in the Safety and Analysis Department of the Electric Power Research Institute, responsible for research projects in the area of structural integrity for nuclear power plant systems and components with specific responsibility for the development of advanced thermal hydraulic and structural analysis methods for BWR containment response under postulated accident conditions.

From April, 1978 to December, 1979, I was a Technical Specialist/Program Engineer with the Safety and Analysis Department of the Electric Power Research Institute with responsibility for the development and implementation of research programs in the area of structural integrity for nuclear power plant systems and components.

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In December, 1979, I was appointed Program Manager with the Safety and Analysis Department of the Electric Power Research Institute with responsibility for the development, implementation, and overall management of the EPRI PWR Safety and Relief Valve Test Program. The objective of this program is to provide full scale data on the operational performance of Pressurized Water Reactor primary system relief and safety valves under expected operating conditions for design basis transients and accidents. The data from this program is expected to be utilized by PWR utilities in response to regulatory requirements for safety/relief valve testing.

A partial list of my publications follows:

"Fluid-Solid Interaction of Reactor Core Components, a Preliminary Analysis." D. Krajcinovic and J. J. Carey, submitted for publication, Nucl. Eng. Des., April 1974.

"Dynamic Response of an Elastic-Linear Strain Hardening Thin Ring Under Impulsive Load," J. J. Carey, LMFBR Steam Generator Systems Development Program, March 1974.

"Thermal-Mechanical Testing of Treat Converter Fuel Elements," J. J. Carey and F. A. Rough, RAS/ANL, March 1974.

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"Evaluation of a Class of Methods for Bounding Steady Creep Deformation," J. J. Carey and R. A. Valentin, ANL 8016, December 1973.

"Fast Flux Test Facility Core Basket Stress Analysis," (Report to Westinghouse Advanced Reactors Division), J. J. Carey, ANL/ETD/AM0534, December 1974.

"On Thermal Stresses in Clad, Pellet Stacks and the Problem of Interface Stress States," J. J. Carey and R. A. Valentin, First International Conference on Structural Mechanics in Reactor Technology, Berlin, September 1971.

"Thermal Stesses and Displacements in Finite, Heat-Generating Circular Cylinders," R. A. Valentin and J. J. Carey, Nucl. Eng. Des., 12 (1970).

"Exact Analysis of Local, Non-Plane, Elastic Stresses in Fuel Element Geometries," R. A. Valentin and J. J. Carey, ANS, June 1970.

PROFESSIONAL QUALIFICATIONS OF

THOMAS E. AUBLE

My name is Thomas E. Auble. I am a Professional Engineer in the State of Pennsylvania.

My educational background is as follows: University of Toledo - BS in Mechanical Engineering, 1973

Since September, 1973, I have been employed by Westinghouse Electric Corporation Power Systems Company in the Water Reactor Division. I have been responsble for valve designs provided for the Westinghouse Nuclear Steam Supply Systems in Westinghouse Nuclear Power Plants. My responsibilities have included development and maintenance of specifications for most Westinghouse supplied valve types; valve vendor design evaluation, procurement negotiations, and vendor selection; stress analysis in accordance with the ASME Boiler and Pressure Vessel Code, Section III for Class 1, 2, and 3 components; development of fluid system functional requirements to insure proper valve and Westinghouse system operation; writing and presenting proposals on costs, schedules, objectives for presentations to Westinghouse management and customers; Lead Design Engineer for Westinghouse manufactured liquid sodium valves; development and implementation of Quality Assurance requirements; and training program development and coordination.

Since July, 1981, I have been on loan to the Electric Power Research Institute Safety and Analysis Department. I am a Project Manager in the EPRI/PWR Safety and Relief Valve Test Program. My specific responsibilities include management of test valve selection, documentation of the basis for selection, valve procurement, valve performance evaluation, and resolution of valve test performance problems. I also manage field support of test facilities and valve test data dissemination.

I am a member of the American Society of Mechanical Engineers.