11/06/81

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
PACIFIC GAS AND ELECTRIC COMPANY	)	Docket Nos. 50-275 O.L.
	)	50-323 O.L.
(Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2)	)	

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NRC STAFF RESPONSE TO JOINT INTERVENORS' SECOND SET OF INTERROGATORIES

I. Interrogatories

Contention 10 (as admitted by the Board in the September 30, 1981 Memorandum and Order)

The Staff recognizes that pressurizer heaters and associated controls are necessary to maintain natural circulation at hot stand-by conditions. Therefore, this equipment should be classified as "components important to safety" and required to meet all applicable safety-grade design criteria, including but not limited to diversity (GDC 22), seismic and environmental qualification (GDC 2 and 4), automatic initiation (GDC 20), separation and independence (GDC 3 and 22), quality assurance (GDC 1), adequate, reliable on-site power supplies (GDC 17) and the single failure criterion. The Applicant's proposal to connect two out of four of the heater groups to the present on-site emergency power supplies does not provide an equivalent or acceptable level of protection.

Contention 12 (as admitted by the Board in the September 30, 1981 Memorandum and Order)

Proper operation of power operated relief valves, 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.

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The interrogatories presented to the Staff by Joint Intervenors requested that each interrogatory be answered in 4 parts. The Staff has labeled the responses A through D, corresponding to Joint Intervenors' request. The Staff objects to Part D(2) of the interrogatories. Any summary of the witnesses testimony would be privileged as trial preparatory material. See Kansas Gas and Electric Co. (Wolf Creek Nuclear Generating Station, Unit 1), ALAB-327, 3 NRC 408 (1976). Further, since at present no such summaries exist, requiring the Staff to compile data and create such a summary is objectionable. See 4A Moore's Federal Practice, ¶ 33.20 (3). Therefore, throughout this document Part (2) of Subpart D will not be answered. The Staff further notes that Joint Intervenors will have a complete copy of all Staff testimony prior to any hearing.

Interrogatory 29

Explain the present Staff position on Joint Intervenors' contention 10, regarding pressurizer heater design, and state each and every fact on which that position is based.

Response

A. Operation of the pressurizer heaters at Diablo Canyon is not a critical safety function and, therefore, the heaters are not required to be designed to safety grade criteria.

The critical safety functions which must be provided by safety grade systems are identified in Section III.C of Appendix A to 10 C.F.R. 100.

Operation of the pressurizer heaters is not assumed in the safety analyses of design basis accidents for Diablo Canyon. Although operation of the pressurizer heaters is required to maintain primary system pressure for power operation and hot standby, operation of the pressurizer heaters is not required to bring the plant to cold shutdown, which is a safe and stable condition. Tests at the Sequoyah Nuclear Power Plant have demonstrated that the effect of deenergizing the pressurizer heaters would be gradual depressurization of the primary system (100 psig/hour) with no loss of natural circulation.

Failure of the pressurizer heaters to operate would allow the reactor system to gradually depressurize which, in the absence of any corrective operation action, would eventually cause automatic actuation of the ECCS. The capability to provide emergency power to the pressurizer heaters is available at Diablo Canyon to reduce the number of demands for ECCS to operate in accordance with Item II.E.3.1 of NUREG-0737 and Item 2.1.1 of NUREG-0578.

B. 10 C.F.R. 100. Letter dated July 29, 1980, from L.M. Mills, Manager, Nuclear Regulation and Safety, TVA, to A. Schwencer, Division of Licensing, NRC (a copy of which has been provided to the service list as an attachment to "NRC Staff's Response to Governor Edmund G. Brown Jr.'s Second Set of Interrogatories."

C. The Staff and/or independent contractor are not presently engaged in or intend to engage in further research or work which may bear on

the issues covered in the interrogatory other than the normal staff review.

D. (1) Walton L. Jensen is the expert whom the Staff intends to have testify on the subject matter covered in the interrogatory. A copy of his professional qualifications is in Attachment A.

(2) See paragraph prior to Interrogatory 29.

(3) Walton L. Jensen has testifed: (a) on PORV, safety and block valves, natural circulation and small break LOCAs in Metropolitan Edison Co. (Three Mile Island Nuclear Station, Unit No. 1), Docket No. 50-289;

(b) on LOCAs in Commonwealth Edison Co. (Zion Station, Units 1 and 2), Docket Nos. 50-295, 50-304; and

(c) on a steam generator tube rupture in Northern States Power Co. (Prairie Island Nuclear Generating Plant, Units 1 and 2), Docket Nos. 50-282, 50-306.

Interrogatory 30

Does the current position differ from the position of the Staff in any prior proceedings? If so, identify the proceeding(s), explain the prior position, and explain the basis for the change in position.

Response

A. The current position does not differ from the position of the Staff in any prior proceedings.

B. None.

C. The Staff and/or independent contractor are not presently engaged in or intend to engage in further research or work which may bear on the issues covered in the interrogatory other than the normal staff review.

D. (1) Walton L. Jensen is the expert whom the Staff intends to have testify on the subject matter covered in the interrogatory. A copy of his professional qualifications is in Attachment A.

(2) See response to Interrogatory 29.D(2).

(3) See response to Interrogatory 29.D(3).

Interrogatory 31

Identify any officers or employees of, or consultants to, the Staff who dissent from the present Staff position on Joint Intervenors' contention 10. Explain the reasons for which any such person dissents.

Response

A. There are no identified dissenting Staff members.

B. None.

C. The Staff and/or independent contractor are not presently engaged in or intend to engage in further research or work which may bear on the issues covered in the interrogatory other than normal staff review.

D. (1) Walton L. Jensen is the expert whom the Staff intends to have testify on the subject matter covered in the interrogatory. A copy of his professional qualifications is in Attachment A.

(2) See response to Interrogatory 29.D(2).

(3) See response to Interrogatory 29.D(3).

Interrogatory 32

Identify the specific sections and page numbers of the FSAR for Diablo Canyon and the NRC Staff's SER and SER Supplements for Diablo Canyon, which are relied upon in formulating the Staff position on Joint Intervenors' contention 10.

Response

A.

FSAR	SER	SER Supplements
Section 5.5.10	Chapter 15	#6 Section 6.3
Chapter 15	Section 6.3	#14 Section II.E.1.1
Section 6.3	Section 5.5	#14 Section II.E.1.2
		#14 Section II.E.3.1

B. See response to Interrogatory 32.A.

C. The Staff and/or independent contractor are not presently engaged in or intend to engage in further research or work which may bear on the issues covered in the interrogatory other than normal staff review.

D. (1) Walton L. Jensen is the expert whom the Staff intends to have testify on the subject matter covered in the interrogatory. A copy of his professional qualifications is in Attachment A.

(2) See response to Interrogatory 29.D(2).

(3) See response to Interrogatory 29.D(3).

Interrogatory 33

Identify all sections and page numbers of the FSAR, SER, and SER Supplements which contain subject matter pertaining to Joint Intervenors' contention 10.

Response

The Staff notes that this interrogatory is objectionable in that it asks the Staff to compile data which is as readily available to Joint Intervenors as to Staff. The Joint Intervenors can read the FSAR, SER and SER Supplements and find for themselves any portions relevant to their contention. See 4A Moore's Federal Practice, ¶ 33.20(3).

Interrogatory 34

The Staff has recognized that the "maintenance of natural circulation capability is important to safety (and) depends on the maintenance of pressure control . . . (which) is normally achieved through the use of pressurizer heaters." NUREG-0578, p. A-2.

(a) Do you continue to agree with that view?

(b) Explain why pressurizer heaters and their associated controls are not classified as "components important to safety," as discussed in GDC 17 and the Introduction to Appendix A to CFR Part 50.

Response

A. (a) Analyses of various transients and accidents following the accident at TMI-2 indicated that loss of pressure control in a plant having once-through steam generators such as TMI-2 could cause a temporary loss of natural circulation. This effect would be caused by the formation of steam bubbles in the upper part of the hot legs of the coolant loops, "candy canes", in the event that the pressure in the hot legs dropped sufficiently that boiling occurred. Natural circulation was shown to be reestablished if the steam bubble increased in size sufficiently so that a portion entered the steam generators so that steam could be condensed by the auxiliary feedwater that is sprayed on the top of the tube bundle. This effect is the basis for the statement in NUREG-0578.

However, for plants with U-tube steam generators, such as Diablo Canyon, the high points of the coolant loops are the U-bends of the steam generator tubes which are continually covered with secondary coolant supplied by the main or auxiliary feedwater system. Steam formed in the coolant loops of a plant of the Diablo Canyon design would be condensed by the steam generators with no loss of natural circulation. If sufficient steam were present, the mode of natural circulation would change from single-phase natural convection to two-phase boiling condensation. Tests at the LOFT and Setmiscale facilities have demonstrated that loss of natural circulation will not occur at plants equipped with U-tube steam generators in the presence of steam in the coolant loops as long as steam generator cooling remains available. Semiscale results are documented in Report No. EGG-SEMI-5507, "Quick Look Report for Semiscale Mod-2A Test S-NC-2," July 1981. LOFT results are documented in report No. NUREG CR-1570 "Experimental Data Report for LOFT Nuclear Small Break Experiments L3-7," August 1980.

(b) The pressurizer heaters are considered "components important to safety" with respect to their pressure-control function. This pressure-control function does not mean it is necessary to meet safety grade criteria for the reasons summarized below:

(1) The term "important to safety" applies generally to the broad class of structures, systems, and components addressed in the General Design Criteria.

(2) "Safety-grade" structures, systems and components are a sub-class of all those "important to safety."

(3) All structures, systems, and components encompassed by the term "important to safety" (including the "safety-grade" sub-class) are necessary to meet the broad safety goal articulated in Appendix A to 10 CFR Part 50 of the regulations (i.e., provide reasonable assurance that a facility can be operated without undue risk to the health and safety of the public).

(4) Only "safety-grade" structures, systems and components are required for the critical accident prevention, safe shutdown, and accident consequence mitigation safety functions identified in Section III.C of Appendix A to 10 CFR Part 100.

GDC 17 requires that "components important to safety" be provided with reliable power supplies to maintain the vital functions of protecting the core, reactor coolant pressure boundary and containment in the event of anticipated operational occurrences and postulated accidents. Operation of the pressurizer heaters is not required to provide one of these vital functions.

B. 10 C.F.R. 100, 10 C.F.R. 50, NUREG-0578, NUREG-CR-1570, Report No. EGG-SEMI-5507, Testimony of J.H. Conran in response to UCS Contention 14 TMI-1 Restart Hearing.

C. The Staff and/or independent contractor are not presently engaged in or intend to engage in further research or work which may bear on the issues covered in the interrogatory other than normal staff review.

D. (1) Walton L. Jensen is the expert whom the Staff intends to have testify on the subject matter covered in the interrogatory. A copy of his professional qualifications is in Attachment A.

(2) See response to Interrogatory 29.D(2).

(3) See response to Interrogatory 29.D(3).

Interrogatory 35

Explain in detail whether and in what manner the following design criteria would be met with respect to the pressurizer heater and its associated controls.

(a) GDC 22 (diversity)

(b) GDC 2 and 4 (seismic and environmental qualification)

(c) GDC 10 (automatic initiation)

(d) GDC 3 and 22 (separation and independence)

Response

A. The referenced design criteria have been met as follows:

(a) GDC 22 addresses protection systems. Because pressurizer heaters are not considered part of the protection system, there is no diversity requirement.

(b) Because pressurizer heaters do not perform a critical safety function, there is no requirement for seismic and environmental qualifications.

(c) One purpose of the pressurizer heaters is to maintain reactor pressure so that a hot standby condition can be maintained for extended periods of time. Failure to turn on the pressurizer heaters would initiate a reactor cooldown. Fuel design limits would not be exceeded. Thus, automatic initiation is not a requirement.

(d) One purpose of the pressurizer heaters is to maintain reactor pressure so that a hot standby condition can be maintained for extended periods of time. Because credit has not been taken in the plant's fire hazards analysis for extended operation at hot standby using pressurizer heaters and because pressurizer heaters are not considered part of the protection system, protection from the effects of fires, separation and independence is not a requirement.

B. Diablo Canyon SER, Supplement No. 14.

C. The Staff and/or independent contractor are not presently engaged in or intend to engage in further research or work which may bear on the issues covered in the interrogatory other than normal staff review.

D. (1) John L. Knox is the expert whom the Staff intends to have testify on the subject matter covered in the interrogatory. A copy of his professional qualifications is in Attachment B.

(2) See paragraph prior to Interrogatory 29.

(3) John L. Knox has testified on equipment qualifications in Pacific Gas and Electric Co. (Diablo Canyon, Units 1 and 2), Docket Nos. 50-275 and 50-323.

Interrogatory 36

Specify precisely under what conditions the pressurizer heaters will be relied upon at Diablo to:

(a) regulate and/or control pressure;

(b) initiate and/or maintain natural circulation;

(c) mitigate the consequences of inadequate core cooling;

(d) stabilize the reactor in post-accident conditions;

(e) any other functions performed by the pressurized heaters.

Response

A. (a) Pressurizer heaters are required to regulate and control reactor system pressure during power operation and to maintain hot standby. Reactor system pressure could temporarily be controlled and regulated during these modes of operation by controlling charging and letdown flow without operation of the pressurizer heaters but the pressurizer water level would eventually get too high from the effect of pressurizer heat loss and the reactor would have to be shutdown. An automatic reactor trip is provided for high pressurizer level.

(b) Not Required.

(c) Not Required.

(d) Not Required.

(e) None.

B. None.

C. The Staff and/or independent contractor are not presently engaged in or intend to engage in further research or work which may bear on the issues covered in the interrogatory other than normal staff review.

D. (1) Walton L. Jensen is the expert whom the Staff intends to have testify on the subject matter covered in the interrogatory. A copy of his professional qualifications is in Attachment A.

(2) See response to Interrogatory 29.D(2).

(3) See response to Interrogatory 29.D(3).

Interrogatory 37

Assuming inoperability of the pressurizer heaters, specify in detail each and every means, system, and/or component available at Diablo Canyon to perform the functions listed in Interrogatory No. 36 under the conditions described in your response to that interrogatory. State each and every fact upon which you base your contention that such other means, systems, and/or components can adequately perform the functions listed.

Response

A. The response to interrogatory 36a states that the reactor would have to be shutdown following a complete loss of the pressurizer heater control function. Safe shutdown could be accomplished utilizing the Emergency Core Cooling System, the Auxiliary Feedwater System and the Residual Heat Removal System. These systems are designed to safety-grade criteria at Diablo Canyon.

B. None.

C. The Staff and/or independent contractor are not presently engaged in or intend to engage in further research or work which may bear on the issues covered in the interrogatory other than normal staff review.

D. (1) Walton L. Jensen is the expert whom the Staff intends to have testify on the subject matter covered in the interrogatory. A copy of his professional qualifications is in Attachment B.

(2) See response to Interrogatory 35.D(2).

(3) See response to Interrogatory 35.D(3).

Interrogatory 38

Specify precisely each and every way in which the pressurizer heaters and associated controls at Diablo Canyon do not meet the safety-grade design criteria set forth in Appendix A to 10 C.F.R. Part 50, and list each design criteria not complied with.

Response

A. The pressurizer heaters and associated controls meet all applicable design criteria set forth in Appendix A to 10 CFR Part 50.

B. Diablo Canyon SER, Supplement No. 14.

C. The Staff and/or independent contractor are not presently engaged in or intend to engage in further research or work which may bear on the issues covered in the interrogatory other than normal staff review.

D. (1) John L. Knox is the expert whom the Staff intends to have testify on the subject matter covered in the interrogatory. A copy of his professional qualifications is in Attachment B.

(2) See response to Interrogatory 35.D(2).

(3) See response to Interrogatory 35.D(3).

Interrogatory 39

Describe in detail what changes, if any, have been made in the design, construction, installation, or operation of the pressurizer heaters and associated controls at Diablo Canyon since the TMI-2 accident in March 1979. With respect to any changes or alterations, specify how, if at all, they are expected or intended to enhance the reliability of the components and/or safe operations of the plant, and state each and every fact upon which your response is based.

Response

A. Two manual transfer switches with associated safety-grade protective devices have been added to connect the pressurizer heaters to onsite standby power sources. The switches give the plant operator the added option of using onsite standby power to supply the pressurizer heaters when there is a loss of offsite power. The added option enhances safe operation of the plant.

B. Diablo Canyon SER, Su ..ement No. 14.

C. The Staff and/or independ..nt contractor are not presently engaged in or intend to en.. in further research or work which may bear on the issues covered in the interrogatory other than normal staff review.

D. (1) John L. Knox is the expert whom the Staff intends to have testify on the subject matter covered in the interrogatory. A copy of his professional qualifications is in Attachment B.

(2) See response to Interrogatory 35.D(2).

(3) See response to interrogatory 35.D(3).

Interrogatory 40

Describe in detail what you consider to be the implications, if any, of the experience at TMI-2 in March 1979 with respect to the design, installation, maintenance, and/or operation of the pressurizer heaters and associated controls at Diablo Canyon. State each fact upon which your response is based.

Response

A. The implications of the experience at TMI-2 in March 1979 with respect to pressurizer heaters are discussed in NUREG-0578 "TMI-2 Lessons Learned Task Force Status Report" and Short-Term Recommendations," pages 6 and 7. A prolonged loss of pressurizer heater power without mitigating action by the operator would lead to

ECCS actuation. The task force expressed concern for the frequency with which some safety systems are called on to function and recommended that the pressurizer heaters be provided with emergency power. This recommendation has been implemented at Diablo Canyon.

B. NUREG-0578.

C. The Staff and/or independent contractor are not presently engaged in or intend to engage in further research or work which may bear on the issues covered in the interrogatory other than normal staff review.

D. (1) Walton L. Jensen is the expert whom the Staff intends to have testify on the subject matter covered in the interrogatory. A copy of his professional qualifications is in Attachment A.

(2) See response to Interrogatory 29.D(2).

(3) See response to Interrogatory 29.D(3).

Interrogatory 41

With respect to the pressurizer heaters and associated controls at Diablo Canyon, specify in detail:

(a) their precise location in Units 1 and 2;

(b) the precise specifications to which they were ordered and/or designed and any differences between the design specifications on the one hand and the heaters and associated controls as initiated on the other;

(c) their manufacturer;

(d) the precise location of all seismic-related supports, hangers, snubbers, etc., which are attached to, relate to, or in any way could affect operation of the heaters, associated controls, and/or associated cables, electrical or otherwise;

(e) the precise polar position and elevation and coordinate location with respect to the center of the containment at which the cables for the pressurizer heaters cross the annulus in Diablo Canyon, Unit 1.

Response

A. (a)-(e) With respect to the pressurizer heaters and associated controls at Diablo Canyon, precise locations, specifications, and manufacturer are beyond the scope of the staff's audit review.

B. Not applicable.

C. The Staff and/or independent contractor are not presently engaged in or intend to engage in further research or work which may bear on the issues covered in the interrogatory other than normal staff review.

D. (1) John L. Knox is the expert whom the Staff intends to have testify on the subject matter covered in the interrogatory. A copy of his professional qualifications is in Attachment B.

(2) See response to Interrogatory 35.D(2).

(3) See response to Interrogatory 35.D(3).

Interrogatory 42

List and describe in detail all analyses and tests conducted by you, your agents, or your consultants with respect to the pressurizer heaters and associated controls. Specify:

(a) the person or entity conducting the analyses or tests;

(b) the purpose(s) of the analyses or tests;

(c) the range of test conditions or conditions assumed in the analyses;

(d) the specification of the components tested or analyzed;

(e) the results of the tests or analyses;

(f) any other tests or analyses planned to be conducted prior to full power operation.

Response

A. (a)-(f) There were no analyses or tests conducted by the NRC, NRC's agents or consultants with respect to the pressurizer heaters and associated controls.

B. Not applicable.

C. The Staff and/or independent contractor are not presently engaged in or intend to engage in further research or work which may bear on the issues covered in the interrogatory other than normal staff review.

D. (1) John L. Knox is the expert whom the Staff intends to have testify on the subject matter covered in the interrogatory. A copy of his professional qualifications is in Attachment B.

(2) See response to Interrogatory 35.D(2).

(3) See response to Interrogatory 35.D(3).

Interrogatory 43

State whether you contend that the pressurizer heaters and associated controls at Diablo Canyon should be classified as important to safety and required to meet all applicable safety-grade design criteria, and state each and every fact upon which your response is based.

Response

A. The pressurizer heaters and associated controls are "components important to safety" at Diablo Canyon but are not required to meet safety grade design criteria for the reasons stated in the response to Interrogatory 34b above.

B. 10 C.F.R. 100, 10 C.F.R. 50, NUREG-0578, NUREG-CR-1570, Report No. EGG-SEMI-5507, Testimony of J.H. Conran in response to USC Contention 14 TMI-1 Restart Hearing.

C. The Staff and/or independent contractor are not presently engaged in or intend to engage in further research or work which may bear on the issues covered in the interrogatory other than normal staff review.

D. (1) Walton L. Jensen is the expert whom the Staff intends to have testify on the subject matter covered in the interrogatory. A copy of his professional qualifications is in Attachment A.

(2) See response to Interrogatory 29.D(2).

(3) See response to Interrogatory 29.D(3).

Interrogatory 44

Describe what modifications would have to be made in the Diablo Canyon pressurizer heaters and associated controls to bring them into compliance with all applicable safety-grade design criteria. Estimate the minimum time period necessary to make those modifications, and state each and every fact upon which your estimate is based.

Response

A. There are no modifications that would have to be made in the Diablo Canyon pressurizer heaters and associated controls because the pressurizer heaters and associated controls are not required to meet safety grade design criteria for the reasons stated in the response to Interrogatory 34A(b) above.

B. Diablo Canyon SER, Supplement No. 14.

C. The Staff and/or independent contractor are not presently engaged in or intend to engage in further research or work which may bear on the issues covered in the interrogatory other than normal staff review.

D. (1) Walton L. Jensen and John Knox are the experts whom the Staff intends to have testify on the subject matter covered in the interrogatory. A copy of their professional qualifications are in Attachments A and B respectively.

(2) See response to Interrogatory 29.D(2) and 35.D.(2).

(3) See response to Interrogatory 29.D(3) and 35.D.(3).

Interrogatory 45

Specify precisely (a) which Emergency Operating Procedures for Diablo Canyon include the use of pressurizer heaters and (b) which require that the heaters be switched to the on-site power supplies.

Response

A. (a) The following emergency procedures recognize the existence of the pressurizer heaters and provide guidance to the operator in the use of pressurizer heaters:

OP-0	Reactor Trip with Safety Injection
OP-1	Loss of Coolant Accident
OP-2	Loss of Secondary Coolant
OP-3A	Steam Gen Tube Failure
OP-3B	Minor Steam Gen Tube Failure
OP-4	Loss of Electrical Power
OP-5	Reactor Trip without Safety Injection
OP-7	Loss of Condenser Vacuum
OP-8	Control Room Inaccessibility
OP-9	Loss of Reactor Coolant Pump
OP-10	Loss of Auxiliary Salt Water
OP-11	Loss of Component Cooling Water
OP-12A	Failure of a Control Bank to Move to Auto
OP-12C	Cont Insertion of a Control Rod Bank

OP-12D	Control Rod Pos Indication Sys Malfunc
OP-12E	Control Rod Misalignment
OP-12F	Dropped Control Rod
OP-13	Malfunction of Reactor Press Control Sys
OP-14	High Activity in Reactor Coolant
OP-15	Loss of Feedwater
OP-16	Nuclear Instrumentation Malfunctions
OP-18	Charging or Letdown Line Failure
OP-19	Malfunction of Reactor Makeup Control
OP-20	Excessive Reactor Coolant System Leakage
OP-21	Loss of a Coolant RTD
OP-23	Natural Circulation of Reactor Coolant
OP-24	Loss of Containment Integrity
OP-26	Excessive Feedwater Flow
OP-28	Startup of an Inactive Reactor Coolant Loop
OP-29	Excessive Load Increase
OP-30	Inadvertent Load Fuel Assembly Improper Position
OP-31	System Under Frequency
OP-33	Loss of Instrument Air
OP-35	Loss of Vital or Non-Vital Instr AC Sys
OP-36	Turbine Trip
OP-37	Loss of Protection System Channel
OP-40	Accidental Depressurization of MS System
OP-44	Gaseous voids in the RCS.

(b) Emergency operating procedure OP-4 requires that the heaters be switched to onsite power supplies in accordance with operating procedure A-4.

It should be noted that although the pressurizer heaters would be expected to be normally available in a number of anticipated transients and accidents (as they are during routine daily operations), they are not required to protect the reactor. As discussed in Chapter 15 of the FSAR, reliance is placed on the reactor protection system and the engineered safety features which are designed to safety grade criteria. It must be recognized that emergency procedures are written to present guidance on all options available to the operator for coping with a plant transient or accident. To do otherwise would be an unwise limitation of design and operator capability during each event. The licensing arena (FSAR Chapter 15) in which conservative assumptions on the unavailability of non-safety grade components are typically imposed should not be confused with an actual transient or accident during which the operator would be expected to make maximum use of ALL available systems, whether they are safety grade or not.

B. Diablo Canyon Power Plant Units 1 and 2 Emergency Procedures, Volume 3A.

C. The Staff and/or independent contractor are not presently engaged in or intend to engage in further research or work which may bear on

the issues covered in the interrogatory other than normal staff review.

D. (1) Walton L. Jensen is the expert whom the Staff intends to have testify on the subject matter covered in the interrogatory. A copy of his professional qualifications is in Attachment A.

(2) See response to Interrogatory 29.D(2).

(3) See response to Interrogatory 29.D(3).

Interrogatory 46

Explain the present Staff position on Joint Intervenors contention 12, regarding valve design, and state each and every fact on which that position is based.

Response

A. Proper operation of the PORV's and the associated block valves is not a critical safety function and, therefore, the PORV's and block valves are not required to be designed to safety grade criteria. The critical safety functions which must be provided by safety grade systems are identified in Section III.C of Appendix A to 10 C.F.R. 100.

The function of the PORV is to prevent unnecessary opening of the pressurizer safety valves and to provide a backup means of depressurization and overpressure protection. The function of the block valves is to permit isolation of a leaking or failed open PORV. Proper operation of the PORV and block valve is not required to mitigate the consequences of any design bas accident. Failure of a PORV and block valve to function can cause the equivalent of a small-break LOCA, but if the failure occurred in conjunction with a LOCA, the consequences would not be significantly altered. An unisolated stuck-open PORV would not result in core damage (see response to interrogatory #52).

Emergency power has been provided to two of the three PORV's and to the three block valves to reduce the number of challenges to safety valves and ECCS during operation in accordance with Item II.G.1 of NUREG-0737 and Item 2.1.1 of NUREG-0578.

B. 10 C.F.R. 100, NUREG-0737.

C. The Staff and/or independent contractor are not presently engaged in or intend to engage in further research or work which may bear on the issues covered in the interrogatory other than normal staff review.

D. (1) Walton L. Jensen is the expert whom the Staff intends to have testify on the subject matter covered in the interrogatory. A copy of his professional qualifications is in Attachment A.

(2) See response to Interrogatory 29.D(2).

(3) See response to Interrogatory 29.D(3).

Interrogatory 47

Does the current position differ from the position of the Staff in any prior proceedings? If so, identify the proceeding(s), explain the prior position, and explain the basis for the change in position.

Response

A. The current position does not differ from the position of the Staff in any prior proceedings.

B. None.

C. The Staff and/or independent contractor are not presently engaged in or intend to engage in further research or work which may bear on the issues covered in the interrogatory other than normal staff review.

D. (1) Walton L. Jensen is the expert whom the Staff intends to have testify on the subject matter covered in the interrogatory. A copy of his professional qualifications is in Attachment A.

(2) See response to Interrogatory 29.D(2).

(3) See response to Interrogatory 29.D(3).

Interrogatory 48

Identify any officers or employees of, or consultants to, the Staff who dissent from the present Staff position on Joint Intervenors' contention 12. Explain the reasons for which any such person dissents.

Response

A. There are no identified dissenting Staff members.

B. None.

C. The Staff and/or independent contractor are not presently engaged in or intend to engage in further research or work which may bear on the issues covered in the interrogatory other than normal staff review.

D. (1) Walton L. Jensen is the expert whom the Staff intends to have testify on the subject matter covered in the interrogatory. A copy of his professional qualifications is in Attachment A.

(2) See response to Interrogatory 29.D(2).

(3) See response to Interrogatory 29.D(3).

Interrogatory 49

Identify the specific sections and page numbers of the FSAR for Diablo Canyon and the NRC Staff's SER and SER Supplements for Diablo Canyon, which are relied upon in formulating the Staff position on Joint Intervenors' contention 12.

Response

A.

FSAR	SER	SER Supplements
Section 5.2.2	Section 5.2.2	#6 Section 5.2.2
Section 5.5.10	Section 6.3	#6 Section 6.3
Section 5.5.13	Chapter 15	#10 Section II.G.1
	Section 5.5	#13 Chapter 15
		#14 Section II.E.1.1
		#14 Section II.E.1.2

B. See response to Interrogatory 49.A.

C. The Staff and/or independent contractor are not presently engaged in or intend to engage in further research or work which may bear on the issues covered in the interrogatory other than normal staff review.

D. (1) Walton L. Jensen is the expert whom the Staff intends to have testify on the subject matter covered in the interrogatory. A copy of his professional qualifications is in Attachment A.

(2) See response to Interrogatory 29.D(2).

(3) See response to Interrogatory 29.D(3).

Interrogatory 50

Identify all sections and page numbers of the FSAR, SER, and SER Supplements which contain matter pertaining to Joint Intervenors' contention 12.

Response

A. The Staff notes that this interrogatory is objectionable in that it asks the Staff to compile data which is as readily available to Joint Intervenors as to Staff. The Joint Intervenors can read the FSAR, SER and SER Supplements and find for themselves any portions relevant to their contention. See 4A Moore's Federal Practice, ¶ 33.20(3). While preserving said objection, the Staff provides the following responsive information:

Section	Page	Description of Technical Content
- SER 3.9.1	3-21	Discusses preoperational dynamic effects test program.
- SER 5.2.1	5-1 & 5-2	Discusses design of reactor coolant system components
- SER 5.2.2	5-3	Discusses reactor coolant system overpressurization protection.

- SER 17.4	17-8	Discusses compliance with Appendix B of 10 CFR Part 50.
- SER Suppl. 9.3.9.3.7	3-9 &	Discusses Hosgri PORV seismic
	3-10	qualification.
- SER Suppl. 10	II.D.1 -	Discusses relief and safety
II.D.2, II.D.5	II.D.3	valve testing and Position Indication.
- SER Suppl. 14 II.D.1	3-12 - 3-14	Discusses performance testing of relief, safety and block valves.

B. See response to Interrogatory 50.A.

C. The Staff and/or independent contractor are not presently engaged in or intend to engage in further research or work which may bear on the issues covered in the interrogatory other than normal staff review.

D. (1) Frank C. Cherny is the expert whom the Staff intends to have testify on the subject matter covered in the interrogatory. A copy of his professional qualifications is in Attachment C.

(2) See paragraph prior to Interrogatory 29.

(3) Frank C. Cherny has testified on relief, safety and block valves in Pacific Gas and Electric Co. (Diablo Canyon, Units 1 and 2), Docket Nos. 50-275 and 50-323.

Interrogatory 51

Does the Staff agree that proper operation of PORVs, associated block valves and the instruments and controls for these valves is essential to mitigate the consequences of accidents? Explain your response fully.

Response

A. No, proper operation of the PORV's and block valves is not required to mitigate any of the design basis events chapter 15 of the FSAR.

B. FSAR, Chapter 15.

C. The Staff and/or independent contractor are not presently engaged in or intend to engage in further research or work which may bear on the issues covered in the interrogatory other than normal staff review.

D. (1) Walton L. Jensen is the expert whom the Staff intends to have testify on the subject matter covered in the interrogatory. A copy of his professional qualifications is in Attachment A.

(2) See response to Interrogatory 29.D(2).

(3) See response to Interrogatory 29.D(3).

Interrogatory 52

Does the Applicant agree that failures of these valves, instruments and controls can cause or aggravate a LOCA? Explain your response fully.

Response

A. This interrogatory was addressed solely to the applicant. However, the NRC staff believes that the small-break analyses and procedures described below indicate the applicant's position on this issue.

Analyses of stuck open PORVs were performed in WCAP-9600 for the purpose of providing guidance to plant operators and as a basis for supporting small-break LOCA procedures. The emergency procedures for loss-of-coolant accidents for Diablo Canyon recognize that a stuck open PORV can produce the symptoms of a LOCA and instruct the operator to close the block valve.

WCAP-9600 also provides an analyses of a small-break LOCA in coincidence with a complete loss of feedwater. This sequence of events was shown to produce inadequate core cooling. During the course of the event, the PORVs were assumed to be opened by the operator. Since opening of the PORV was shown to enhance core cooling, therefore, the procedures for inadequate core cooling resulting from a small break LOCA at Diablo Canyon instruct the operator to open the PORVs. (OP-1 Appendix F).

B. WCAP-9600, Volume 3. (WCAP-9601, Volume 3, a non-proprietary version of this document is available at the NRC Public Document Room.)

C. The Staff and/or independent contractor are not presently engaged in or intend to engage in further research or work which may bear on the issues covered in the interrogatory other than normal staff review.

D. (1) Walton L. Jensen is the expert whom the Staff intends to have testify on the subject matter covered in the interrogatory. A copy of his professional qualifications is in Attachment A.

(2) See response to Interrogatory 29.D(2).

(3) See response to Interrogatory 29.D(3).

Interrogatory 53

Provide the justification for the failure to classify power operated relief valves (PORVs) and associated block valves and their respective instruments and controls as "components important to safety," requiring compliance with safety-grade design criteria.

Response

A. The PORVs and block valves are considered "components important to safety" with respect to their pressure control function. They are not required to meet safety grade criteria for the reasons summarized below:

(1) The term "important to safety" applies generally to the broad class of structures, systems, and components addressed in the General Design Criteria.

(2) "Safety-grade" structures, systems and components are a sub-class of all those "important to safety".

(3) All structures, systems, and components encompassed by the term "important to safety" (including the "safety-grade" sub-class) are necessary to meet the broad safety goal articulated in Appendix A to 10 C.F.R. Part 50 of the regulations (i.e., provide reasonable assurance that a facility can be operated without undue risk to the health and safety of the public).

(4) Only "safety-grade" structures, systems and components are required for the critical accident prevention, safe shutdown, and accident consequences mitigation safety function identified in Section III.C of Appendix A to 10 C.F.R. Part 100.

B. Testimony of J.A. Conran in response to UCS Contention 14 TMI-1 Restart Hearing.

C. The Staff and/or independent contractor are not presently engaged in or intend to engage in further research or work which may bear on the issues covered in the interrogatory other than normal staff review.

D. (1) Walton L. Jensen is the expert whom the Staff intends to have testify on the subject matter covered in the interrogatory. A copy of his professional qualifications is in Attachment A.

(2) See response to Interrogatory 29.D(2).

(3) See response to Interrogatory 29.D(3).

Interrogatory 54

Explain how the motive and control components of the PORVs and their associated block valves and the vital instruments shall be supplied by the on-site emergency power source when offsite power is not available without degrading the capacity, capability and reliability of emergency power in violation of GDC 17.

Response

A. The PORVs and block valve loads are connected to the Class IE power system. The load is small in comparison to the overall capacity of the system. The load is connected through overload and short circuit protective devices and a second short circuit protective device is to be added in series for the purpose of protecting the electrical penetrations from damaging faults. The protective devices and circuits with associated raceways that connect the

PORVs and block valves to the Class IE power system meet safety-grade requirements.

B. Diablo Canyon SER, Supplement No. 10.

C. The Staff and/or independent contractor are not presently engaged in or intend to engage in further research or work which may bear on the issues covered in the interrogatory other than normal staff review.

D. (1) John L. Knox is the expert whom the Staff intends to have testify on the subject matter covered in the interrogatory. A copy of his professional qualifications is in Attachment B.

(2) See response to Interrogatory 35.D(2).

(3) See response to Interrogatory 35.D(3).

Interrogatory 55

How have the devices through which motive and control power components for the PORVs and their associated block valves are connected to emergency buses been qualified in accordance with safety-grade requirements?

Response

A. The protective devices and circuits that connect the PORVs and block valves to the Class IE power system are qualified by test and analysis in accordance with safety grade requirements.

B. Diablo Canyon SER, Supplement No. 10.

C. The Staff and/or independent contractor are not presently engaged in or intend to engage in further research or work which may bear on

the issues covered in the interrogatory other than normal staff review.

D. (1) John L. Knox is the expert whom the Staff intends to have testify on the subject matter covered in the interrogatory. A copy of his professional qualifications is in Attachment B.

(2) See response to Interrogatory 35.D(2).

(3) See response to Interrogatory 35.D(3).

Interrogatory 56

With respect to the valves, instruments, and controls cited in contention 12, list each and every General Design Criterion in Appendix A to 10 C.F.R. Part 50 which is not complied with, and describe precisely in what respects those valves, instruments, and controls do not comply.

Response

A. The Diablo Canyon PORV's, PORV block valves, instruments and controls cited in contention 12 are in full compliance with all applicable General Design Criteria in Appendix A to 10 C.F.R. Part 50.

B. Diablo Canyon SER, Supplement Nos. 10 and 14.

C. The Staff and/or independent contractor are not presently engaged in or intend to engage in further research or work which may bear on the issues covered in the interrogatory other than normal staff review.

D. (1) John L. Knox and Frank C. Cherny are the experts whom the Staff intends to have testify on the subject matter covered in the interrogatory. A copy of his professional qualifications is in Attachment B & C respectively.

(2) See response to Interrogatory 35.D(2) and 50.D(2).

(3) See response to Interrogatory 35.D(3) and 50.D(3).

Interrogatory 57

Describe precisely each and every function of the PORVs at Diablo Canyon, and for each such function, specify in detail the operating conditions in which the PORVs would be relied upon to perform that function.

Response

A. The function of the PORVs at Diablo Canyon is to open at the set pressure of 2350 psig and relieve pressurizer steam so as to preclude the necessity of the safety valves from being opened for mild transients. The set pressure for the safety valves is 2485 psig. The PORVs may also be manually opened by the operator at any pressure below their setpoint to provide a backup means of pressure control in accordance with the operating procedures. Manual opening of the PORVs would cause them to relieve steam at the pressure of the reactor system. Reliance is placed on the Engineered Safety Features to mitigate design basis events rather than on the PORV.

B. FSAR, Chapter 5.

C. The Staff and/or independent contractor are not presently engaged in or intend to engage in further research or work which may bear on the issues covered in the interrogatory other than normal staff review.

D. (1) Walton L. Jensen is the expert whom the Staff intends to have testify on the subject matter covered in the interrogatory. A copy of his professional qualifications is in Attachment A.

(2) See response to Interrogatory 29.D(2).

(3) See response to Interrogatory 29.D(3).

Interrogatory 58

Describe precisely each and every function of the block valves at Diablo Canyon, and for each such function, specify in detail the operating conditions in which the block valves would be relied upon to perform that function.

Response

A. The function of the block valve is to isolate a leaking or failed open PORV. The PORV might leak or fail open during any operating condition. Analyses of a stuck open PORV performed by Westinghouse and described in WCAP-9600 indicate that they would pass steam initially until the reactor system pressure decreased to about 1300 psia. Then they would pass a two-phase mixture. The reactor system pressure was shown to stabilize at about 1060 psia with an alternate vapor and liquid flow out the stuck open valve. The operator might close the block valve at any time to isolate the break. The analyses demonstrated that adequate core cooling occurred without operator action. Reliance is placed on the Engineered Safety Features to mitigate the event, rather than manual closure of the block valve.

B. WCAP-9600, Volume 3. See Response 52.B.

C. The Staff and/or independent contractor are not presently engaged in or intend to engage in further research or work which may bear on the issues covered in the interrogatory other than normal staff review.

D. (1) Walton L. Jensen is the expert whom the Staff intends to have testify on the subject matter covered in the interrogatory. A copy of his professional qualifications is in Attachment A.

(2) See response to Interrogatory 29.D(2).

(3) See response to Interrogatory 29.D(3).

Interrogatory 59

Specify precisely which Emergency Operating Procedures for Diablo Canyon include the use of (a) PORVs and (b) block valves.

Response

A. (a) The following emergency procedures recognize the existence of the PORVs and provide guidance to the operator on the use of the PORVs:

OP-0	Reactor trip with Safety Injection
OP-1	Loss of Coolant Accident
OP-3A	Steam Generator Tube Failure
OP-3B	Minor Steam Generator Tube Failure
OP-22	Emergency Shutdown

The PORVs may automatically open on high pressure for certain anticipated transients. The events include the following:

Loss of electric power, reactor trip, loss of condenser vacuum, reactor pressure control malfunction, loss of feedwater, reactor coolant pump locked rotor and turbine trip. Reliance for over pressure protection is placed on the safety valves, not the PORVs.

(b) The following procedures recognize the existence of the PORV block valves and provide guidance to the operator on the use of the PORV block valves to isolate leaking or failed to open PORVs:

OP-1	Loss of Coolant Accident
OP-3A	Steam Generator Tube Rupture
OP-3B	Minor Steam Generator Tube Leaks
OP-4	Loss of Electric Power
OP-7	Reactor Trip without Safety Injection
OP-13	Malfunction of Reactor Pressure Control System
OP-38	Anticipated Transient Without Trip
OP-39	RCP Locked Rotor

Emergency Procedure OP-22 "Emergency Shutdown" describes use of the block valve to control reactor system pressure in the event of a failure in the safety grade emergency boration system.

It should be noted that although operation of the PORVs and block valves would be expected to be normally available in a number of transients and accidents, these components are not required to protect the reactor. As discussed in Chapter 15 of the FSAR, reliance is placed on the Reactor Protection System and the Engineered Safety Features, which are designed to Safety Grade criteria. It must be recognized that emergency procedures are written to present guidance on all options available to the operator for coping with a plant transient or accident. To do otherwise would be an unwise limitation of design and operator capability during such events. The Licensing arena (FSAR Chapter 15) in which conservative assumptions on the unavailability of non-safety grade components are typically imposed should not be confused with an actual transient or accident during which the operator would be expected to make maximum use of ALL available systems, whether they are safety grade or not.

B. Diablo Canyon Power Plant Units 1 and 2 Emergency Procedures, Volume 3A.

C. The Staff and/or independent contractor are not presently engaged in or intend to engage in further research or work which may bear on the issues covered in the interrogatory other than normal staff review.

D. (1) Walton L. Jensen is the expert whom the Staff intends to have testify on the subject matter covered in the interrogatory. A copy of his professional qualifications is in Attachment A.

(2) See response to Interrogatory 29.D(2).

(3) See response to Interrogatory 29.D(3).

Interrogatory 60

Describe in detail what modifications would have to be made in the PORVs, block valves, instruments, and controls referred to in contention 12 to bring them into compliance with all applicable safety-grade design criteria. Estimate the minimum time period necessary to make those modidifications, and state each and every fact upon which your estimate is based.

Response

A.-D. See response to Interrogatory 56.

Interrogatory 61

Describe in detail the current status of the EPRI valve performance testing program. In your response, state:

(a) when the relief and safety valve testing will be completed;

(b) under what conditions (e.g., transition flow, full water flow, saturated steam, etc.) have the relief and safety valves been tested to date;

(c) whether any of the relief and safety valves tested have failed, suffered galling, or been in any way damaged during the testing, and, if so, describe in detail the circumstances of such occurrences;

(d) why the relief and safety valve testing program completion date has been delayed and when the program is now scheduled to be completed;

(e) whether an EPRI block valve testing program is planned and, if so, when it will be completed;

(f) other than the block valve failures discussed at the Diablo Canyon low power test hearing in May 1981, whether any of the block valves tested have failed, suffered galling, or been in any way damaged during the testing, and, if so, describe in detail the circumstances of such occurrences;

(g) whether PG and E has submitted to the NRC a correlation or other evidence to substantiate that the valves tested in the EPRI program demonstrate the functionability of the relief and safety valves installed at Diablo Canyon, and, if so, describe that correlation or other evidence in detail;

(h) to what extent, if at all, the control circuitry, piping, and supports associated with the Diablo Canyon relief and safety valves have been qualified, and, if so, describe precisely how they have been qualified and the results of any related tests or analyses;

(i) when the "correlation" referred to in subpart (g) of this interrogatory is expected to be received by the NRC.

Response

A. (a) EPRI testing of the relief valve representative of those on the Diablo Canyon, the Masoneilan 20,000 series valve, has been completed. Testing of the safety valve, Crosby HB-BP-86 size 6M6 with loop seal internals materials, and inlet piping installation representative of the Diablo Canyon insallation is scheduled to begin in early November. It is expected that about two weeks will be required to complete the tests.

(b) Both the relief valves and safety valves are being tested on steam, solid water, and steam to water transition at normal reactor operating temperature and pressures or slightly higher. In addition, the relief valves are being tested on water at lower pressures and temperatures representative of fluid conditions the valves could be exposed to during reactor startup and shutdown when the valve is used to provide "low temperature" overpressure protection for the reactor vessel during those modes of plant operation.

(c) The Masoneilan 20,000 relief valve passed all test screening criteria for all of the EPRI tests. Stated briefly, the screening criteria applicable for relief valves are that the valve open and close on demand and not experience any damage that would adversely affect valve opening or closing capability.

Based upon preliminary information that has been made available to the staff, it is our understanding that near the end of the last series of tests some degradation was noted in one gasket that is exposed to the relieving fluid environment. It was reported that the cage to body gasket had "washed out" during testing. It is our understanding that degradation of this gasket has no effect on valve open and closure capability but could result in a small amount of leakage from the valve in to the discharge pipe when the valve is in the closed position. The amount of such leakage is insignificant in terms of safe operation of the plant.

(d) The primary reason for the delay in completion of the EPRI program is related to the safety valve portion of the program being performed at Combustion Engineering. The testing of the 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, has been devoted to this part of the program, delays in construction and shakedown testing resulted in a significant delay in the safety valve test schedule. Additionally, test results from the first two safety valves tested indicated a need to obtain additional information regarding the effects of inlet piping configurations and adjusting ring settings on safety valve operation. In order to obtain this information it has been necessary to expand the test matrix.

Based on estimates in mid-July that completion of safety valve testing would take four to eight months longer than originally estimated, the NRC in September approved the following changes to NUREG-0737 completion dates, for PWR's only, completion of valve testing was changed from July 1, 1981 to April 1, 1982. Plant specific submittals confirming adequacy of safety and relief valves based on licensee/applicant preliminary review of generic test program results was changed from July 1, 1981 to April 1, 1982. Plant specific reports for safety and relief valve qualification was changed from October 1, 1981 to July 1, 1982. Plant specific report submittals for piping and support evaluations were changed from January 1, 1982 to July 1, 1982.

It should be noted the staff recommendation to the Commission to extend the NUREG dates is based on the fact that the testing to date has not uncovered problems with safety or relief valves which are considered significant to the safety of operating plants. Also based upon the staff's continuous monitoring of the EPRI program, the staff has concluded that the program represents a fully responsive effort to meet Commission requirements and that the additional testing being conducted will provide needed information to assure that the technical requirements of item II.D.1 of NUREG-0737 will be fully met.

(e) At this time, there are no plans for EPRI to perform any further block valve testing. At a meeting in July with the NRC Staff, the PWR utilities Owners Group presented its conclusion that it need not commit to any PORV block valve testing beyond that already completed by EPRI for the following reasons: the isolation of a stuck-open PORV is not required to safely shut down a plant, as shown by PWR NSSS vendor analyses; post TMI plant procedures address means for dealing with stuck-open PORV's; EPRI's testing of PORV's has demonstrated that PORVs perform well over a wide range of accident conditions. The PWR Owners Group is, therefore, satisfied with the degree of "operability" demonstrated at the Marshall facility for PORV block valves.

A report justifying these conclusions and the Owner's Group assertion that further block valve testing is not warranted is due to the staff in April 1982. After review of the block valve report, NRC will determine further actions to be taken to assure compliance with the NUREG-0737 block valve qualification requirement.

(f) The NRC is not aware of any other block valve tests performed by EPRI other than the tests discussed at the low power test hearing.

(g) By letter dated June 29, 1981 from Philip A. Crane, PG&E to Frank J. Miraglia, Jr., NRC, PG&E has identified Diablo Canyon valves represented in the EPRI program as Masoneilan model no. 20,000 series relief valve and Crosby model HB-BP-86, size 6M6 safety valve with loop seal internal materials.

A detailed report providing justification for all valves selected for test in the EPRI program is due to be submitted from EPRI to the NRC December 1, 1981. Additional plant specific information as noted above under item (d) is required to be submitted by July 1, 1982.

(h) The valve circuitry that is normally supplied with a power operated relief valve is being tested in the EPRI program. Other circuitry, such as that which runs from the valve to the control room would be qualified under requirements of other programs or standards such as the applicable requirements of IEEE-323.

As far as piping and supports are concerned, EPRI is heavily instrumenting the discharge piping at Combustion Engineering to obtain thermal hydraulic data which will be used to verify one or more computer codes that utilities will be able to use to verify that their discharge piping is adequate to handle all loads that could result from steam, liquid, or transition fluid flow.

(i) This response is provided under (g) above.

B. (a) NUREG-0737, Item II.D.1

(b) SER Suppl. 10 p. II.D.1 - II.D.2

(c) SER Suppl. 14 p. 3-12 through 3-14

(d) EPRI Memorandum of July 17, 1981 from J. Carey to Distribution. (Attachment K)

(e) September 4, 1981 Memorandum from W. J. Dircks, NRC to Robert D. Pollard, UCS. (Attachment L)

(f) October 19, 1981 Memorandum from F.C. Cherny to R.J. Bosnak entitled, "October 2, 1981 meeting with EPRI and PWR Utilities to review test results to date and proposed schedule for completion of balance of tests in response to item II.D.1, NUREG-0737." (Attachment M)

(g) SECY-81-491 dated August 17, 1981.

(h) Letter dated June 29, 1981 from Philip Crane, PG&E, to Frank J. Miraglia, Jr., NRC. (Attachment N)

C. Idaho National Laboratories are presently engaged in further research concerning the EPRI test program.

D. (1) Frank C. Cherny is the expert whom the Staff intends to have testify on the subject matter covered in the interrogatory. A copy of his professional qualifications is in Attachment C.

(2) See response to Interrogatory 50.D(2).

(3) See response to Interrogatory 50.D(3).

<u>Interrogatory 62</u>

On August 19, 1981, an emergency planning exercise for Diablo Canyon was held in San Luis Obispo. Based on your involvement in or observation of that exercise and your knowledge of the involvement of other persons, officials, agencies (including FEMA), or other entities, describe the exercise in detail and include in your response at least the following information:

(a) a detailed description of the exercise scenario employed, including the simulated events, time period and locations involved;

(b) the number of persons participating in the drill, including the specific company, agency, or other entity represented, if any, and the extent and nature of their involvement;

(c) (1) the number of PG and E personnel assumed or deemed to have been evacuated during the course of the exercise; (2) the number of PG and E employees actually evacuated, and (3) when such evacuation was begun and when completed;

(d) (1) the number of non-PG and E persons (e.g., members of the public) assumed or deemed to have been evacuated and/or sheltered during the course of the exercise, (2) the number of such persons actually evacuated and/or sheltered, and (3) when such evacuation was begun and when completed;

(e) (1) the number of ambulances assumed or deemed to have been utilized during the course of the exercise and (2) the number of ambulances actually utilized;

(f) (1) the number of simulated injured persons assumed or deemed to have been transported to and treated at French Hospital during the course of the exercise and (2) the number of simulated injured persons actually transported and treated at French Hospital;

(g) (1) the number of simulated injured persons assumed or deemed to have been transported to and treated at St. Francis Hospital in San Francisco during the course of the exercise and (2) the number of simulated injured persons actually transported to and treated at St. Francis Hospital;

(h) (1) the number of residences and/or households in San Luis Obispo and Santa Barbara Counties assumed or deemed to have been contacted during the exercise, (2) the number and location of such residence and/or households actually contacted, and (3) the time period required to contact such residences and/or households;

(i) (1) the number of automobiles assumed or deemed to have utilized Highway 101 as an evacuation route during the course of the exercise and (2) the number of automobiles which actually utilized Highway 101 as an evacuation route;

(j) (1) the number of persons or automobiles assumed or deemed to have utilized Highway 1 as an evacuation route during the course of the exercise and (2) the number of persons or automobiles which actually utilized Highway 1 as an evacuation route;

(k) (1) the number of persons or automobiles assumed or deemed to have utilized Avila Road as an evacuation route during the course of the exercise and (2) the number of persons or automobiles which actually used Avila Road as an evacuation route;

(l) (1) the number of persons assumed or deemed to have been notified of a radiological emergency occurring at Diablo Canyon during the course of the exercise, (2) the number and location of persons actually notified of such emergency, and (3) the time period required to complete such notification.

(m) (1) the number of emergency response personnel (i.e., law enforcement, fire, health, park, military, monitoring, etc.) assumed or deemed to have been mobilized and/or dispatched during the course of the exercise and (2) the number of such personnel actually mobilized and/or dispatched;

(n) (1) the protective actions assumed or deemed to have been taken within the plume exposure pathway EPZ by public officials, emergency response personnel, and members of the public during the course of the exercise and (2) the protective actions actually taken by such persons within the area specified;

(o) (1) the protective actions assumed or deemed to have been taken within the ingestion pathway EPZ by public officials, emergency response personnel, and members of the public during the course of the exercise and (2) the protective actions actually taken by such persons within the area specified;

(p) (1) the number and location of radiological monitoring samplings assumed or deemed to have been taken during the course of the exercise and (2) the number and location of such samplings actually taken;

(q) (1) the number of persons involved in the exercise and (2) the number of persons reasonably expected to be involved in an actual radiological emergency at Diablo Canyon;

(r) (1) the number of automobiles accidents or collisions, if any, assumed or deemed to have occurred on main evacuation routes during the course of the exercise and (2) the number of such accidents or collisions reasonably expected to occur in the event of full scale evacuation is ordered in response to an actual radiological emergency at Diablo Canyon;

(s) (1) the types and quantities of emergency response equipment (e.g., communications equipment, respiratory equipment, protective clothing, monitoring equipment, vehicles, helicopters, signs, placards, medical equipment, etc.) assumed or deemed to be available or to have been used during the course of the accident and (2) the types and quantities of such equipment actually available or used;

(t) (1) the number of media personnel present and inquiries from the public received during the course of the exercise and the number of such personnel likely to be present and inquiries from the public likely to be received in the event of an actual radiological emergency at Diablo Canyon;

(u) the names of all local and state officials, agencies, offices, and/or other entities actually notified as part of the exercise, by telephone or otherwise, regarding the simulated emergency at Diablo Canyon; the approximate time of each such notification; the precise language of the notification message; the name of the person who notified such officials and/or agencies; the names of each person who received the notice; and the time period required to complete notification of all such persons;

(v) a detailed description of the Staff's role, if any, in the exercise.

Response

A. (a) Responsive information is contained in the IE Inspection Report Nos. 50-275/81-21 & 50-323/81-15, attached to "NRC Staff's Response to Governor Edmund G. Brown Jr.'s Second Set of Interrogatories", served November 3, 1981; and the exercise scenario, attached hereto, as Attachment O.

(b) See 62(a). John Sears observed the full compliment of PG&E personnel as described in Section 5 of the PG&E Emergency Plan, and counted 46 in the TSC and 12 in the Control Room, plus two controllers and one consultant.

(c) (1) (2) & (3) Not known.

(d) (1) John Sears recollects that a busload of people from Montana de Oro Park was evacuated to Camp Roberts.

(2) Same as (d)(1).

(3) Not known.

(e) (1) One assumed in original scenario.

(2) Two used, due to collision.

(f) (1) One assumed in original scenario.

(2) Three, as scenario actually developed.

(g) (1) None.

(2) None.

(h) (1) All resident personnel within the EPZ were assumed to have been contacted.

(2) None were actually contacted.

(3) No time period has been established or determined.

(i) (1) Not known.

(2) No evacuating vehicles are known to have actually utilized this route.

(j) (1),(2) Same as (i)(1) & (2), above.

(k) (1),(2) Same as (i)(1) & (2), above.

(l) (1) All personnel within the EBS and TV range were assumed to have been notified during the course of the accident.

(2) Only persons notified were the actual exercise participants. The siren system was not actually activated.

(3) No specific time period has been determined.

(m) The number of personnel deemed or actually dispatched is not known since it was not estimated by the players.

(n) (1) Both sheltering and evacuation were simulated. Decisions were made during the exercise to evacuate affected downwind areas of the county.

(2) No specific protective actions were taken by members of the general public in the affected areas.

(o) (1) No protective action was assumed, only monitoring. The UDAC made a recommendation to put dairy cattle on stored feed.

(2) Same as (n)(2), above.

(p) (1) The specific number of samples assumed to have been taken is not known.

(2) Not known.

(q) (1) The number of persons involved in the exercise is estimated at approximately 200.

(2) The number of persons expected to be involved in an actual radiological emergency at Diablo Canyon would depend on the nature of the emergency.

(r) (1) One accident, between an ambulance and a station wagon, was assumed to have occurred for this particular exercise.

(2) No accidents would be anticipated as a result of an actual radiological emergency.

(s) (1) All equipment described in Applicant's Emergency Plan was assumed to be available.

(2) The exact amount of such equipment actually available cannot be recollected.

(t) No personal knowledge of the information requested.

(u) This request is objectionable for the reasons stated in Response No. 63.

(v) The Staff's role in the exercise was primarily limited to observation and evaluation. In addition, the NRC Resident Inspector played a role by communicating with Headquarters and the Regional office.

B. Not applicable.

C. The Staff and/or independent contractor are not presently engaged in or intend to engage in further research or work which may bear on the issues covered in the interrogatory other than normal staff review.

D. (1) John R. Sears and Dean Kunihiro are the experts whom the Staff intends to have testify on the subject matter covered in the interrogatory. Copies of their professional qualifications are in Attachments D and E respectively.

(2) See paragraph prior to Interrogatory 29.

(3) John R. Sears has testified:

(a) on emergency planning and security in Pacific Gas and Electric Company (Diablo Canyon Nuclear Power Plant, Units 1 and 2), Docket Nos. 50-275, 50-323;

(b) on emergency planning and security in Southern California Edison Company (San Onofre, Units 2 and 3), Docket Nos. 50-361, 50-360 OL;

(c) on emergency planning in a proceeding culminating in Commonwealth Edison Company (Zion Station, Units 1 and 2), LBP-80-7, 11 NRC 245 (1980);

(d) on emergency planning in Boston Edison Company (Pilgrim Nuclear Generating Station, Unit 2), Docket No. 50-471;

(e) on emergency planning in Long Island Lighting Company (Jamesport Nuclear Power Station, Units 1 and 3), Docket Nos. 50-516, 50-517; and

(f) on implementation of plant operations in Yankee Atomic Electric Company (Yankee Nuclear Power Station), Docket No. 50-029.

(3) Dean Kunihiro has testified:

(a) on security matters in Commonwealth Edison Company (Zion Station, Units 1 and 2), Docket No. 50-295, 50-304.

Interrogatory 63

Based on your involvement in or observation of the August 19, 1981 emergency planning exercise and your knowledge of the involvement of other persons, officials, agencies (including FEMA), or other entities in that exercise, provide a detailed chronology of all actions taken by the participants in connection with the exercise, and include in that chronology at least the following information:

(a) the time each action was taken;

(b) the name of the person taking the action;

(c) the office, agency, or other entity represented by that person;

(d) any problems or difficulties encountered by that person in taking the action;

(e) the location of the action, including, for example, point of origin and point of destination;

(f) any equipment (i.e., vehicles, walkie-talkie, radio, protective clothing, etc.) utilized in taking the action;

(g) the consequences resulting from the action.

Response

A. The observations and evaluation of the NRC Staff are reviewed in IE Inspection Report Nos. 50-275/81-21, 50-323/81-15, which, together with letter from H.E. Book, Chief, Radiological Safety Branch, USNRC, Region V, to Philip A. Crane, Jr., Assistant General Counsel, PG&E, has been provided to persons on the service list, as an attachment to "NRC Staff's Response to Governor Edmund G. Brown Jr., Second Set of Interrogatories." As noted therein, the examination of the exercise was selective. It was not practicable, nor deemed necessary, to review everything everyone who participated in the exercise did.

Insofar as this interrogatory seeks a chronology of "each action," and information related thereto, the request is too vague to answer, and is objectionable on that ground. Further, assembly of the information sought would require considerable research into information not in the Staff's possession; a request therefor is unduly burdensome and objectionable. <u>See</u> 4A Moore's Federal Practice, Para. 33.20(3).

B. Not applicable.

C. The Staff and/or independent contractor are not presently engaged in or intend to engage in further research or work which may bear on the issues covered in the interrogatory other than the normal staff review.

D. (1) John R. Sears and Dean Kunihiro are the experts whom the Staff intends to have testify on the subject matter covered in the interrogatory. Copies of their professional qualifications are in Attachments D and E respectively.

(2) See paragraph prior to Interrogatory 29.

(3) John R. Sears has testified:

(a) on emergency planning and security in Pacific Gas and Electric Company (Diablo Canyon Nuclear Power Plant, Units 1 and 2), Docket Nos. 50-275, 50-323;

(b) on emergency planning and security in Southern California Edison Company (San Onofre, Units 2 and 3), Docket Nos. 50-361, 50-360 OL;

(c) on emergency planning in a proceeding culminating in Commonwealth Edison Company (Zion Station, Units 1 and 2), LBP-80-7, 11 NRC 245 (1980);

(d) on emergency planning in Boston Edison Company (Pilgrim Nuclear Generating Station, Unit 2), Docket No. 50-471;

(e) on emergency planning in Long Island Lighting Company (Jamesport Nuclear Power Station, Units 1 and 3), Docket Nos. 50-516, 50-517; and

(f) on implementation of plant operations in Yankee Atomic Electric Company (Yankee Nuclear Power Station), Docket No. 50-029.

(3) Dean Kunihiro has testified:

(a) on security matters in Commonwealth Edison Company (Zion Station, Units 1 and 2), Docket No. 50-295, 50-304.

Interrogatory 64

Explain in detail how the exercise included such things as

(a) simulated casualties;

(b) offsite fire department assistance;

(c) rescue of personnel;

(d) use of protective clothing;

(e) deployment of radiological monitoring teams; and

(f) public information and notification activities.

Response

A. (a) See the scenario used for the August 19, 1981, exercise, attached.

(b) Same as 64A(a), above.

(c) Same as 64A(a), above.

(d) Same as 64A(a), above.

(e) Same as 64A(a), above.

(f) Same as 64A(a), above.

B. See the scenario used for the August 19, 1981, exercise, attached, and the IE Inspection Report, referenced in Responses 62 and 63, which has already been provided to persons on the service list.

C. The Staff and/or independent contractor are not presently engaged in or intend to engage in further research or work which may bear on the issues covered in the interrogatory other than the normal staff review. The Staff intends to publish its final conclusions on Diablo Canyon emergency preparedness after receipt and review of the FEMA evaluation of state and local plans.

D. (1) John R. Sears and Dean Kunihiro are the experts whom the Staff intends to have testify on the subject matter covered in the interrogatory. Copies of their professional qualifications are in Attachments D and E respectively.

(2) See paragraph prior to Interrogatory 29.

(3) John R. Sears has testified:

(a) on emergency planning and security in Pacific Gas and Electric Company (Diablo Canyon Nuclear Power Plant, Units 1 and 2), Docket Nos. 50-275, 50-323;

(b) on emergency planning and security in Southern California Edison Company (San Onofre, Units 2 and 3), Docket Nos. 50-361, 50-360 OL;

(c) on emergency planning in a proceeding culminating in Commonwealth Edison Company (Zion Station, Units 1 and 2), LBP-80-7, 11 NRC 245 (1980);

(d) on emergency planning in Boston Edison Company (Pilgrim Nucle Generating Station, Unit 2), Docket No. 50-471;

(e) on emergency planning in Long Island Lighting Company (Jamesport Nuclear Power Station, Units 1 and 3), Docket Nos. 50-516, 50-517; and

(f) on implementation of plant operations in Yankee Atomic Electric Company (Yankee Nuclear Power Station), Docket No. 50-029.

(3) Dean Kunihiro has testified:

(a) on security matters in Commonwealth Edison Company (Zion Station, Units 1 and 2), Docket No. 50-295, 50-304.

Interrogatory 65

State how, if at all, the August 19 exercise simulated and/or tested for the complicating effects of a major earthquake on emergency response capability at Diablo Canyon.

Response

A. No earthquake was simulated.

B. Not applicable.

C. The Staff and/or independent contractor are not presently engaged in or intend to engage in further research or work which may bear on the issues covered in the interrogatory other than the normal staff review. See Response 64C.

D. (1) John R. Sears and Dean Kunihiro are the experts whom the Staff intends to have testify on the subject matter covered in the interrogatory. Copies of their professional qualifications are in Attachments D and E respectively.

(2) See paragraph prior to Interrogatory 29.

(3) John R. Sears has testified:

(a) on emergency planning and security in Pacific Gas and Electric Company (Diablo Canyon Nuclear Power Plant, Units 1 and 2), Docket Nos. 50-275, 50-323;

(b) on emergency planning and security in Southern California Edison Company (San Onofre, Units 2 and 3), Docket Nos. 50-361, 50-360 OL;

(c) on emergency planning in a proceeding culminating in Commonwealth Edison Company (Zion Station, Units 1 and 2), LBP-80-7, 11 NRC 245 (1980);

(d) on emergency planning in Boston Edison Company (Pilgrim Nuclear Generating Station, Unit 2), Docket No. 50-471;

(e) on emergency planning in Long Island Lighting Company (Jamesport Nuclear Power Station, Units 1 and 3), Docket Nos. 50-516, 50-517; and

(f) on implementation of plant operations in Yankee Atomic Electric Company (Yankee Nuclear Power Station), Docket No. 50-029.

(3) Dean Kunihiro has testified:

(a) on security matters in Commonwealth Edison Company (Zion Station, Units 1 and 2), Docket No. 50-295, 50-304.

Interrogatory 66

State what, if any, critical emergency response equipment (i.e., vehicles, communications sytems and lines, monitoring equipment, notification sirens, etc.) were assumed to fail during the course of the August 19 exercise.

Response

A. An ambulance carrying a contaminated injured person from the plant was assumed to be in a collision with a station wagon.

B. See Response 64B.

C. The Staff and/or independent contractor are not presently engaged in or intend to engage in further research or work which may bear on the issues covered in the interrogatory other than the normal staff review. See Response 64C.

D. (1) John R. Sears and Dean Kunihiro are the experts whom the Staff intends to have testify on the subject matter covered in the interrogatory. Copies of their professional qualifications are in Attachments D and E respectively.

(2) See paragraph prior to Interrogatory 29.

(3) John R. Sears has testified:

(a) on emergency planning and security in Pacific Gas and Electric Company [illegible] nyon Nuclear Power Plant, Units 1 and 2), Docke[illegible]0-275, 50-323;

(b) on emergen[illegible]ning and security in Southern California Edison Company (San Onofre, Units 2 and 3), Docket Nos. 50-361, 50-360 OL;

(c) on emergency planning in a proceeding culminating in Commonwealth Edison Company (Zion Station, Units 1 and 2), LBP-80-7, 11 NRC 245 (1980);

(d) on emergency planning in Boston Edison Company (Pilgrim Nuclear Generating Station, Unit 2), Docket No. 50-471;

(e) on emergency planning in Long Island Lighting Company (Jamesport Nuclear Power Station, Units 1 and 3), Docket Nos. 50-516, 50-517; and

(f) on implementation of plant operations in Yankee Atomic Electric Company (Yankee Nuclear Power Station), Docket No. 50-029.

(3) Dean Kunihiro has testified:

(a) on security matters in Commonwealth Edison Company (Zion Station, Units 1 and 2), Docket No. 50-295, 50-304.

Interrogatory 67

State what, if any, evacuation routes

(a) for the site and

(b) for the plume exposure pathway EPZ

were assumed to be fully or partially blocked during the course of the August 19 exercise.

Response

A. (a) None.

(b) None.

B. See Response 64B.

C. The Staff and/or independent contractor are not presently engaged in or intend to engage in further research or work which may bear on the issues covered in the interrogatory other than the normal staff review. See Response 64C.

D. (1) John R. Sears and Dean Kunihiro are the experts whom the Staff intends to have testify on the subject matter covered in the interrogatory. Copies of their professional qualifications are in Attachments D and E respectively.

(2) See paragraph prior to Interrogatory 29.

(3) John R. Sears has testified:

(a) on emergency planning and security in Pacific Gas and Electric Company (Diablo Canyon Nuclear Power Plant, Units 1 and 2), Docket Nos. 50-275, 50-323;

(b) on emergency planning and security in Southern California Edison Company (San Onofre, Units 2 and 3), Docket Nos. 50-361, 50-360 OL;

(c) on emergency planning in a proceeding culminating in Commonwealth Edison Company (Zion Station, Units 1 and 2), LBP-80-7, 11 NRC 245 (1980);

(d) on emergency planning in Boston Edison Company (Pilgrim Nuclear Generating Station, Unit 2), Docket No. 50-471;

(e) on emergency planning in Long Island Lighting Company (Jamesport Nuclear Power Station, Units 1 and 3), Docket Nos. 50-516, 50-517; and

(f) on implementation of plant operations in Yankee Atomic Electric Company (Yankee Nuclear Power Station), Docket No. 50-029.

(3) Dean Kunihiro has testified:

(a) on security matters in Commonwealth Edison Company (Zion Station, Units 1 and 2), Docket No. 50-295, 50-304.

Interrogatory 68

In light of the information and experience gained from the August 19 exercise, what revisions, changes, or alterations, if any, will be required in the following documents prior to full power operation of Diablo Canyon:

(a) the Diablo Canyon on-site emergency plan and emergency procedures;

(b) the San Luis Obispo County emergency and evacuation plans;

(c) the State of California emergency plan;

(d) the San Luis Obispo County Sherrif's "plan" (Board Exh. 5 at Diablo Low Power Test hearing).

Response

A. (a) None.

(b) No specific determination has been made by FEMA with regard to what modifications to the referenced plans are necessary. However, since modifications to operating procedures were identified and discussed in an October 15, 1981 meeting of representatives of the State, County, NRC Region V, and Applicant. The County and Applicnt are working on the modifications identified during that meeting.

(c) Same as 68A(b).

(d) Same as 68A(a).

B. See IE Inspection Report referenced in Responses 62 and 63.

C. The Staff and/or independent contractor are not presently engaged in or intend to engage in further research or work which may bear on the issues covered in the interrogatory other than the normal staff review. See Response 64C.

D. (1) John R. Sears and Dean Kunihiro are the experts whom the Staff intends to have testify on the subject matter covered in the interrogatory. Copies of their professional qualifications are in Attachments D and E respectively.

(2) See paragraph prior to Interrogatory 29.

(3) John R. Sears has testified:

(a) on emergency planning and security in Pacific Gas and Electric Company (Diablo Canyon Nuclear Power Plant, Units 1 and 2), Docket Nos. 50-275, 50-323;

(b) on emergency planning and security in Southern California Edison Company (San Onofre, Units 2 and 3), Docket Nos. 50-361, 50-360 OL;

(c) on emergency planning in a proceeding culminating in Commonwealth Edison Company (Zion Station, Units 1 and 2), LBP-80-7, 11 NRC 245 (1980);

(d) on emergency planning in Boston Edison Company (Pilgrim Nuclear Generating Station, Unit 2), Docket No. 50-471;

(e) on emergency planning in Long Island Lighting Company (Jamesport Nuclear Power Station, Units 1 and 3), Docket Nos. 50-516, 50-517; and

(f) on implementation of plant operations in Yankee Atomic Electric Company (Yankee Nuclear Power Station), Docket No. 50-029.

(3) Dean Kunihiro has testified:

(a) on security matters in <u>Commonwealth Edison Company</u> (Zion Station, Units 1 and 2), Docket No. 50-295, 50-304.

Respectfully submitted,

George E. Johnson

George E. Johnson
Counsel for NRC Staff

Dated at Bethesda, Maryland
this 6th day of November 1981

ATTACHMENT A

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
PACIFIC GAS AND ELECTRIC COMPANY	)	Docket Nos. 50-275 O.L.
	)	50-323 O.L.
(Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2)	)	

WA'.TON L. JENSEN, JR.
PROFESSIONAL QUALIFICATIONS

I am a Senior Nuclear Engineer in the Reactor Systems Branch of the Nuclear Regulatory Commission. In thi- 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 qualifications 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.

ATTACHMENT B

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
PACIFIC GAS AND ELECTRIC COMPANY	)	Docket Nos. 50-275 O.L.
	)	50-323 O.L.
(Diablo Canyon Nuclear Power Plant,	)	
Unit Nos. 1 and 2)	)	

JOHN L. KNOX

PROFESSIONAL QUALIFICATIONS

Education

A.A. Electrical Technology, 1962
Montgomery College
Takoma Park, Maryland

Major: Electrical Power Systems

B.S. Electrical Engineering, 1971
University of Maryland
College Park, Maryland

Major: Electronic Systems Engineering

Professional Qualifications

From 1971-1974, I worked for Potomac Electric Power Company in Washington, D. C. I was assigned to the underground power Transmission Engineering Group and my duties included relocation and restoration of underground power and transmission cables due to the subway construction project. (Prior to this, I spent four years in the Air Force working on the F4 aircraft electronic weapons control systems.)

From 1974 to the present, I have worked for the Nuclear Regulatory Commission involved in the technical review of electrical systems (onsite and offsite power, instrumentation and control). Through 1976, I was a member of the Electrical Instrumentation and Control Systems Branch. This branch was split in January 1977 into an I&C branch and a power branch. Since this split, I have been a member of the Power Systems Branch. My present title is Senior Reactor Systems Engineer (Electrical).

ATTACHMENT C

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
PACIFIC GAS AND ELECTRIC COMPANY	)	Docket Nos. 50-275 O.L.
	)	50-323 O.L.
(Diablo Canyon Nuclear Power Plant,	)	
Unit Nos. 1 and 2)	)	

FRANK C. CHERNY

PROFESSIONAL QUALIFICATIONS

MECHANICAL ENGINEERING BRANCH

DIVISION OF ENGINEERING

I am a Mechanical Engineer in the Mechanical Engineering Branch responsible for the review and evaluation of design criteria of mechanical components, of methods of dynamic analysis and testing of safety related systems and components and of criteria for protection against dynamic effects associated with postulated failures of fluid system components for nuclear service.

I graduated from Marquette University with a B.S. degree in Mechanical Engineering in 1965.

From July 1965 to November 1968 I was employed by the Babcock & Wilcox Co. at offices in both Barberton and Akron, Ohio. During the majority of this period I was engaged in materials engineering work, primarily writing technical ordering requirements for primary pressure boundary materials to be used for reactor vessels, steam generators, and pressurizers for both commercial and U. S. Navy nuclear systems. In addition I had assignments of several months duration each in quality control engineering and nuclear steam supply system performance engineering.

From November 1968 to May 1974 I was employed in the Pressurized Water Reactors Division of Westinghouse Nuclear Energy Systems. My work experience during this period includes the following:

From November 1968 to May 1970 and September 1970 to April 1971 as a Reactor Vessel Project Engineer based in Monroeville, Pa.:

(1) I had overall project engineer responsibility for design and construction of reactor vessels for several Westinghouse nuclear power plants in the U.S. My responsibilities included preparation of Design Specifications and review of vendor decumentation for compliance with Westinghouse, Utility, ASME, Architect Engineer and AEC requirements. I was personally responsible for coordination of the technical aspects of the transfer of two partially completed reactor vessels from a U.S. manufacturer's shop to a European manufacturer for completion when schedular problems developed at the U.S. manufacturer.

(2) After the U.S.-Europe transfer of these components, I assumed responsibility for technical coordination between the primary Westinghouse nuclear engineering office in the U.S. and an overseas office established in Brussels, Belgium to do project engineering work for mechanical components used in Westinghouse nuclear plants both in the U.S. and in Europe.

From June 1970 to September 1970 and from April 1971 to December 1972 I was employed by Westinghouse Nuclear Energy Systems in Europe based in Brussels, Belgium. My responsibilities included:

(1) During the June-September, 1970 period and from April 1971 to about April 1972 I had project engineer responsibility for several reactor pressure vessels and a pressurizer. I also acted as Westinghouse engineering representative for U.S. AEC Quality Assurance audits of European vessel manufacturers.

(2) From April 1972 to December 1972 I served as a lead engineer with a broader scope of responsibility. I was responsible for reactor pressure vessels, pressurizers and reactor vessel supports fabricated in Europe for Westinghouse Nuclear Plants. Several engineers and a technician reported directly to me during this period. The work included preparation of Design Specifications, review and approval of vendor design and manufacturing documentation, and coordination with both U.S. and European utility and regulatory representatives.

From December 1972 to May 1974 I was again based in Monroeville, Pa., this time as Senior Reactor Vessel Project Engineer. I was responsible for the technical adequacy of several reactor pressure vessels being manufactured in the U.S. for use in Westinghouse Nuclear Plants in Europe. I was also responsible, during the majority of this period, for the training of a Westinghouse Nuclear - Europe engineer temporarily based in the U.S.

In May of 1974 I started work for the Regulatory Division of the U.S. Atomic Energy Commission and have remained through the transition into the U.S. Nuclear Regulatory Commission. In 1977 I was appointed as a Section Leader in the Mechanical Engineering Branch. As a branch technical reviewer and later as a Section Leader I have been participating in the review of construction permit and Operating License applications.

Since July of 1974 I have served as a member of the ASME Section III Subgroup On Pressure Relief which is responsible for writing industry standards for the overpressure protection of light water reactor plant components. Additionally, since 1977 I have been a member of the ASME Working Group on Safety and Relief Valves. The Working Group has recently completed work on a proposed industry standard entitled "Requirements for Inservice Performance Testing of Nuclear Power Plant Pressure Relief Devices." The proposed standard was issued for public comment by ASME early in 1981.

ATTACHMENT D

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
PACIFIC GAS AND ELECTRIC COMPANY	)	Docket Nos. 50-275 O.L.
	)	50-323 O.L.
(Diablo Canyon Nuclear Power Plant	)	
Unit Nos. 1 and 2)	)	

JOHN R. SEARS

PROFESSIONAL QUALIFICATIONS

EMERGENCY PREPAREDNESS PROGRAM OFFICE

Prior to 1952, I was employed in field jobs in various aspects of mechanical engineering. In 1952, I joined Brookhaven National Laboratory as a Reactor Shift Supervisor on the Brookhaven Graphite Reactor. While at Brookhaven, I completed a series of courses given by the Nuclear Engineering Department in nuclear engineering. These courses were patterned on the ORSORT programs. In 1956, I was appointed Project Engineer on the Brookhaven Medical Research Reactor. I was a member of the design group, participated in critical design experiments, wrote specifications, coauthored the hazards report, was responsible for field inspection and contractor liaison, trained operators and loaded and started up the reactor. About three months after start-up, in 1959, following the successful completion of proof tests and demonstration of the reactor in its design operating mode for boron capture therapy of brain cancer, I accepted a position as reactor inspector with the Division of Inspection, U. S. Atomic Energy Commission. In 1960, I transferred, as a reactor inspector, to the newly-formed Division of Compliance. I was responsible for the inspection, for safety and compliance with license requirements, of the licensed reactors and the fuel fabrication and fuel processing plants, which use more than critical amounts of special nuclear material, in the Eastern United States.

In September 1968, I transferred to the Operational Safety Branch, Directorate of Licensing. My responsibility included development of appropriate guides for evaluation of operationsl aspect of license applications and staff assistance in review of power reactor applicants submittals in the areas of Organization and Management. Personnel Qualifications, Training Programs, Procedures and Administrative Control, Review and Audit, Start-up Testing Programs Industrial Security and Emergency Planning.

The Branch was reorganized as the Industrial Security and Emergency Planning Branch in April 1974 to place increased emphasis and attention upon areas of physical security and emergency planning.

In 1976, I transferred to the Division of Operating Reactors as the sole reviewer responsible for review of emergency planning for all the operating reactors in the United States.

New York City College, 1950 - Mechanical Engineering

Argonne International School of Reactor Technology, 1961 - Reactor Control Course

GE BWR System Design Course, 1972

Popo-U.S. Army, 1974 - Course in Industrial Defense and Disaster Planning

Instructor at DCPA, 1976, 1977 - Course in Emergency Planning

Director, 1962 - Reactor Program, Atoms for Peace Exhibit, Bangkok, Thailand

Director, 1966 - Atoms for Peace Exhibit, Utrecht, Holland

ATTACHMENT E

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
PACIFIC GAS AND ELECTRIC COMPANY	)	Docket Nos. 50-275 O.L.
	)	50-323 O.L.
(Diablo Canyon Nuclear Power Plant Unit Nos. 1 and 2)	)	

DEAN M. KUNIHIRO
PROFESSIONAL QUALIFICATIONS

I joined the Nuclear Regulatory Commission Staff in 1976 as a Program Analyst in the Office of Nuclear Material Safety and Safeguards. Following that assignment I was assigned as a Reactor Safeguards Analyst in the Office of Nuclear Reactor Regulation. In that position I was charged with the technical evaluation of the physical security plans submitted pursuant to 10 C.F.R. 73.55 for 13 nuclear power facilities. In addition, I was selected to serve on the Emergency Planning Task Force and was tasked with the technical evaluation of emergency plans for 8 nuclear power facilities. Among these facilities was the Diablo Canyon Nuclear Power Plant.

I am presently assigned as the Regional State Liaison Officer, U.S. NRC Region V. As part of my duties I serve as a member of the Regional Assistance Committees in Federal Regions IX and X. These committees were established to assist in development and evaluation of the offsite emergency preparedness around nuclear power facilities.

Prior to joining the Nuclear Regulatory Commission, I served in the U.S. Army in a variety of assignments to include combat infantry duty in Viet Nam, and radiobiology research with the Defense Nuclear Agency.

I have a Bachelor of Science degree from the United States Military Academy and a Master of Science degree in Physics from the Naval Post-graduate School.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of PACIFIC GAS AND ELECTRIC COMPANY (Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2)	Docket Nos. 50-275 O.L. 50-323 O.L.

AFFIDAVIT OF WALTON L. JENSEN

I, Walton L. Jensen, being duly sworn, state as follows:

1. I am employed by the U.S. Nuclear Regulatory Commission as a Senior Nuclear Engineer, Reactor Systems Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation.

2. I am duly authorized to participate in answering Interrogatories 29-32, 34, 36, 37, 40, 43, 45-49, 51-53, and 57-59 and I hereby certify that the answers given are true to the best of my knowledge.

Walton L. Jensen Jr
Walton L. Jensen

Subscribed and sworn to before me this 27th day of October, 1981.

Marilyn Albertson
Notary Public

My Commission expires: July 1, 1982

ATTACHMENT G

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
PACIFIC GAS AND ELECTRIC COMPANY	)	Docket Nos. 50-275 O.L.
	)	50-323 O.L.
(Diablo Canyon Nuclear Power Plant Unit Nos. 1 and 2)	)	

AFFIDAVIT OF JOHN R. SEARS

I, John R. Sears, being duly sworn, state as follows:

1. I am employed by the U.S. Nuclear Regulatory Commission as a Senior Reactor Safety Engineer, Emergency Preparedness, Office of Inspection and Enforcement.

2. I am duly authorized to participate in answering Interrogatories 62 through 68 and I hereby certify that the answers given are true to the best of my knowledge.

John R. Sears
John R. Sears

Subscribed and sworn to before me
this 23rd day of October, 1981

Linda M. Oyler
Notary Public

My Commission Expires: July 1, 1982

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
PACIFIC GAS AND ELECTRIC COMPANY	)	Docket Nos. 50-275 O.L.
	)	50-32- O.L.
(Diablo Canyon Nuclear Power Plant	)	
Units Nos. 1 and 2)	)	

AFFIDAVIT OF DEAN M. KUNIHIRO

I, Dean M. Kunihiro, being duly sworn, state as follows:

1. I am employed by the U.S. Nuclear Regulatory Commission as Regional State Liaison Officer, Office of Inspection and Enforcement.

2. I am duly authorized to participate in answering Interrogatories 62 through 68 and I hereby certify that the answer given is true to the best of my knowledge.

Dean M. Kunihiro

Subscribed and sworn to before me this 20th day of October, 1981

Notary Public

NOTARY ...
CONTRA ...
My Comm. ...

My Commission Expires: ______________

ATTACHMENT I

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of PACIFIC GAS AND ELECTRIC COMPANY (Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2)	))))))	Docket Nos. 50-275 O.L. 50-323 O.L.

AFFIDAVIT OF FRANK C. CHERNY

I, Frank C. Cherny, being duly sworn, state as follows:

1. I am employed by the U.S. Nuclear Regulatory Commission as a Mechanical Engineer, Mechanical Engineering Branch, Division of Engineering, Office of Nuclear Reactor Regulation.

2. I am duly authorized to participate in answering Interrogatories 50, 56, and 61 and I hereby certify that the answers given are true to the best of my knowledge.

Frank C. Cherny
Frank C. Cherny

Subscribed and sworn to before me
this 27th day of October, 1981.

Marilyn Jallerstein
Notary Public

My Commission expires: July 1, 1982

ATTACHMENT J

UN ..? STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
PACIFIC GAS AND ELECTRIC COMPANY	)	Docket Nos. 50-275 O.L.
	)	50-323 O.L.
(Diablo Canyon Nuclear Power Plant,	)	
Unit Nos. 1 and 2)	)	

AFFIDAVIT OF JOHN L. KNOX

I, John L. Knox, being duly sworn, state as follows:

1. I am employed by the U.S. Nuclear Regulatory Commission as a Senior Electrical Engineer, Power Systems Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation.

2. I am duly authorized to participate in answering Interrogatories 33, 35, 36, 39, 41, 42, 44 and 54-56 and I hereby certify that the answers given are true to the best of my knowledge.

John L Knox
John L. Knox

Subscribed and sworn to before me
this 27th day of October, 1981.

Marilyn Jollenstein
Notary Public

My Commission expires: July 1, 1982

Memorandum

EPRI

July 17, 1981

TO: DISTRIBUTION

FROM: John J. Carey JJC.

SUBJECT: S/RV TEST ACTIVITIES

The EPRI/PWR Safety and Relief Valve Test Program testing activities for the period of July 13-17 were as follows:

WYLE

During the period from Friday, July 10 through Wednesday, July 16 testing was performed on the Masoneilan relief valve. The tests were performed under steam, preload, water, transition and water seal simulation conditions. One additional full pressure, 330°F water test and two repeat tests were also performed. The two repeat tests had a slightly increased air supply pressure to the air actuator to improve valve opening time. A total of eleven tests were performed. For all tests the valves opened and closed on demand. The valve was disassembled and inspected by the Masoneilan valve representative. No damage was observed that would affect future valve performance. The cage to body gasket had washed out during testing.

The Copes Vulvan relief valve utilizing the 17-4 ph plug and cage was installed today. Testing is scheduled to start tomorrow, July 18.

COMBUSTION ENGINEERING

During this week four tests were performed on the Crosby 3K6 safety valve. This valve has a design set pressure of 2500 psia. The first three tests were low ramp rate, short duration, high backpressure, steam tests. Due to computer and instrumentation problems encountered during the first two tests, all data was not recorded.

The third test was performed on Thursday, July 16. For this test, the valve opened at a pressure within $\pm$ 3% of the valve design set pressure. A maximum stem position of 98% of rated lift was achieved. Rated flow was achieved.* The valve closed at a pressure of 2245 psia, which is .2% below the EPRI blowdown pressure criterion of 2250 psia. Peak backpressure for this test was 680 psia.

On Friday, July 17 the fourth test on the Crosby 3K6 safety valve was performed. This test was a high ramp rate, high backpressure, steam test. The valve opened at a pressure within $\pm$ 3% of the valve design set pressure. A maximum stem position of 99% of rated lift was achieved at a pressure of 6% above the valve design set pressure. Rated flow was achieved.* The valve closed at a pressure of 2225 psia, which is 1% below the EPRI blowdown pressure criterion of 2250 psia. Peak backpressure for this test was 620 psia.

The next test on the Crosby 3K6 safety valve is scheduled for Monday, July 20.

*Based on preliminary venturi flow data.

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

ATTACHMENT L

September 4, 1981

Mr. Robert D. Pollard
Union of Concerned Scientists
1725 I Street, N.W., Suite 601
Washington, D.C. 20006

Dear Mr. Pollard:

We have reviewed your letter of March 11, 1981 and its attachment which discusses the recent testing performed by the Electric Power Research Institute (EPRI) on closure of block valves of the type used on Pressurized Water Reactors (PWRs) for isolation of power-operated relief valves (PORVs). Your letter raises questions about the testing of the block valves, and the treatment of unfavorable test results by EPRI and the NRC staff. Your letter also questions the schedule that the Commission has accepted for completion of valve qualification in light of the test results to date.

The proposed full scale qualification testing of PORV block valves, with a completion date of July 1, 1982, was first provided to the utilities in a September 5 draft of NUREG-0737. The item was formally issued, with Commission approval, in NUREG-0737 on October 31, 1980. The block valve qualification was imposed primarily as an additional means of reducing the number of challenges to the emergency core cooling system (ECCS). As you note in your letter, repeated unnecessary challenges to the ECCS is undesirable.

In establishing a qualification completion date for block valves, the Commission took into account that the PORVs were to be qualified by July 1, 1981, and that some, if not all, of the same test facilities being used for NUREG-0737 safety valve and PORV testing must also be used for block valve testing. EPRI has subsequently expanded the PORV and safety valve test programs and the completion dates have been revised to about November 1, 1981, and April 1, 1982, respectively. Further, after the TMI-2 accident, updated analyses for all PWRs with PORVs were submitted by licensees and reviewed by the staff. The analyses demonstrate that all of these plants can be safely shut down despite one or more stuck-open PORVs without taking credit for the block valve(s).

During the public comment period, following issuance of the September 5, 1980 draft of NUREG-0737, no exceptions were taken by any utility to the proposed qualification of block valves. However, after issuance of NUREG-0737, the NRC was notified by letter of December 15, 1980 (R.C. Youngdahl of Consumers Power Company to D.G. Eisenhut of NRC) that the PWR Owners

Group would not commit to undertake a block valve testing program until the safety and relief valve testing program was completed. All PWR licensees in responding to this NUREG-0737 item referred to the December 15, 1980 Owners Group letter.

Since receipt of these letters, the staff has had discussions with the Owners Group concerning the block valve testing. The staff met with the Owners Group and EPRI representatives in Bethesda on April 8, 1981 to discuss the content of a Block Valve Testing Program Plan. Since that time, EPRI and the PWR Owners Group have had additional discussions regarding formal establishment of a generic block valve program. The NRC has been kept informed on the progress of these discussions.

An additional meeting was held between the staff and the PWR Owners Group on July 17, 1981, in which the PWR Owners Group presented its conclusion that it need not commit to any PORV block valve testing beyond that already completed by EPRI for the following reasons: the isolation of a stuck-open PORV is not required to safely shut down a plant, as shown by PWR NSSS vendor analyses; post-TMI plant procedures address means for dealing with stuck-open PORVs; EPRI's testing of PORVs has demonstrated that PORVs perform well over a wide range of accident conditions. The PWR Owners Group is therefore satisfied with the degree of "operability" demonstrated at the Marshall facility for PORV block valves.

A report justifying these conclusions and the Owners Group assertion that further block valve testing is not warranted is due to the staff in April 1982. NRC will determine further actions to be taken to assure compliance with the NUREG-0737 block valve qualification requirement. From the point of view of the PWR licensees, there has not as yet been a formally agreed-upon test program for block valves up to the present time.

With the preceding as background, I will now address the limited block valve testing that EPRI has conducted to date and the specific questions raised in your letter with regard to them.

For the safety valve and PORV testing program that EPRI is conducting for the PWR licensees and permittees pursuant to NUREG-0737, EPRI is utilizing three test facilities. One is the Marshall facility owned by Duke Power Company, the second is the Norco, California, facility of Wyle Laboratories, and the third facility is at Combustion Engineering in Windsor, Connecticut.

Of the three facilities, the one at Marshall was the most readily adaptable for use in the PORV qualification effort. Steam testing of PORVs was thus started there in July 1980 and continued through January 1981.

The NRC staff and representatives of the PWR Owners Group and EPRI have held several meetings since December 1979 to discuss the program plan for safety valve and PORV testing. From informal conversations at meetings such as these, PWR licensees and EPRI personnel were made aware that an NRC requirement for PWR block valve testing would probably be issued at some future date.

In anticipation of such a requirement, EPRI decided to make provisions in the Marshall test facility for the installation of block valves between the test steam source and the test PORV. Test PORVs had been carefully selected, with close coordination between EPRI, its consultants and PWR utilities, to assure that PORVs representative of those in service or intended for service would be tested. However, the block valves that were tested concurrently were selected on an as-available basis because there was no formal NRC block valve test program requirement (NUREG-0737 had not yet been issued) and no specific test program had been formulated.

Therefore, seven readily available valves were obtained by EPRI and tested, primarily to obtain some general baseline information on block valve closure capability. For the block valves that were tested, EPRI, at least at the time of testing, had not established what population of plants, either operating or under construction, might have a valve of the type tested.

In addition, it should be noted that the test conditions used at Marshall were only those that were determined to be applicable for steam testing of PORVs. These test conditions were selected after review by EPRI, the PWR utilities, and the PWR NSSS vendors. They were also reviewed and concurred in by the NRC staff. To date, EPRI and NRC have made no determination about the applicability of the Marshall block valve test conditions to any specific plant.

To date, EPRI has tested a total of seven PORV block valves, all at the Marshall facility. In two of the tests, valves manufactured by Westinghouse would not close fully under the test conditions. These are the two Westinghouse valves referred to in the unsigned January 14, 1981 EPRI letter attached to your letter. In addition, a valve made by Anchor-Darling exhibited similar behavior. In general, the failures resulted because the valve operators had too low a torque setting on the valve operator. After modification of the torque setting, the previously failed valves were retested and all closed satisfactorily.

Your letter raises questions about the two Westinghouse valve failures to close; the following questions and responses refer to your letter:

<u>Question:</u> Did the staff bring to the attention of the Commission any information regarding adverse results from block valves tests: if so, when?

<u>Response:</u> As of the date of your letter, the staff had not brought the information about these block valve failures to the attention of the Commission. As explained further below, the staff had concluded that, based on information it received, all affected utilities and Westinghouse were taking the test results properly into account and were instituting programs to modify the block valves in a timely manner.

<u>Question:</u> Was EPRI in compliance with 10 CFR Part 21 reporting requirements insofar as the two Westinghouse valves are concerned?

Response: Although important to safety, both PORVs and PORV block valves are not classified as safety grade equipment. No credit is taken for the overpressure protection capability of the PORVs in the plant safety analysis which must demonstrate that the plant can be safely shut down with all PORVs in their open position. As a result, these valves are not classified as safety grade equipment and the reporting requirements of 10 CFR 21 are not directly applicable to them. There were not any violations of the specific requirements of 10 CFR Part 21 by EPRI regarding the results of Westinghouse gate valves tested in PORV block valve applications.

Even though the PORV block valves are not subject to the reporting requirements of 10 CFR 21, because of their importance to plant safety the staff reviewed the handling of the EPRI test results by Westinghouse between the time of the initial test failure in July, 1980 up to and including the formal communication of information to affected customers about the reduced valve capability. The staff believes that Westinghouse's conduct of the safety review committee meetings and subsequent reporting of results to NRC could have been more expeditious because of the increased public and industry awareness of the relative importance of these valves for terminating a small-break LOCA that could result from a stuck-open PORV.

Question: Which of the three Oconee plants utilize the block valve which is known to be incapable of closing against full flow? Is the affected Oconee unit(s) in operation? If so, on what basis is continued operation being permitted?

Response: Westinghouse advisory letters on the three-inch gate valves in question were sent out to the affected utilities in late October-early November 1980. One such utility was the Duke Power Company. Duke Power purchased some of the three-inch valves as spares and subsequently installed them as PORV block valves on the Oconee plants, Units 1, 2, and 3. These valves are the same model that exhibited the failure to close at Marshall in September.

Duke Power received two Westinghouse advisory letters near the end of October. The licensee recognized that one of the valves discussed in these letters had been installed as a PORV block valve on each Oconee unit. The licensee performed a safety evaluation in accordance with 10 CFR 50.59 requirements and concluded there was no unanalyzed safety problem; first, because the plants have been analyzed for safe shutdown capability with a stuck-open PORV, and second, because of various plant modifications made after the TMI-2 accident, the licensee was confident that the probability of a PORV actuating in response to a plant transient had been substantially reduced on all three Oconee reactors. At the time of this determination by the licensee, Unit 2 was shut down and Units 1 and 3 were operating.

Prior to startup, the utility staff increased the operator torque switch setting for the Unit 2 block valve to increase the valve closure force. Shortly thereafter, Unit 3 was shut down for refueling. During that outage, modifications recommended by Westinghouse were made to the block valve. These same modifications were made shortly thereafter on Unit 2. The Unit 1 block valve is being modified in accordance with the Westinghouse recommendations during the current extended shutdown which began June 26, 1981. The Office of Inspection and Enforcement has verified the described actions by the licensee through the IE resident inspector.

The staff believes that the licensee has taken suitable corrective action to upgrade the Oconee block valves by making the necessary modifications at the first outage of sufficient duration to accomplish the changes and that continued operation of the Oconee units, for the period of time until all modifications have been made, is acceptable.

Question: Does any other operating PWR utilize a block valve which can not "be operated, closed, and opened for all fluid conditions expected under operating and accident conditions" (NUREG-0737, page 3-73) ... For each operating plant, what is the basis for this determination?

Response: The staff is aware of only one other use of a Westinghouse block valve on an operating reactor, at the North Anna 2 facility of Virginia Electric Power Company (VEPCO). North Anna 2 has two PORVs and two block valves. One of the two block valves is the Westinghouse design. VEPCO received two Westinghouse advisory letters about the same time as Duke Power.

VEPCO, following a rationale similar to that used by Duke Power for Oconee, made the decision that the possible lack of block valve closure capability was not reportable as a safety issue because of the recently performed Westinghouse analyses approved by the NRC, that demonstrated the North Anna units could be adequately cooled with one or both PORVs stuck open.

VEPCO continued operating North Anna 2 with both block valves open and completed the Westinghouse recommended valve modifications at an extended shutdown which began in May 1981. In addition, the North Anna 2 reactor operators were cautioned that under certain flow conditions the Westinghouse block valve might not fully close. The IE resident inspector was informed of these actions and has verified that the reactor operators have been cautioned. We note that the Westinghouse PORV has had a higher operating reliability than the TMI-2 PORV or those on other B&W NSSS designs. The staff believes that there was sufficient safety basis for continued operation of North Anna 2 until the extended outage that commenced in May.

As noted above, in addition to the Westinghouse failures, one additional block valve tested by EPRI at Marshall failed to close. The staff has received information of this failure since receipt of your letter. The valve was subsequently modified by the manufacturer, Anchor-Darling, and then successfully retested. As far as is known, none of the unmodified models of this valve is currently in reactor service.

The Office of Inspection and Enforcement has been reviewing Westinghouse gate valve applications at other operating reactors. In early April, IE issued Bulletin 81-02 which requests information on installation of gate-type valves which may not isolate adequately against pressure differentials. Responses to the Bulletin received from operating plant utilities, indicate that there is one other reactor, Indian Point 2, that utilizes the Westinghouse gate valve as a block valve. On Indian Point 2, the block valves have been modified according to Westinghouse recommendations.

Additionally, responses to the Bulletin indicate that the Westinghouse gate valves have been used in other system applications. However, no licensees determined that the reduced closure capability for these valves constitute an unanalyzed safety problem for their plants.

Question: Has the staff reported the information provided by Westinghouse to any licensing board? If not, why not and what steps have you taken to ensure that the staff promptly reports relevant information in the future?

Response: The staff was made aware, but not expeditiously, of the failure to close of two Westinghouse valves at Marshall. Based on information supplied by Westinghouse, the staff concluded that reasonable actions were being taken by Westinghouse and by known users of this type of block valve. The staff was also aware of the successful block valve retests at Marshall and of several instances in operating plants, where successful block valve operation against differential pressures had occurred. In addition, Westinghouse had provided assurance that the valves would be modified before being placed into service on new plants and had sent out advisory letters as discussed above. The staff had under preparation, at the time of your letter, a generic notification that was issued as Bulletin 81-02 to licensees and construction permit holders to provide additional assurance that any safety related applications of the Westinghouse gate valves would be properly evaluated. On these bases, the staff had decided that the failure of the Westinghouse valves to close did not meet the Commission's requirement for reporting of "relevant and material new information" to be reported to licensing boards.

In retrospect, Board notification should have been initiated when it was determined that some operating plants had in service valves which had failed testing even though such testing may not have represented actual service conditions. Since there are no sitting boards for operating plants where PORV block valve failures are relevant, none has been subsequently notified. With regard to plants under construction the staff has issued SERs on only one plant that has the affected PORV block valves; however, that plant has already modified the valve operator. The staff will address the PORV block valve issue in the SERs for other plants under construction having the PORV block valves. Therefore, separate Board notifications for these plants will not be required.

In addition to the above questions, you closed your letter of March 11, 1981 by specifically urging the Commission to take the following four actions:

(1) Order the immediate modification or replacement of all valves failing EPRI tests.

(2) Direct the staff to report the EPRI results to Licensing Boards.

(3) Direct the staff to undertake responsible oversight of the EPRI program, to obtain all test results expeditiously, and to inform the Commissioners of significant developments.

(4) In light of the test failures, move the July 1, 1982 deadline for "verification" of block valve functionability to the earliest possible date.

Items 2 and 4 we have responded to earlier in this letter.

As for Item 1, all operating plants that have block valves of the type that failed the EPRI tests have modified the valves to ensure their operability. Additionally, all plants have previously analyzed the consequence of a stuck open PORV, and show acceptable plant response. Bulletin 81-02 requires all plants under construction to either modify their block valves to ensure their operability, or replace them with suitably qualified valves. The staff will continue to monitor the progress of the qualification program and any adverse test data will continue to be evaluated on a case-by-case basis. The staff will take appropriate action to assure that any required modifications are made in a timely manner. Such action would depend, of course, on the relation of the test conditions and test results to actual reactor applications.

Regarding Item 3, we have concluded that the present monitoring system established by the staff is adequate to effect essential oversight of the EPRI program provided that better communication is maintained among the staff elements involved. For this monitoring, the licensing staff maintains overall cognizance of the program from a licensing perspective supported by detailed technical monitoring performed by the Office of Research and its contractor and by assistance from IE.

I hope that this letter adequately responds to the concerns raised in your March 11, 1981 letter. If you have any further questions on any of these matters, please do not hesitate to contact us.

Sincerely,

(Signed) William J. Dircks

William J. Dircks
Executive Director for Operations

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

OFFICE OF THE SECRETARY

August 28, 1981

MEMORANDUM FOR: William J. Dircks, Executive Director for Operations

FROM: Samuel J. Chilk, Secretary

SUBJECT: CR-81-058, RESPONSE TO MARCH 11, 1981 LETTER FROM ROBERT POLLARD, UCS, CONCERNING TESTING OF PORV BLOCK VALVES

This is to inform you that the Commission has approved the enclosed subject letter for your signature.

Upon signature and dispatch, please provide a copy to the Office of the Secretary.

Enclosure:
As stated

cc: Chairman Palladino
Commissioner Gilinsky
Commissioner Bradford
Commissioner Ahearne
Commissioner Roberts
OPE
OGC
OCA
OPA
OIA

ATTACHMENT M

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

OCT 1 9 1981

MEMORANDUM FOR: R. J. Bosnak, Chief
Mechanical Engineering Branch

FROM: F. C. Cherny, Section Leader
Mechanical Engineering Branch

SUBJECT: OCTOBER 2, 1981 MEETING WITH EPRI AND PWR UTILITIES TO REVIEW TEST RESULTS TO DATE AND PROPOSED SCHEDULE FOR COMPLETION OF BALANCE OF TESTS IN RESPONSE TO ITEM II.D.1, NUREG-0737

A meeting was held with PWR utilities and EPRI to discuss results of PORV and safety valve tests completed as of 10/1/81.

The meeting began with a brief review of how the test valves and valve test conditions were selected and how valve test results are being disseminated to PWR utilities, NSSS vendors and the NRC.

This was followed by detailed discussion of available valve test results. All PORV tests were completed as of August 26, 1981. As a group of valves, overall PORV valve performance has been satisfactory. Only a few anomalies were noted in the tests. These involve primarily a few valves that failed to close on demand after exposure to a variable temperature loop seal environment. Details of all the PORV test results can be found in the attached slides from the meeting.

The safety valve test results to date were also discussed. Through September 25, 1981, a total of five safety valve/inlet piping configurations were tested. Tests have been performed under steam, loop seal, transition (steam to water) and water (subcooled) conditions. Valve performance, as has been widely reported, has been observed to be dependent on the test fluid conditions, the test loop piping configuration and valve ring adjustment. Details of the specific tests are summarized in the attached slides from the meeting. In general, these test results indicate that safety valves installed in piping configurations representative of those in PWR operating plants i.e., relatively long inlet pipes, for steam relief, need to be adjusted for a relatively long blowdown, 10-20%, to achieve full lift at a relatively low overpressure, 6% or lower, and operate in a stable, non-chattering, manner under a wide variety of transient ramp rates and discharge system backpressures.

For steam to water transition and the continuous sub-cooled water tests, the safety valves have exhibited some tendency to chatter on water with the likelyhood of chattering probably increasing with greater sub-cooling of the water.

The attached slides from the meeting contain detailed safety valve test results for tests completed through September 25, 1981.

Although there is no slide from the meeting that discusses it, it was noted that testing of the Target Rock 69C pilot actuated safety valve was started the week of September 28. This valve currently is only used in one plant - Beaver Valley 1 and is used with a loop seal inlet piping configuration. Plans are to test this valve only with the inlet loop seal piping of the same dimensions as at Beaver Valley.

It was noted at the meeting that several steam tests of the Target Rock valve had been completed and in general it was performing within the originally established safety valve "screening criteria".

It should be noted that EPRI did briefly review the philosophy behind the establishment of the test screening criteria for both PORV's and safety valves. Specific screening criteria were developed, partially in response to encouragement from NRC. The key point emphasized at the meeting, however, is that these criteria were developed as a general guideline for valve operation and to expedite disemination of the information when the criteria are not met.. The criteria do not represent specific pass/fail criteria for the test valve.

After the discussion of test results, the revised schedule for completion of safety valve testing was presented. At the present time, it appears that testing can be completed by 12/31/81. This is somewhat earlier than reported to the Commission in SECY-81-491. The Commission paper provided an estimated test completion date of March 31, 1982.

Finally the EPRI schedule for submittal to NRC of the various test program output reports was presented. The schedule is provided in the attached slides. Although not on the slides, it was reconfirmed that the EPRI PORV Block Valve Report would be sent to NRC April 1, 1982.

In terms of plant specific documentation, the PWR Owners Group proposed the following submittal dates:

1) Plant specific submittals confirming adequacy of safety and relief valves based on licensee/applicant preliminary review of generic program test results.

   October 1, 1982

2) Plant specific reports for safety and relief valve qualification.

   January 1, 1983

3) Plant specific submittals for safety and relief valve discharge piping and support evaluations.

   July 1, 1983

The NRC staff pointed out that these dates were later than the July 1, 1982 schedule extension date recently approved by the Commission for submittal of all plant specific documentation. Additionally, the staff noted that letters, already signed by D. Eisenhut, were being mailed, probably the week of October 5, 1981 to all PWR Licensees and Applicants informing them of the extended July 1, 1982 submittal date.

It was noted by the Owners Group that some utilities could probably easily meet the July 1, 1982 date and some could not depending upon the amount of work, additional analyses etc., that might be required, after receipt of generic program results, to justify plant specific adequacy.

The staff stated that at this juncture it would be better if each utility would individually respond to the forthcoming Eisenhut letter, hopefully acknowledging intent to provide a final plant specific submittal by July 1, 1982 or proposing an alternative date and stating why the extra time was needed.

The Owners Group agreed to the individual response approach with EPRI to provide assistance to the utilities in providing their individual responses.

F. C. Cherny

F. C. Cherny, Section Leader
Mechanical Engineering Branch
Division of Engineering

Attachment:
As stated

cc w/att:
See next page

ATTENDANCE

PWR Owners Meeting 10/2/81

Franck C. Cherny	NRC/DE/MEB
Harold I. Gregg	NRC/RES/M-SEB
James A. Hunter	INEL/EG&G
Earl J. Brown	NRC/AEOD
Ed Hemminger	NRC/MEB
Bob LaGrange	NRC/EQB
James H. Correa	GPU Nuclear
Frank Highland	Commonwealth Edison
Terry Edwards	Duke Power
Joe Blanco	South California Edison
Bill Jones	Yankee Atomic
Jim Scott	PSE&G
T. M. Su	NRC/GIB
A. Marchese	NRC/NRR
Dick Verduin	Florida Power & Light
Edward M. Burns	Westinghouse
Anthony Wheeler	EPRI
Lep Chin	TVA
H. W. McCurdy	MPR Associates
Ray Crawford	SAI/CEC
Spyros Traiforos	Bechtel
Walton Jensen	NRC
Michael D. Quinton	SCE&G
T. Satyan-Sharma	Am. Electric Power
Kenneth Daledda	American Electric Power
D. T. Clift	TVA
M. A. Schoppman	NUS Corp.
Steve Weismantel	Combustion Engineering
G. J. Kanupka	Combustion Engineering
L. R. Cartin	B&W
R. C. Kellogg	OPPD
T. L. Palterson	OPPD
R. L. Kinsaul	Southern Co. Services
J. R. Crane	Southern Co. Services
Ted Harlan	NSP
T. F. Timmons	Westinghouse
Kulin D. Desai	NRC/DE/EQB
A. B. Bennett	NRC/IE
D. W. Spediell	VEPCO
John Carey	EPRI
John Hosler	EPRI
Tom Auble	EPRI
Warren Bilanin	EPRI
David Hoffman	Consumers Power

EPRI/PWR SAFETY AND RELIEF VALVE TEST PROGRAM

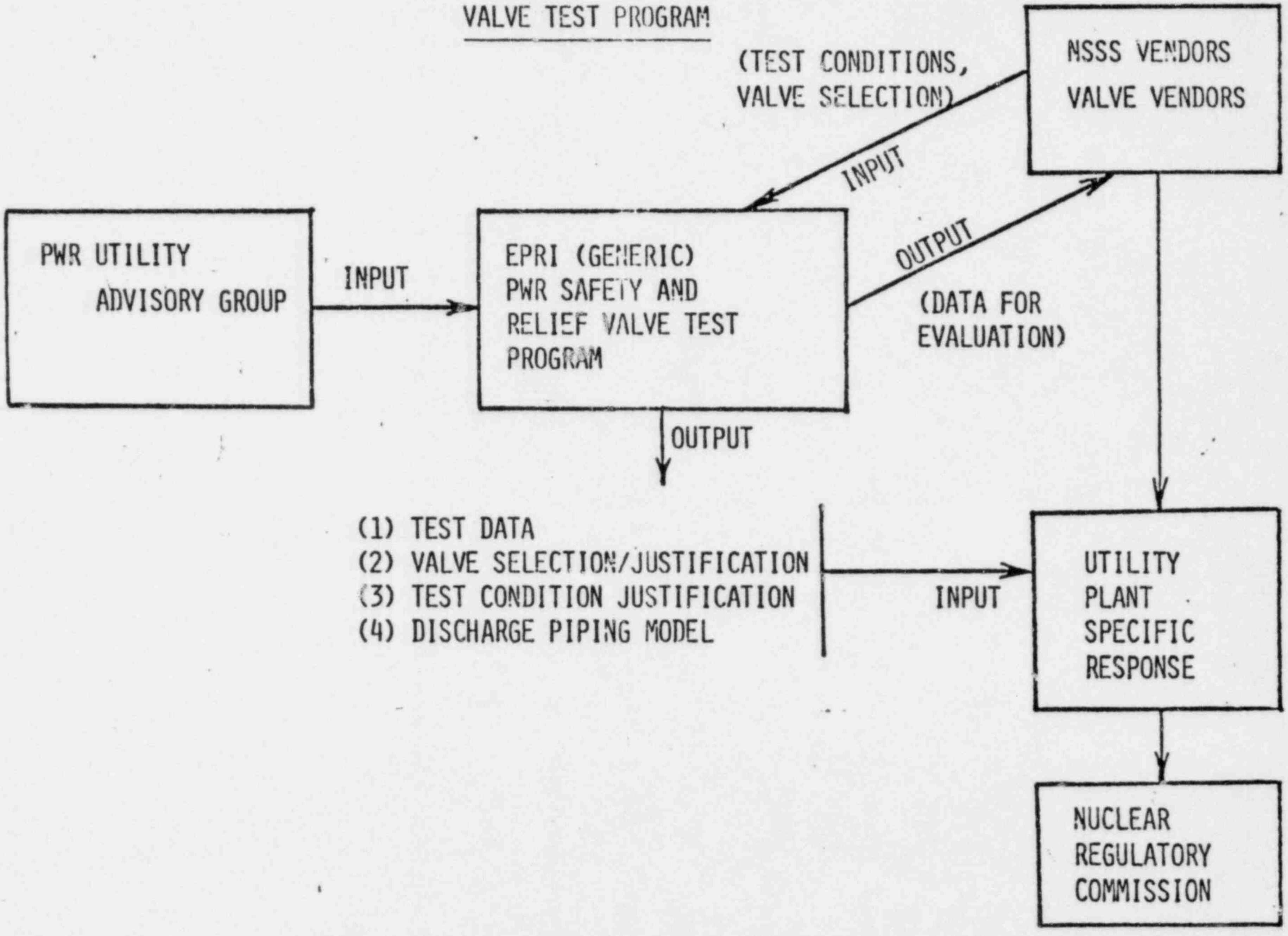
PWR UTILITY/NRC PROGRAM STATUS MEETING

BETHESDA, MARYLAND

OCTOBER 2, 1981

9:00	INTRODUCTION - PROGRAM OVERVIEW - AGENDA	HOFFMAN
9:30	TEST CONDITION . STIFICATION	HOSLER
9:45	RELIEF AND SAFET ALVE SELECTION	AUBLE
10:00	RELIEF VALVE TEST RESULTS	HOSLER
10:30	SAFETY VALVE TEST RESULTS	AUBLE
11:30	ANALYTICAL EFFORTS	WHEELER
11:45	SCHEDULE FOR SUBMITTAL OF PROGRAM OUTPUTS	BILANIN

PWR UTILITY/INDUSTRY VALVE TEST PROGRAM



TEST CONDITION DEFINITION/JUSTIFICATION

- BACKGROUND
- CRITERIA FOR DETERMINING PLANT/TEST CONDITIONS
- SUMMARY OF TEST CONDITIONS
  - RELIEF VALVES
  - SAFETY VALVES
- PLANT/TEST CONDITION JUSTIFICATION REPORTS STATUS

BACKGROUND

- INITIAL SELECTION OF ENVELOPING TEST CONDITIONS MADE IN JULY 80

  - BASED ON DISCUSSIONS WITH PWR NSSS VENDORS

  - USED AS A BASIS FOR SPECIFYING TEST FACILITY DESIGN REQUIREMENTS

- PWR NSSS VENDORS CONTRACTED TO DEVELOP DETAILED PLANT CONDITION JUSTIFICATION REPORTS

- EPRI UTILIZES PWR NSSS VENDOR REPORTS AS BASIS FOR DEVELOPMENT OF A TEST CONDITIONS JUSTIFICATION REPORT

- EPRI AND VENDOR REPORTS TO BE SUBMITTED TO NRC AND REFERENCED BY UTILITIES IN THEIR RESPONSES TO NUREG 0737 ITEM IID1

ANALYTICAL EFFORTS

- DISCHARGE PIPING LOAD MODELING
- COUPLED PIPING - SAFETY VALVE DYNAMICS MODEL
- INLET PIPING CLASSIFICATION ANALYSIS - TEST DATA APPLICABILITY TO PLANTS

DISCHARGE PIPING LOAD MODEL

OBJECTIVE

- ESTABLISH VERIFIED COMPUTER MODELS TO PREDICT DYNAMIC LOADS ON PIPING

BACKGROUND

- PLANT PIPING SYSTEMS TOO DIVERSE TO PERMIT PROTOTYPIC TESTING
- EXISTING STRUCTURAL MODELS CODES CONSIDERED SIMILAR AND ADEQUATE
- VERIFICATION OF THERMAL-HYDRAULIC CODE CONSIDERED PRIMARY REQUIREMENT

APPROACH

- EVALUATE SEVERAL EXISTING THERMAL-HYDRAULIC CODES
- SELECT MOST PROMISING
- PERFORM DETAILED VERIFICATION USING PIPING LOAD TEST DATA

STATUS

- RELAP 5-MOD 1 SELECTED FOR PRIMARY VERIFICATION EFFORT
- SOLA-NET TO BE VERIFIED ALSO AS ALTERNATIVE
- RELAP 5 VERIFICATION PACKAGE AVAILABLE APRIL 1, 1982

COUPLED PIPING - SAFETY VALVE DYNAMICS MODEL

OBJECTIVE

- DEVELOP A COMPUTER MODEL TO PREDICT DYNAMIC BEHAVIOR OF SPRING—LOADED SAFETY VALVES TO ASSIST IN UNDERSTANDING OBSERVED VALVE PERFORMANCE

BACKGROUND

- VALVE OPERATION MAY BE DEPENDENT ON COUPLING BETWEEN THE VALVE SPRING-MASS SYSTEM AND THE FLUID DYNAMICS OF ATTACHED PIPING
- NEED TOOL TO BETTER UNDERSTAND PHENOMENA

APPROACH

- CREATE SAFETY VALVE DYNAMIC MODEL
- COUPLE TO EXISTING PIPING FLUID-MECHANICS CODE
- COMPARE AGAINST DATA

STATUS

- PRELIMINARY CODE WORKING
- WORK IN PROGRESS TO IMPROVE MODEL AND EVALUATE AGAINST DATA

INLET PIPING CLASSIFICATION ANALYSIS

OBJECTIVE

- USE SIMPLIFIED ANALYSIS TO CLASSIFY PLANTS RELATIVE TO TESTED CONFIGURATION
- USE TO SPECIFY APPLICABILITY OF TEST RESULTS

APPROACH

- COMPUTE EFFECTIVE LENGTH OF THE INLET PIPING
- COMPUTE EFFECTIVE LENGTH OF PLANT PIPING
- PLANT IS COVERED IF ITS EFFECTIVE LENGTH IS SHORTER THAN TEST EFFECTIVE LENGTH

STATUS

- METHOD COMPLETE
- BEING EVALUATED AGAINST TEST DATA

SCHEDULE FOR SUBMITTAL OF EPRI PROGRAM REPORTS

A. REQUIRED FOR PLANT-SPECIFIC RESPONSE TO NUREG 0737, ITEM IID1 A

REPORT	SUBMITTAL DATE
1. INTERIM DATA REPORT	JULY 1, 1981
2. INTERIM DATA REPORT, REVISION 1	DECEMBER 1, 1981
3. INTERIM DATA REPORT, REVISION 2	APRIL 1, 1982
4. VALVE SELECTION/JUSTIFICATION REPORT	DECEMBER 1, 1981
5. TEST CONDITION JUSTIFICATION REPORT	APRIL 1, 1982
6. DISCHARGE PIPING LOAD MODEL REPORT	APRIL 1, 1982

B. SUPPLEMENTARY PROGRAM INFORMATION

REPORT	SUBMITTAL DATE
1. MARSHALL TEST REPORT	OCTOBER 1, 1981
2. WYLE TEST REPORTS (PHASES II & III)	JANUARY 1, 1982
3. COMBUSTION ENGINEERING TEST REPORT	JULY 1, 1982

CRITERIA FOR DETERMINING PLANT/TEST CONDITIONS

- PLANT CONDITIONS DEVELOPED BASED ON CONSIDERATION OF:
  - FSAR EVENTS
  - EXTENDED HIGH PRESSURE LIQUID INJECTION EVENTS
  - COLD OVERPRESSURIZATION EVENTS

- TEST CONDITIONS SELECTED TO ENVELOP EXPECTED PLANT CONDITIONS
  - SINCE TEST CONDITIONS ARE ENVELOPING THEY DO NOT APPLY DIRECTLY TO ALL PLANTS, I.E., EXPECTED CONDITIONS IN MANY PLANTS ARE LESS SEVERE THAN THOSE TESTED

SUMMARY OF TEST CONDITIONS

- RELIEF VALVES

  - STEAM 2465 - 2750 PSIA

  - PRELOAD (STEAM) 2465-2750 PSIA, BENDING MOMENTS FROM 36,000 TO 43,000 IN-LB

  - WATER 665 PSIA, 100-450°F
    2465-2750 PSIA, 450*-650°F

  - STEAM TO WATER TRANSITION 2465-2750 PSIA, 650°F WATER

  - NITROGEN TO WATER TRANSITION** 1500 PSIA, 300°F WATER

  - WATER SEAL SIMULATION** 2465-2750 PSIA, TRANSITION FROM 100°F TO 650°F WATER

*SELECTED VALVES TESTED AT TEMPS AS LOW AS 250°F

**ONLY PERFORMED ON VALVES TO WHICH CONDITION APPLIES

SUMMARY OF TEST CONDITIONS (CONT'D)

- SAFETY VALVES (OPENING SET POINT ~2500 PSIG)
  - STEAM
    - PEAK PRESSURES TO ~2750 PSIA
    - PRESSURIZATION RATES FROM 5 TO 250 PSI/SEC
  - LOOP SEAL
    - PEAK PRESSURES TO ~2750 PSIA
    - PRESSURIZATION RATES FROM 5 TO 250 PSI/SEC
  - TRANSITION (STEAM TO 650°F WATER)
    - LIQUID SURGE RATES FROM 100 TO 3000 GPM
  - WATER (400, 550, 650°F)
    - SURGE RATES FROM 100 TO 3000 GPM (650°F)
    - SURGE RATES FROM 800 TO 3000 GPM (400, 550°F)
  - INLET PIPING CONFIGURATIONS
    - TYPICAL OF PWR INSTALLATIONS
  - DISCHARGE PIPING EFFECTS
    - BACKPRESSURES
      - 200 TO 700 PSIA
    - TEST SYSTEM INDUCES SUBSTANTIAL FORCES, MOMENTS, AND ACCELERATIONS ON VALVE DURING ACTUATION
    - DATA UNDER EVALUATION TO ASSESS NEED FOR ADDITIONAL STATIC LOADING TESTS
    - VALVE STATIC LOADING DEVICE AVAILABLE IF REQUIRED

PLANT/TEST CONDITION JUSTIFICATION REPORTS
STATUS

- PWR NSSS VENDOR REPORTS (PLANT CONDITION JUSTIFICATION)
  - DRAFTS UNDER REVIEW
- EPRI TEST CONDITION JUSTIFICATION REPORT
  - IN PROGRESS
- NSSS VENDOR AND EPRI REPORTS TO BE SUBMITTED 4/82

RELIEF AND SAFETY VALVE SELECTION AND JUSTIFICATION

THE PURPOSE OF THE VALVE SELECTION/JUSTIFICATION REPORT IS TO SUPPORT UTILITY REQUIRED RESPONSES TO NUREGS 0578/0737

- ONLY SELECTED VALVES ARE BEING TESTED
  - NINE SAFETY VALVES
  - TEN PORVs
- ALL UTILITIES MUST SUBSTANTIATE THE FUNCTIONABILITY OF THEIR RELIEF AND SAFETY VALVES
- THE UTILITIES MUST USE TEST RESULTS WITH THE SELECTION/JUSTIFICATION REPORT TO SHOW FUNCTIONABILITY
- THE EPRI JUSTIFICATION REPORT ADDRESSES DESIGN DIFFERENCES BETWEEN VALVES
- THE REPORT SHOWS THAT THE SELECTED TEST VALVES REPRESENT ALL S/RV VALVES

THE 19 VALVES SELECTED FOR TEST WERE SELECTED TO PERMIT APPLICATION OF THE TEST RESULTS TO ALL PWR S/RV's

- EACH S/RV TYPE (I.E., PILOT OPERATED VALVES, SPRING LOADED VALVES, ELECTRICALLY OPERATED VALVES, AIR OPERATED VALVES) WAS SELECTED FOR TEST.

- EACH VALVE MANUFACTURER'S S/RV DESIGN WITHIN A TYPE WAS SELECTED FOR TEST

- VALVE MANUFACTURER'S DESIGN VARIATIONS IN THE VALVE INTERNAL GUIDING CONFIGURATION AND/OR MATERIAL COMBINATION WERE INCORPORATED INTO THE TEST PROGRAM (E.G., COPES-VULCAN 17-4/17-4 CAGE/DISC COMBINATION AND ALSO 316 W/ STELLITE DISC/CAGE COMBINATION).

- VARIATIONS IN SAFETY VALVE SIZES ARE BEING TESTED TO PERMIT EXTRAPOLATION ACROSS AN ENTIRE VALVE LINE (E.G., CROSBY 3K6, 6M8 TESTING).

- JUSTIFICATION REPORTS HAVE BEEN DEVELOPED BY EACH MANUFACTURER TO APPLY SELECTED VALVE TEST RESULTS TO ALL OF THEIR S/RV DESIGNS SUPPLIED TO PWR PLANTS

THE KEY DATA IN THE REPORT WILL BE THE VALVE VENDOR'S JUSTIFICATION REPORTS. THE REPORTS INCLUDE:

- DETAILED DESIGN VARIATIONS BETWEEN VALVES DESCRIBED E.G. MATERIALS, SIZES, CONFIGURATIONS
- TECHNICAL EVALUATION OF EACH VARIATION
- A LIST OF SELECTED TEST VALVES
- EVALUATION SHOWING THAT THE SELECTED VALVES COVER THE VARIATIONS
- SUBMITTAL OF THE ABOVE IN A REPORT FOR PUBLICATION BY EPRI FOR SUBMITTAL TO THE UTILITIES

EPRI S/RV TEST PROGRAM

SELECTED VALVES, VALVES REPRESENTED, VALVE DISTRIBUTION IN PARTICIPATING PWR'S

Pressurizer Safety Valves

Valve Manufacturer	Selected Test Valves				Valves Represented				No. Pl
	Model No.	Size			Model No.	Size			
		Inlet	Orifice	Outlet		Inlet	Orifice	Outlet	
Crosby Valve & Gage Company	HB-BP-86	3	K	6	HB-BP-86	3	K	6(smallest)	
		6	M	6		3	K2	6	
		6	N	8		4	K2	6	
						6	K2	6	
						4	M1	6	
						6	M1	6	
						4	M	6	
						6	M	6	
						6	N	8(largest)	
Dresser Industries	31739A	2.5	No.3	6	31709KA	2.5	K	6(smallest)	
	31709NA	6	N	8	31739A	2.5	No.3	6	
					31749A	3	No.4	6	
					31759A	3	No.5	6	
					31709NA	6	N	8(largest)	
Target Rock Corp.	69C	6	$3.513 in^2$	6	69C	6	$3.513 in^2$	6	
								Total	1

Note: Inlet and outlet sizes are nominal pipe sizes in inches

EPRI S/RV TEST PROGRAM

SELECTED VALVES, VALVES REPRESENTED, VALVE DISTRIBUTION IN PARTICIPATING PWR'S

Power Operated Relief Valves

Valve Manufacturer	Selected Test Valves Model No.	Size	Valves Represented Model No.	Size	No. of Plants
Control Components, Inc.	Drag Valve	3" NPS	Drag Valve	3" NPS	4
Copes-Vulcan, Inc.	Globe D-100-160 with 17-4PH cage and plug	3" NPS	Globe D-100-160 with 17-4PH cage and plug	2" NPS	13
	Globe D-100-160 with 316 w/stellite plug and 17-4PH cage	3" NPS	Globe D-100-160 with 316 w/stellite plug and 17-4PH cage	3" NPS	23
			Globe D-100-160 with 316 w/stellite plug & Haynes #25 cage	3" NPS	2
Crosby Valve & Gage Co.	HPV-SN	1 3/8" bore	HPV-SN	1 3/8" bore	2
				1 1/2" bore	1

Note: NPS is the valve nominal pipe size

EPRI S/RV TEST PROGRAM

SELECTED TEST VALVES, VALVES REPRESENTED, VALVE DISTRIBUTION IN PARTICIPATING PWR'S

Power Operated Relief Valves

Valve Manufacturer	Selected Test Valves		Valves Represented		
	Model No.	Size	Model No.	Size	No. of
Dresser Industries	31533VX-30	1 5/16" bore	31533VX-30	1 3/32" bore	6
				1 5/32" bore	4
				1 5/16" bore	10
			31533VX	1 3/8" bore	1
Fisher Controls Co.	SS-103-SS-95	3" NPS	SS-103-SS-95	3" NPS	3
Garrett Pneumatic Systems Division	Straight Through	3" inlet 6" outlet	Angle	3" inlet 8" outlet	1
			Straight Through	3" inlet 6" outlet	6
Masoneilan	20,000 Series	2" NPS	20,000 Series	2" NPS	9
Muesco Controls, Inc.	70-18-9 DRTX	2" NPS	70-18-9 DRTX	2" NPS	1
Target Rock Corp.	80X-006	2½" inlet 4" outlet	80X-006	2½" inlet 4" outlet	2
				Total	88

Note: NPS is the valve nominal pipe size

VALVE SELECTION/JUSTIFICATION REPORT STATUS

- THE DRAFT VALVE SELECTION/JUSTIFICATION REPORT
  - COMPLETE
- UTILITY COMMENTS ARE BEING OBTAINED FOR THE FINAL REPORT
- A FINAL REPORT IS SCHEDULED FOR SUBMITTAL DECEMBER 1, 1981

RELIEF VALVE TEST RESULTS

- INTRODUCTION
- MARSHALL STEAM STATION TESTS
- WYLE PHASE II TESTS
- WYLE PHASE III TESTS
- SUMMARY

INTRODUCTION

- ALL PRESSURIZER RELIEF VALVE TESTING IN RESPONSE TO NUREG 0737 ITEM IID1 COMPLETE
- TESTING PERFORMED IN THREE PHASES
  - EPRI/MARSHALL STEAM STATION TESTING
  - EPRI/WYLE (NORCO) PHASE II TESTING
  - EPRY/WYLE (NORCO) PHASE III TESTING
- FULL RANGE OF EXPECTED CONDITIONS TESTED
  - STEAM
  - STEAM (EXTERNAL BENDING MOMENT APPLIED)
  - WATER (RANGE OF SUBCOOLINGS)
  - TRANSITION
    - STEAM TO WATER (ALL VALVES)
    - N_2 TO WATER (CV-316 ONLY)
    - COLD TO HOT WATER (WATER SEAL SIMULATION) (ALL BUT MUESCO)
- OVERALL OBSERVED RELIEF VALVE PERFORMANCE WAS EXCELLENT

EPRI/MARSHALL TESTING

- ALL 10 RELIEF VALVES TESTED
- TESTS PERFORMED UNDER FULL PRESSURE STEAM FLOW CONDITIONS
- A MINIMUM OF 10 EVALUATION TEST CYCLES ON EACH VALVE DESIGN
- SEVERAL SUPPLEMENTAL TESTS PERFORMED DURING CHECKOUT AND/OR AS PART OF BLOCK VALVE TESTING
- DURING ALL EVALUATION TESTS ALL RELIEF VALVES OPENED AND CLOSED ON DEMAND AND SUSTAINED NO DAMAGE WHICH WOULD AFFECT OPERABILITY
  - FISHER CONTROLS VALVE OPENED SLOWLY (13-17 SEC) DUE TO SPRING TENSION SETTING GREATER THAN IN-PLANT SETTINGS

EPRI/MARSHALL TESTING (CONT'D)

- DURING ALL SUPPLEMENTARY TESTS, THE VALVES OPENED AND CLOSED ON DEMAND AND SUSTAINED NO DAMAGE WHICH WOULD AFFECT OPERATION EXCEPT:

  - THE FISHER VALVE DID NOT CLOSE FULLY ON DEMAND DURING SOME TESTS (ALWAYS CLOSED TO WITHIN 4% OF THE FULLY CLOSED POSITION) PROBLEM ATTRIBUTED TO INCORRECT INTERNALS SUPPLIED BY MANUFACTURER

  - CROSBY RELIEF VALVE FAILED TO OPEN ON DEMAND DURING SOME TESTS

    - NOTE: FOLLOWING THESE TESTS, IT WAS DISCOVERED THAT THE VALVE HAD AN INCORRECTLY MACHINED BELLOWS ASSEMBLY. IT WAS ALSO OBSERVED THAT THE BELLOWS INNER DIAMETER WELD HAD FAILED CIRCUMFERENTIALLY. (EVALUATION TESTS PERFORMED WITH CORRECTED BELLOWS ASSEMBLY)

EPRI/MARSHALL TESTING (CONT'D)

- COPIES-VULCAN VALVE UTILIZING A 17-4 PH PLUG AND CAGE FAILED TO CLOSE FULLY ON DEMAND (ALWAYS CLOSED TO WITHIN 13% OF FULL CLOSURE)

- IN ADDITION, THE FOLLOWING WAS OBSERVED:

  - DURING MASONEILAN VALVE TESTING, AIR SUPPLY PRESSURE HAD TO BE INCREASED ABOVE MANUFACTURER'S RECOMMENDED VALUE OF 55 PSIG TO ~60 PSIG TO BRING OPENING TIMES BELOW 2 SECONDS

  - BELLOWS CRACKED DURING DRESSER TESTING - NO AFFECT ON VALVE OPERATION OBSERVED.

EPRI/WYLE PHASE II TESTING

- TWO RELIEF VALVE DESIGNS TESTED
  - DRESSER ELECTROMATIC
  - CROSBY PRESSUREMATIC
- A TOTAL OF 11 TESTS PERFORMED UNDER STEAM AND WATER CONDITIONS
- BOTH VALVES OPENED AND CLOSED ON DEMAND WITH NO DAMAGE WHICH WOULD AFFECT OPERATION
  - CROSBY BELLOWS FOUND TO LEAK AFTER TESTING WITH NO EFFECT ON VALVE OPERATION

SUPPLEMENTARY INFORMATION:

- THE CROSBY VALVE WAS RECEIVED FROM MANUFACTURER WITH AN INCORRECTLY MACHINED BELLOWS - BELLOWS REPLACED PRIOR TO TESTING

EPRI/WYLE PHASE III TESTING

- ALL RELIEF VALVE TESTS COMPLETED 8-26-81

- A TOTAL OF 106 TESTS WERE PERFORMED ON 10 PORVs

- TESTS PERFORMED UNDER

  - STEAM
  - STEAM/PRELOAD
  - WATER (RANGE OF SUBCOOLINGS)
  - TRANSITION
    - STEAM TO WATER (ALL VALVES)
    - N_2 TO WATER (CV-316 ONLY)
    - WATER SEAL SIMULATION (ALL BUT MUESCO)

- OVERALL, VALVE PERFORMANCE OBSERVED WAS EXCELLENT

EPRI/WYLE PHASE III

PORV	RESULTS SUMMARY
TARGET ROCK	-OPENED/CLOSED ON DEMAND DURING ALL TESTS EXCEPT UNDER WATER SEAL SIMULATED CONDITIONS WHEN A 15 SECOND DELAY ON CLOSURE WAS OBSERVED -OBSERVED NO DAMAGE WHICH WOULD AFFECT VALVE OPERATION -OPEN/CLOSE LIMIT SWITCHES FOUND TO BE SENSITIVE TO VIBRATION
DRESSER ELECTROMATIC	-OPENED/CLOSED ON DEMAND DURING ALL TESTS EXCEPT UNDER WATER SEAL SIMULATION CONDITIONS - 40 SECOND DELAY ON CLOSURE OBSERVED DURING COLD WATER SEAL TESTING - ELEVATED WATER SEAL TEMP RESULTED IN A REDUCED CLOSURE DELAY TIME -NO DAMAGE WHICH WOULD AFFECT VALVE OPERATION OBSERVED
CROSBY PRESSURMATIC	-OPENED/CLOSED ON DEMAND DURING ALL TESTS -BELLOWS DAMAGED - NO APPARENT EFFECT ON VALVE OPERATION, NO OTHER SIGNIFICANT DAMAGE -EXHIBITED TENDENCY NOT TO SEAL AT LOW PRESSURES (BELOW 100 PSI)

EPRI/WYLE PHASE III

PORV — RESULTS SUMMARY CON'D.

CONTROL COMPONENTS

- OPENED/CLOSED ON DEMAND FOR ALL TESTS EXCEPT UNDER 650°F LIQUID CONDITIONS WHEN SPRING ONLY USED FOR CLOSURE (SIMULATED FAILURE OF NORMAL AIR SUPPLY)
- CLOSURE DELAYS OF UP TO 40 SECONDS OBSERVED UNDER 650°F LIQUID CONDITION (FAILED A
- DURING 650° LIQUID TESTS (SIMULATED FAILED AIR) VALVE ALWAYS CLOSED AT VALVE INLET PRESSURE OF ~2200 PSI
- NO DAMAGE WHICH WOULD AFFECT VALVE OPERATION OBSERVED

MASONEILAN

- OPENED/CLOSED ON DEMAND FOR ALL TESTS
- NO DAMAGE WHICH WOULD AFFECT VALVE OPERATION OBSERVED
- VALVE OPENING TIMES OBSERVED TO BE SENSITIVE TO AIR SUPPLY PRESSURE AND AIR SUPPLY TUBING SIZE
- AIR SUPPLY TUBING SIZE RECOMMENDED IN VALVE MANUAL INSUFFICIENT TO ACHIEVE VOTS IN RANGE OF 2 SECONDS
- TESTS PERFORMED WITH INCREASED AIR TUBING SIZE (1/2" vs 1/4")
- VOTS OBSERVED WITH INCREASED TUBING SIZE RANGED FROM 1.5 TO 6.4 SECONDS (AIR SUPPLY PRESSURES RANGED FROM 54±1 TO 59±1 PSIG)

EPRI/WYLE PHASE III

PORV	RESULTS SUMMARY
COPES-VULCAN (17-4PH)	- OPENED/CLOSED ON DEMAND DURING ALL TESTS - SUSTAINED NO DAMAGE WHICH WOULD AFFECT VALVE OPERATION
COPES-VULCAN (316w STELLITE)	
MUESCO CONTROLS	
FISHER CONTROLS	
GARRETT	

EPRI/WYLE PHASE III

PORV	RESULTS SUMMARY
COPES-VULCAN (17-4PH)	- OPENED/CLOSED ON DEMAND DURING ALL TESTS - SUSTAINED NO DAMAGE WHICH WOULD AFFECT VALVE OPERATION
COPES-VULCAN (316w STELLITE)	
MUESCO CONTROLS	
FISHER CONTROLS	
GARRETT	

EPRI RELIEF VALVE TESTS

SUMMARY

- ALL RELIEF VALVE TESTING IN RESPONSE TO NUREG 0737 ITEM IID1 COMPLETE

- OVERALL, PERFORMANCE OF PORVs WAS EXCELLENT

- DATA FOR ALL RELIEF VALVE TESTS WILL BE INCLUDED IN THE DECEMBER 1, 1981 REVISION OF THE EPRI/PWR SAFETY AND RELIEF VALVE PROGRAM INTERIM DATA REPORT

EPRI RELIEF VALVE TESTS

SUMMARY

- ALL RELIEF VALVE TESTING IN RESPONSE TO NUREG 0737 ITEM IID1 COMPLETE
- OVERALL, PERFORMANCE OF PORVs WAS EXCELLENT
- DATA FOR ALL RELIEF VALVE TESTS WILL BE INCLUDED IN THE DECEMBER 1, 1981 REVISION OF THE EPRI/PWR SAFETY AND RELIEF VALVE PROGRAM INTERIM DATA REPORT

EPRI PWR SAFETY AND RELIEF VALVE PROGRAM

EPRI/CE SAFETY VALVE TEST FACILITY DESCRIPTION

AND

RESULTS SUMMARY

EPRI PWR SAFETY AND RELIEF VALVE PROGRAM

EPRI/CE SAFETY VALVE TEST FACILITY DESCRIPTION

AND

RESULTS SUMMARY

THE EPRI/C-E PWR SAFETY VALVE TEST FACILITY
CAPABILITIES ARE AS FOLLOWS

MAXIMUM PRESSURE: UPSTREAM OF TEST VALVE - 3250 PSI
DOWNSTREAM OF TEST VALVE - 1000 PSI

MAXIMUM TEMPERATURE: 700°F

TANK CAPACITY: TANK 1 500 FT^3
TANK 2 1100 FT^3

MAX. OPERATING PRESSURE: 3000 PSI

TYPES OF TESTS: WATER, STEAM, STEAM TO WATER TRANSITION

FLOW RATES:

SATURATED STEAM - 600,000 LB/HR FOR 15 SEC.
- 150,000 LB/HR CONTINUOUSLY

SUBCOOLED (UP TO 200°F) OR SATURATED WATER:

- 5,500 GPM FOR 15 SEC.
- 1,000 GPM CONTINUOUSLY

THE EPRI/C-E PWR SAFETY VALVE TEST FACILITY
CAPABILITIES ARE AS FOLLOWS

MAXIMUM PRESSURE: UPSTREAM OF TEST VALVE - 3250 PSI
DOWNSTREAM OF TEST VALVE - 1000 PSI

MAXIMUM TEMPERATURE: 700°F

TANK CAPACITY: TANK 1 500 FT^3
TANK 2 1100 FT^3

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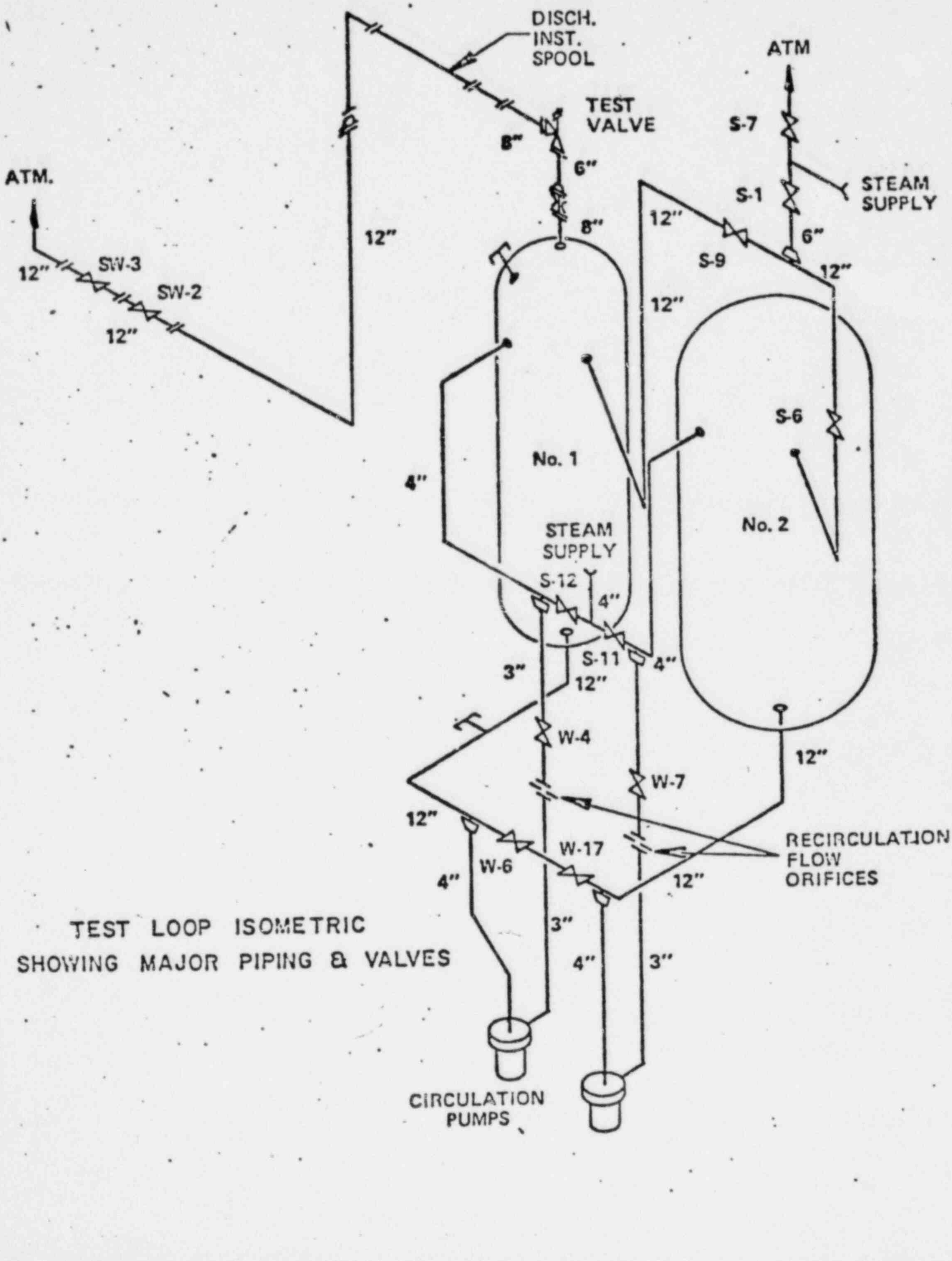
- 150,000 LB/HR CONTINUOUSLY

SUBCOOLED (UP TO 200°F) OR SATURATED WATER:

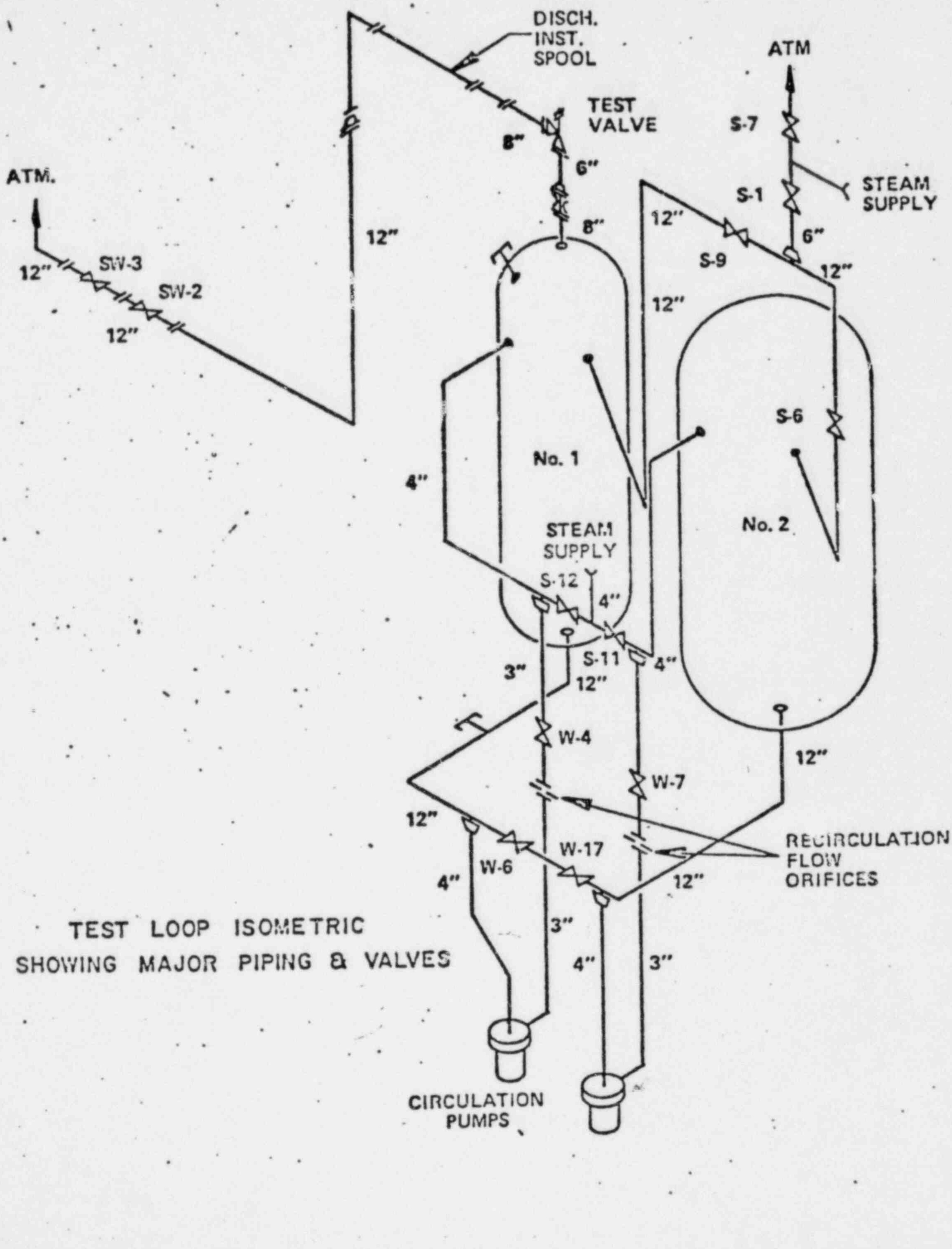
- 5,500 GPM FOR 15 SEC.
- 1,000 GPM CONTINUOUSLY

EPRI/C-E PWR SAFETY VALVE TEST FACILITY



TEST LOOP ISOMETRIC
SHOWING MAJOR PIPING & VALVES

EPRI/C-E PWR SAFETY VALVE TEST FACILITY



TEST LOOP ISOMETRIC
SHOWING MAJOR PIPING & VALVES

SAFETY VALVE TEST RESULTS

• INTRODUCTION	
• DRESSER 31709NA	LONG INLET
• DRESSER 31739A	SHORT INLET
• CROSBY 3K6	SHORT INLET
• CROSBY 3K6	LONG INLET
• DRESSER 31709NA	SHORT INLET
• SAFETY VALVE TEST SCHEDULE	
• SUMMARY	

SAFETY VALVE TEST RESULTS

- INTRODUCTION
- DRESSER 31709NA — LONG INLET
- DRESSER 31739A — SHORT INLET
- CROSBY 3K6 — SHORT INLET
- CROSBY 3K6 — LONG INLET
- DRESSER 31709NA — SHORT INLET
- SAFETY VALVE TEST SCHEDULE
- SUMMARY

INTRODUCTION

- TESTS PERFORMED ON FIVE SAFETY VALVE/INLET PIPING CONFIGURATIONS TO DATE
- ALL TESTS PERFORMED AT COMBUSTION ENGINEERING
- TESTS PERFORMED UNDER THE FOLLOWING CONDITIONS
  - STEAM
  - LOOP SEAL
  - TRANSITION (STEAM TO WATER)
  - WATER (SUBCOOLINGS)
- VALVE PERFORMANCE OBSERVED TO BE DEPENDENT ON TEST CONDITIONS

EPRI HAS ESTABLISHED SCREENING CRITERIA FOR SAFETY VALVE OPERATION:

- SCREENING CRITERIA ARE A GENERAL GUIDELINE FOR VALVE OPERATION
- THE CRITERIA WERE DEVELOPED FOR NRC GUIDANCE AT NRC REQUEST
- THE CRITERIA DO NOT REPRESENT PASS/FAIL CRITERIA FOR THE TEST VALVE

SCREENING CRITERIA

STEAM TESTING:

1. VALVE OPENS WITHIN ± 3% OF DESIGN SET PRESSURE

2. VALVE ATTAINS AT LEAST RATED LIFT AT A PRESSURE 6% ABOVE DESIGN SET PRESSURE

3. THE VALVE IS FULLY CLOSED WHEN THE INLET PRESSURE IS LESS THAN THE VALVE OPENING PRESSURE AND GREATER THAN 2250 PSIG (10% BLOWDOWN)

WATER/TRANSITION TESTING:

ITEMS 1 AND 3 OF STEAM TESTING APPLY

REVIEW OF EPRI SAFETY VALVE TESTS

SUMMARY

VALVE	INLET CONFIGURATION SHORT	INLET CONFIGURATION LONG	TEST CONDITION STEAM	TEST CONDITION WATER	TEST CONDITION TRANSITION	TEST CONDITION LOOP SEAL	NUMBER OF TESTS	DATE TESTED
DRESSER 31709NA		X	X				1	6/3
DRESSER 31739A	X		X				14	6/17 - 7/6
CROSBY 3K6	X		X	X	X		15	7/8 - 7/29
CROSBY 3K6		X	X		X	X	13	8/7 - 9/4
DRESSER 31709NA	X		X	X	X		11	9/15 - 9/25

REVIEW OF EPRI SAFETY VALVE TESTS

SUMMARY

VALVE	INLET CONFIGURATION SHORT	INLET CONFIGURATION LONG	TEST CONDITION STEAM	TEST CONDITION WATER	TEST CONDITION TRANSITION	TEST CONDITION LOOP SEAL	NUMBER OF TESTS	DATE TESTED
DRESSER 31709NA		X	X				1	6/3
DRESSER 31739A	X		X				14	6/17 - 7/6
CROSBY 3K6	X		X	X	X		15	7/8 - 7/29
CROSBY 3K6		X	X		X	X	13	8/7 - 9/4
DRESSER 31709NA	X		X	X	X		11	9/15 - 9/25

EPRI/CE SAFETY VALVE TESTS
DRESSER 31709NA SAFETY VALVE

DATA REVIEW

(LONG INLET CONFIGURATION)

SUMMARY OF DRESSER 31709NA SAFETY VALVE

TEST (LONG INLET PIPING CONFIGURATION)

- SINGLE TEST PERFORMED
  - STEAM
  - HIGH PRESSURIZATION RATE (425 PSI/SEC)/ HIGH BACKPRESSURE
  - LONG INLET PIPING (DRAINED LOOP SEAL)
    - 15 FT LONG
    - 6" DOUBLE X PIPE
- VALVE OPENED AT 2488 PSIA AND BEGAN CHATTERING
- CHATTER PERSISTED FOR NEARLY THE ENTIRE TRANSIENT DURATION (~ 120 SECONDS) AT A FREQUENCY OF 30-40 HZ
- PEAK PRESSURE REACHED ≈ 2680 PSIA
- VALVE RE-CLOSED AT 2010 PSIA
- SYSTEM REPRESSURIZED AND VALVE REOPENED AT ~2150 PSIA
- VALVE AGAIN CHATTERED DURING SECOND OPENING AND CLOSED WITH MINIMAL LEAKAGE
- UPON INTERNAL INSPECTION, GALLING OF GUIDING SURFACES OBSERVED; SEVERAL INTERNAL PARTS DAMAGED

EPRI/CE SAFETY VALVE TESTS

DRESSER 31739A SAFETY VALVE

DATA REVIEW

(SHORT INLET CONFIGURATION)

EPRI/CE SAFETY VALVE TESTS

DRESSER 31739A SAFETY VALVE

TEST SUMMARY

- A TOTAL OF FOURTEEN TESTS WERE PERFORMED, ALL TESTS WERE SHORT PIPE STEAM TESTS
- ESSENTIALLY, TWO GROUPS OF TESTS WERE PERFORMED:

GROUP 1 TESTING

- FOUR TESTS WITH DRESSER RECOMMENDED (AS INSTALLED) RING SETTINGS WITH RAMP RATE AND BACKPRESSURE CHANGES (LOW AND INTERMEDIATE) PER THE EPRI TEST MATRIX WERE PERFORMED.
- THE VALVE OPENED WITH ± 3% OF SET PRESSURE (2515 PSIA).
- THE VALVE OPERATION WAS STABLE.
- THE VALVE DID NOT ACHIEVE RATED LIFT AT EITHER 3% (2590 PSIA) OR 6% (2666 PSIA) ACCUMULATION PRESSURES.
- INCREASING BACKPRESSURE REDUCED VALVE LIFT
- BLOWDOWN RANGED FROM 4 - 7% BELOW THE VALVE SET POINT PRESSURE

EPRI/CE SAFETY VALVE TESTS

DRESSER 31739A SAFETY VALVE

TEST SUMMARY
(CONTINUED)

GROUP 2 TESTING

- IN THE NEXT TEN TESTS, TEST MATRIX CONDITIONS WERE REPEATED WITH RING ADJUSTMENTS.
- THE TESTS WERE PERFORMED TO ACHIEVE RATED LIFT AND MAINTAIN IT AT HIGH AND LOW BACKPRESSURES.
- RATED LIFT WAS ACHIEVED AND MAINTAINED FOR BOTH HIGH AND LOW BACKPRESSURE TESTS AT INLET PRESSURES ABOVE 6% ACCUMULATION PRESSURE.
- BLOWDOWN RANGED FROM 10 TO 16% WHEN RATED LIFT WAS ACHIEVED.
- THE VALVE WAS REMOVED FROM THE TEST FACILITY FOR PERFORMANCE EVALUATION.

PRI/CE SAFETY VALVE TESTS

CROSBY 3K6 SAFETY VALVE

DATA REVIEW

(SHORT INLET CONFIGURATION)

SHORT INLET CONFIGURATION TEST SUMMARY

MATRIX CONDITIONS WERE PERFORMED ON THE CROSBY 3K6 VALVE:

STEAM TESTING

- THE VALVE OPENED WITHIN ± 3% OF SET PRESSURE (2500 PSIA)
- THE VALVE ACHIEVED AND MAINTAINED RATED LIFT AT 3% ACCUMULATION PRESSURE (2575 PSIA)
- VALVE OPERATION WAS STABLE AT RING SETTINGS RESULTING IN 8% BLOWDOWN AT HIGH AND LOW BACKPRESSURES
- THE VALVE EXHIBITED FLUTTER/INSTABILITY AT RING SETTINGS RESULTING IN 4 - 6% BLOWDOWN

NON-STEAM TESTS (USING 8% BLOWDOWN RING SETTINGS)

- ON THE STEAM-TO-WATER TRANSITION TEST, THE VALVE OPENED AND CLOSED WITHIN SCREENING CRITERIA AND HAD STABLE PERFORMANCE.
- THE VALVE OPENED, HAD STABLE PERFORMANCE ON 650°F WATER, WITH LIFT SUFFICIENT TO ACCOMODATE DRIVING FLOW.
- ON SECOND POPS, THE VALVE OPENED AT PRESSURES 4% - 9% BELOW 2500 PSIA WITH STABLE OPERATION.
- THE VALVE EITHER OPENED <10% OF RATED LIFT AT 3% ACCUMULATION PRESSURE OR CHATTERED ON 550°F WATER.
- HIGHER SUB-COOLED TESTS WERE JUDGED BY CROSBY TO RESULT IN PERFORMANCE SIMILAR TO THE 550°F TEST RESULTS AND WERE THEREFORE DEFERRED.

EPRI SAFETY VALVE TESTS
CROSBY 3K6 SAFETY VALVE

DATA REVIEW

(LONG INLET CONFIGURATION)

LONG INLET CONFIGURATION TEST SUMMARY

A TOTAL OF THIRTEEN TESTS WERE PERFORMED:

STEAM TESTS

- THE VALVE OPENED WITHIN ± 3% OF SET PRESSURE (2500 PSIA)
- THE VALVE OPERATION WAS STABLE AT RING SETTINGS RESULTING IN 15% - 20% BLOWDOWN
- THE VALVE OSCILLATED (CHATTERED) WITH RING SETTINGS RESULTING IN 10% BLOWDOWN ON THE SHORT PIPE CONFIGURATION
- CHATTER RESULTED IN VALVE INTERNALS DAMAGE WHICH WAS REFURBISHED PRIOR TO CONTINUED TESTING.

LOOP SEAL TESTS (USING 15-20% BLOWDOWN SETTINGS)

- FOUR LOOP SEAL TESTS WERE PERFORMED AT RAMP RATES OF 3-220 PSI/SEC.
- THE LOOP SEAL WAS A TYPICAL CONFIGURATION FOR A 3" INLET PIPE SIZE SAFETY VALVE
- THE VALVE FULLY LIFTED AT PRESSURES ABOVE THE +3% SCREENING CRITERIA FOR SET POINT
- OPENING PRESSURES INCREASED FROM APPROXIMATELY VALVE SET PRESSURE (2550 PSIA) AT LOW RAMP RATES TO 2710 PSIA AT HIGH RAMP RATES
- WHEN THE VALVE OPENED, IT EXHIBITED TYPICAL STABLE STEAM PERFORMANCE

EPRI SAFETY VALVE TESTS
CROSBY 3K6 SAFETY VALVE
LONG INLET CONFIGURATION
(CONTINUED)

TRANSITION TESTS (USING 15-20% BLOWDOWN SETTINGS)

- ONE LOOP SEAL-STEAM-WATER TRANSITION TEST WAS PERFORMED
- THE LOOP SEAL-STEAM DISCHARGE WAS STABLE
- THE VALVE BECAME UNSTABLE (FLUTTER) DURING STEAM-WATER TRANSITION
- FLUTTER CHANGED TO CHATTER DURING WATER FLOW
- THE TEST WAS TERMINATED BY MANUALLY PULLING THE VALVE OPEN TO STOP CHATTER
- CHATTER RESULTED IN VALVE INTERNALS DAMAGE

EPRI SAFETY VALVE TESTS

DRESSER 31709NA SAFETY VALVE

SHORT INLET CONFIGURATION TEST SUMMARY

A TOTAL OF 11 TESTS WERE PERFORMED:

STEAM

- THE VALVE OPENED WITHIN ± 3% OF SET PRESSURE (2515 PSIA)
- THE VALVE ACHIEVED AND MAINTAINED RATED LIFT AT 3% ACCUMULATION PRESSURE
- VALVE OPERATION WAS STABLE AT ALL RING SETTINGS
- BLOWDOWN RANGED FROM 11.5 - 13.5% AT LOW BACKPRESSURES DOWN TO 7.3% AT INTERMEDIATE BACKPRESSURES

NON-STEAM TESTS (USING STEAM RING SETTINGS)

- o TWO STEAM-WATER TRANSITION TESTS AT INTERMEDIATE BACKPRESSURES WERE PERFORMED,
  - THE VALVE OPENED AND WAS STABLE DURING THE TESTS
  - BLOWDOWN WAS APPROXIMATELY 20%
- o TWO 650°F WATER TESTS WERE PERFORMED AT INTERMEDIATE BACKPRESSURES
  - THE VALVE OPENED AND WAS STABLE, LIFT SUFFICIENT TO ACCOMODATE FLOW
  - BLOWDOWN WAS APPROXIMATELY 20%

EPRI SAFETY VALVE TESTS

SAFETY VALVE TESTS

SCHEDULE

MILESTONES	STATUS
TEST FACILITY CONSTRUCTION AND CHECKOUT	COMPLETE
PERFORMANCE TESTING SAFETY VALVE/PIPING CONFIGURATION (COMPLETION DATE)	
- DRESSER 31709NA/LONG	COMPLETE
- DRESSER 31739A/SHORT	PARTIALLY COMPLETE
- CROSBY 3K6/SHORT	COMPLETE
- CROSBY 3K6[1]/LONG	COMPLETE
- DRESSER 31709NA/SHORT	COMPLETE
- TARGET ROCK 69C/LONG (10/3/81)	
- CROSBY 6M6/SHORT (10/16/81)	
- CROSBY 6M6[1]/LONG (10/30/81)	
- DRESSER 31739A/LONG (11/13/81)	
- CROSBY 6N8/SHORT (11/25/81)	
- DRESSER 31739A/SHORT (12/16/81)	

(1) LOOP SEAL INTERNALS

EPRI SAFETY VALVE TESTS

SUMMARY

- TESTS PERFORMED ON FIVE SAFETY VALVE/INLET CONFIGURATIONS TO DATE

- VALVE PERFORMANCE OBSERVED TO BE DEPENDENT ON TEST CONDITIONS

  - FLUID STATE/TEMPERATURE
  - INLET PIPE CONFIGURATION
  - BACKPRESSURE

- ALL TESTS SCHEDULED TO BE COMPLETED AND DATA AVAILABLE EARLY 1982

ATTACHMENT N

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June 29, 1981

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Licensing Branch No. 3
Division of Licensing
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

RECEIVED
JUN 30 1981
U.S. NUCLEAR REGULATORY COMMISSION

Re: Docket No. 50-275
Docket No. 50-323
Diablo Canyon Units 1 and 2

Dear Mr. Miraglia:

This letter is submitted in partial response to NUREG-0737, Section II.D.1. A requirement of Section II.D.1 is that licensees must submit by July 1, 1981, plant specific submittals confirming the adequacy of safety and relief valves based on a preliminary review of generic (EPRI) test program results.

PGandE has reviewed the valve test program provided by EPRI in its program plan and test status reports. Based upon this review and the tests completed to date, PGandE has determined that the valves included in the EPRI program adequately represent the valves in the reactor coolant system at Diablo Canyon. Specifically, the Diablo Canyon valves represented in the EPRI program are: Masoneilan model no. 20,000 series relief valve; Crosby model HB-BP-86, size 6M6 safety valve with loop seal internal materials.

The Masoneilan valve has been tested under steam conditions and functioned properly. Testing under transition and liquid conditions of the Masoneilan valve is continuing. Testing of the Crosby valve has not yet been

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completed. A determination by PGandE of the adequacy of these valves for Diablo Canyon will be provided following completion of the EPRI tests. We understand that the current schedule for test completion should allow this determination to be made by October 1, 1981.

Very truly yours,

Philip A. Crane, Jr.

CC: J. Carey, EPRI
W. Gangloff, Westinghouse

Service List

DIABLO CANYON POWER PLANT
FEMA/NRC EMERGENCY RESPONSE
FIELD EXERCISE

III. EXERCISE SCENARIO

A. Initial Conditions

The PGandE electric power system is in a "low voltage schedule." Demand is extremely high. The entire Western Systems Coordinating Council Network is in a very low power reserve situation.

Unit 1 has been at full power for 15 days with 290 EFPD utilization of the core. All plant parameters are normal and stable. The turbine-driven auxiliary feedwater pump is in the second day of an estimated 72-hour outage for maintenance. Recently completed normal periodic system testing included the Turbine Building fire protection system and the diesel generator sequencer. Reactor control is in the "automatic" mode. Unit 2 is awaiting initial fuel load.

A Radiation Protection Monitoring Technician, an Auxiliary Operator and an Electrician are working on the motor of containment fan cooler unit 1-2.

B. Meteorological Conditions

Winds are from the SSE (155°) at 15 mph. It is a cool day with light fog. Ambient air temperature is 51°. Humidity is 54%.

C. Detailed Scenario Timeline

Time	Initiating Message Number	Event Summary
H+00:00 (7:00 am)	1	Initial conditions. Note: The Control Room Controller will provide plant parameters to Control Room personnel upon request.
H+00:02 (7:02 am)	2	A fire is detected in the Unit 2, 12KV startup switchgear, El.85' of the north end of the Unit 1 turbine building. Both the Unit 1 and Unit 2 12KV startup switchgear trip out and a loss of 230KV offsite power results. Diesel generators 1-1, 1-2 and 1-3 automatically start on a 12KV startup bus undervoltage signal.

III. EXERCISE SCENARIO

C. Detailed Scenario Timeline (Continued)

Time	Initiating Message Number	Event Summary
	2c	Note: Shift Foreman should declare a NOTIFICATION OF AN UNUSUAL EVENT and dispatch fire brigade.
H+00:10 (7:10 am)	3a,3b	The Turbine Building fire protection system appears to be inoperable. It is suspected that the fire protection system was not properly restored to service following recent testing activities. Manual fire fighting efforts are ineffective. The fire increases in intensity and threatens vital cable in the cable spreading rooms above on El.104' of the Turbine Building.
	3c	Note: The Shift Foreman should declare an ALERT. Offsite fire fighting assistance should be requested (communications check only, offsite assistance should not be mobilized).
H+00:40 (7:40 am)	4	The fire on El.85' in the north end of the Turbine Building is reported to be under control.
H+00:50 (7:50 am)	5	The fire on El.85' in the north end of the Turbine Building is reported to be totally extinguished.
H+00:55 (7:55 am)	6	The Unit 1 12KV startup switchgear has been restored to operability. Apparently, it tripped out due to the fire, but was not damaged. As a result, 230KV offsite power has been restored to availability.

III. EXERCISE SCENARIO

C. Detailed Scenario Timeline (Continued)

Time	Initiating Message Number	Event Summary
H+01:00 (8:00 am)	7	A Bank D rod cluster control assembly (RCCA) is ejected from the reactor core. The reactor trips, followed by the turbine. Auxiliary power transfers to the startup buses. The steam dump system activates and functions normally. The safety injection system initiates as reactor coolant system (RCS) pressure and temperature decrease.
	7c	Note: A SITE AREA EMERGENCY should be declared by the Shift Foreman at this point. Onsite radiological monitoring teams should be dispatched in this time frame.
H+01:01+ (8:01am+)		Note: Onsite radiological monitoring teams should commence reporting survey results. Controllers will provide the radiological monitoring teams with radiation survey instrument readings at each monitoring location upon request.
	8	The motor-driven auxiliary feedwater pumps start normally and provide the steam generators with feedwater. Containment high radiation and high humidity alarms received.
H+01:10 (8:10 am)	9	The Shift Foreman is notified that the Radiation Protection Monitoring (RPM) Technician, Auxiliary Operator and Electrician, who were working on containment fan cooler unit 1-2, have been contaminated. Upon hearing a loud crash and the sound of escaping steam, the men hurriedly attempted to exit the Containment. The Electrician fell and appears to have broken his left leg. All three men were contaminated prior to exiting the Containment through the personnel hatch on El.140'.

III. EXERCISE SCENARIO

C. Detailed Scenario Timeline (Continued)

Time	Initiating Message Number	Event Summary
		Note: An ambulance should be requested to transport the injured Electrician to the hospital. French Hospital should be notified that a contaminated, injured individual will be arriving by ambulance.
H+01:20 (8:20 am)	9c	Note: Although a radioactive release has not taken place, given that the wind direction is from the south, the County should consider sheltering of the populace within 5 miles of the plant in Sectors NNW, N and NNE.
H+01:40 (8:40 am)	10	The reactor is in a stable, hot shutdown condition. The charging pumps have stabilized RCS pressure at 2235 psig and continue to provide makeup for coolant lost through the RCCA ejection break. RCS T_{AVG} is 542°F. The plant operations staff is analyzing all available data, and preparing to cool down and depressurize the RCS.
H+02:00 (9:00 am)	11a, 11b	As the ambulance carrying the injured and contaminated Electrician leaves the Diablo Canyon access road (immediately outside the Avila Beach gate), it collides with a station wagon traveling at a high rate of speed. The people in the ambulance are knocked unconscious in the collision. Personnel arriving at the accident scene are not certain which occupants are contaminated.
H+02:10 (9:10 am)	12	To remain within Technical Specification RCS pressure-temperature cooldown limits, it is estimated that depressurization of the RCS will take approximately 3 to 5 hours. At that time, residual heat removal system operation will be initiated to bring the reactor to a cold shutdown condition.

III. EXERCISE SCENARIO

C. Detailed Scenario Timeline (Continued)

Time	Initiating Message Number	Event Summary
H+02:20 (9:20 am)	13	The Control Room receives continuous indication of high activity inside the Containment. Containment integrity has precluded any release of radiation to the environment.
H+02:30 (9:30 am)	14c-1	Note: Winds have shifted direction and are now from the WSW.
H+02:45 (9:45 am)	14c-2	Note: Although a radioactive release has not taken place nor is one expected, given the shifting winds, the County should consider protective actions in the form of sheltering for portions of the City of San Luis Obispo, east of the plant site.
H+03:15 (10:15 am)	15	Depressurization of the RCS is proceeding in a slow, but orderly and stable manner.
H+03:30 (10:30 am)	16	Due to an electric power system grid disturbance, there is a loss of all 230KV and 500KV offsite power.
H+03:35 (10:35 am)	17	Diesel generators 1-1, 1-2 and 1-3 have picked up all vital loads. However, the motor-driven auxiliary feedwater pumps fail to start. Note: A GENERAL EMERGENCY should be declared at this point.
H+03:45 (10:45 am)	18	The Electric System Dispatcher Shift Supervisor informs the Shift Foreman that offsite power will be unavailable for four to six hours.
	18c	Note: A GENERAL EMERGENCY should be declared at this point if not previously done.

III. EXERCISE SCENARIO

C. Detailed Scenario Timeline (Continued)

Time	Initiating Message Number	Event Summary
H+04:00 (11:00 am)	19	All steam generators boil dry resulting in the loss of the primary system heat sink.
H+04:05 (11:05 am)	20	RCS temperature and pressure are rapidly increasing.
H+04:10 (11:10 am)	21	All pressurizer power-operated relief valves (PORV) fail closed as actuation pressure is reached (or if early actuation is attempted).
H+04:12 (11:12 am)	22	The Control Room receives indication of fuel damage in the reactor core and a rapidly increasing hydrogen concentration in the Containment. The hydrogen recombiners appear to be inoperable.
H+04:15 (11:15 am)	23	Lunch break. The Field Exercise is in recess for 30 minutes. Resume current positions and locations at 11:45 am for resumption of exercise play.
H+04:45 (11:45 am)	24c	Note: Wind direction has again shifted and is now from the WNW.
H+04:45 (11:45 am)	25	A hydrogen explosion occurs inside the Containment as explosive concentration limits are reached. Containment purge exhaust valves RCV-11 and RCV-12 are damaged and appear to be partially open as the unit vent particulate, radiogas and iodine monitors all indicate very high activity levels.
H+04:50 (11:50 am)	25c	Note: If closure is attempted, containment purge exhaust manual valve 35 fails open.

III. EXERCISE SCENARIO

C. Detailed Scenario Timeline (Continued)

Time	Initiating Message Number	Event Summary
H+07:15 (2:15 pm)	26	The motor-driven auxiliary feedwater pumps are restored to service and feed-water is now being delivered to the steam generators.
H+07:45 (2:45 pm)	27	The maintenance team repairs and closes containment purge exhaust fan manual valve 35. The release from the plant is terminated.
H+19:50 (2:50 pm)	28	It is now 12 hours later. The radioactive plume has completely dispersed and there is no trace of it over land.
H+20:00 (3:00 pm)	28c	Note: Ingestion pathway sampling teams should be dispatched if not done so previously.
H+20:00 to 21:00 (3:00 pm to 4:00 pm)		Note: Ingestion pathway sampling teams should procure samples and field data in this time frame.
H+20:50 (3:50 pm)	29	Long-term recovery actions should be discussed by the exercise participants.
H+21:00 (4:00 pm)	30	The Field Exercise is terminated pending completion of ingestion pathway monitoring activities.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
PACIFIC GAS AND ELECTRIC COMPANY	)	Docket Nos. 50-275 O.L.
	)	50-323 O.L.
(Diablo Canyon Nuclear Power Plant,	)	
Unit Nos. 1 and 2)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of "NRC STAFF RESPONSE TO JOINT INTERVENORS' SECOND SET OF INTERROGATORIES" in the above-captioned proceeding have been served on the following by deposit in the United States mail, first class or, as indicated by an asterisk, through deposit in the Nuclear Regulatory Commission's internal mail system, this 6th day of November, 1981:

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