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, Units 1 and 2)	Docket Nos. 50-275 O.L. 50-323 O.L.

RESPONSE OF JOINT INTERVENORS TO APPLICANT PACIFIC GAS AND ELECTRIC COMPANY'S THIRD SET OF INTERROGATORIES

Joint Intervenors hereby respond to Pacific Gas and Electric Company's Third Set of Interrogatories dated October 15, 1981 as follows:

Response 1

The PORV's and Block Valves* are not specifically identified in the FSAR Section 3.2 tables but they are included in the Hosgri Seismic Evaluation (Vol. III Table 7.8, "Summary - Seismic Qualification Valves Required for Normal Shutdown and/or Cold Shutdown.") There are few other details of the classification and qualification of these three types of valves.

*In contrast, Diablo Canyon Safety Valves are classified as safety-grade and subjected to the requirement of Design Class I, Code Class I as described in FSAR Tables 3.2-1, 3.2-2, 3.2-3, and 3.2-4. Similarly, they were identified in the Hosgri Amendment to the FSAR as having been seismically tested (See Hosgri Seismic evaluation, VOL. III, Table 7-7 "Seismic Qualification Minimum Required Active Valves for Hot Shutdown and/or Cold Shutdown.")

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However, 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 that their failure can cause or aggravate a LOCA. Therefore, these valves must also be classified as safety-grade components and required to meet all safety-grade design criteria. There is insufficient information to know if the existing valves and their associated equipment meet the necessary requirement to insure reliable performance of their safety function under worst case accident conditions.

Response 2

See Response 1. The failure of control and/or instruments could lead to failure of the associated valves, thereby causing or aggravating a LOCA. Thus, the associated controls and instruments for these valves must comply with applicable codes, standards, and regulatory practices. The

NRC Standard Review Plan (NUREG 75/087 Section 7, Table 7-1) identifies the acceptance criteria for safety-related instrumentation and control equipment which should be applied to these components. A copy of this table is attached.

Until adequate details are provided on how the valves and components meet the above safety and acceptance criteria, there can be no assurance of their ability to perform properly in all off-normal and accident conditions.

Response 3

In addition to the discussion in Responses 1 and 2, there are conditions where the block valves and PORVs may individually or collectively constitute a potential break in the reactor coolant pressure boundary. Failure to operate correctly, in either opening or closing, may cause or aggravate a small LOCA. The valves can also play an important role in mitigating the effects of an ATWS accident. They may also serve as a mechanism for control and/or mitigation of accident conditions when called upon to operate in the bleed and feed mode (in conjunction with Safety Injection). Components which have this large an impact on pressure boundary integrity, accidents, and safety should be classed as safety-grade. Examples include the following:

(a) A block valve failure to close when the PORV sticks open can create a small LOCA, one of the design basis events in the FSAR. In the preceding example of a PORV stuck open, mitigation of the small LOCA may be accomplished by closing the associated block valve.

(b) There are sequences where failures of the block valves would prevent operation of the PORV's. Thus, block valve failure could prevent the use of PORV's as a means of overpressure protection during low temperature operation. The Applicant's response to NUREG-0578 (TMI Lessons Learned) refers to both block valves and PORV's in regard to low temperature over-pressurization protection. (PG&E response to Short Term Lessons Learned, February 29, 1980, page III-B-13.

(c) Failure of a PORV to close, and the failure of the block valve to be closed by the operator coupled with the failure of the emergency coolant systems and auxiliary feedwater system functions could result in core damage (for example, see the TMI-2 accident scenario).

(d) Although the normal procedures do not appear to call for use of the block valves or PORV's to shutdown the reactor and maintain it in a safe shutdown condition, there are conditions where they may be called upon to assist in maintaining the plant in a safe shutdown condition. The TMI-2 accident and post-accident mitigation is such an example.

(e) ATWS is not a design basis event for Diablo Canyon at this time. Therefore, ATWS has not been protected against solely with safety grade equipment.

Response 4

In addition to the accident scenarios set forth in Responses 1, 2, and 3, during a small break LOCA where there is also a PORV/block valve failure, there is a possibility of erroneous behavior of the pressurizer function, pressurizer level indication, and vessel level indication. Operator action and, thus, system behavior in the light of such possibly misleading information cannot be predicted with certainty.

Response 5

Yes. See also response to Interrogatory 6.

Response 6

Yes. See also response to Interrogatory 5.

Response 7

No applicable.

Response 8

Not applicable.

Response 9:

Diablo Canyon safety valves are classified as safety-grade and subjected to the requirements of Design Class I, Code Class I as described in FSAR Tables 3.2-1, 3.2-2, 3.2-3, and 3.2-4. Similarly, they were identified in the Hosgri Amendment to the FSAR as having been seismically tested (see Hosgri seismic evaluation, Vol. III, Table 7-7, Seismic Qualification Minimum Required Active Valves for Hot Shutdown and/or cold Shutdown.") The PORV's and block valves are not specifically identified in the FSAR Section 3.2 tables but they are included in the Hosgri Seismic Evaluation (Vol. III, Table 7.8, "Summary-Seismic Qualification Valves Required for Normal Shutdown and/or Cold Shutdown." There are few other details of the classification and qualification of these three types of valves.

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 also be classified as safety-grade components and required to meet <u>all</u> safety-grade design criteria. There is insufficient information to know if the existing valves and their associated equipment meet the necessary performance requirements to insure reliability performance of their safety function under worst case accident conditions.

Similarly, the associated control and instruments for these valves must comply with applicable codes, standards, etc. The NRC Standard Review Plan (NUREG-75/087), Section 7, Table 7-1) identifies the acceptance criteria for safety-related instrumentation and control equipment which should be applied to these components. A copy of this table is attached. Until details are provided on how the valves and components meet the above safety and acceptance criteria, there can be no assurance of their adequacy to perform properly in all off-normal and accident conditiona.

<u>Response 10</u>:

(a) and (b) The location and intended purpose of each such valve are set forth in general in the Diablo Canyon Final Safety Analysis Report. The Applicant, as the designer of the plant, should be thoroughly familiar with the location and intended purpose of each such valve. Also see "Applicant's Answers to Joint Intervenors' Second Set of Interrogatories", dated October 26, 1981, including particularly answer Nos. 46, 49, and 50.

(c) and (d) See Response to Interrogatories Nos. 1, 2, and 3.

(e) - See response to Interrogatory 9.

Response 11

While it may be possible to maintain natural circulation at hot stand-by conditions without the pressurizer heaters and associated controls, such operation may be difficult to control and is contrary to the normal plant operating procedures (see PG&E response No. 45 dated October 26, 1981 to Joint Intervenors Second Set of Interrogatories for a list of emergency operating procedures that include the use of pressurizer heaters). Further, plant safety may be affected by many things, not the least of which is the need to minimize the number of challenges to the total system integrity and to optimize the operability and controllability of systems used in the mitigation or control of abnormal events. The NRR Lessons Learned Task Force found that "maintenance of natural circulation capability is important to safety".* Pressurizer heaters are needed for this capability. In addition, the pressurizer heaters must also maintain physical integrity for the reactor coolant pressure boundary to be maintained.

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* NUREG-0578, page A-2.

Response 12

See Response No. 11 concerning the need for classification of the components as important to safety. Further, all components of the pressurizer heater system, including supports and interconnecting wiring should be required to meet the applicable safety-grade design criteria. PG&E has responded that only that equipment associated with the capability of obtaining power from the on-site emergency power supply needs to meet GDC 10, 14, 15, 17 and 20 of Appendix A to 10CFR50.* This is further defined in PG&E's Answer to Interrogatory No. 41 as the 480 volt vital breakers 52-1G-72 & -1H-74, control

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*Applicant Pacific Gas & Electric Company's Answers to Joint Intervenors' Second Set of Interrogatories, page 1 & 2.

switches and cable between the vital bus and the breakers.*
This implies then that all of the rest of the pressurizer heater system has not been designed to meet the safety-grade design criteria listed above. The remainder of the system, therefore, consists of the heaters themselves and their associated controls, along with interconnecting wiring and supports. See PG&E January 26, 1981 submittal to NRC on Full Power License Requirement and associated Figures II.E.3.1-1 & -2 for diagrams showing the components contained within the pressurizer heater system.**

Response 13

See "Applicant's Answers to Joint Intervenors' Second Set of Interrogatories" dated October 26, 1981, particularly Response 34 where the applicant clearly acknowledges that for Diablo Canyon the pressurizer heaters and associated controls are not classified "important to safety".

Contention 10 does not state that the pressurizer heaters and associated controls fail to comply with "any" specific details in the General Design Criteria but rather that this

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* Applicant Pacific Gas & Electric Company's Answers to Joint Intervenors' Second Set of Interrogatories, pages 16 & 17.
**Philip A. Crane to Frank J. Miraglia, January 26, 1981, pages II.E-10 through 19.

equipment has not been classified as safety-grade and therefore not been required to meet the safety-grade design criteria listed. There is obviously no way to evaluate that compliance since PG&E has not submitted any detailed information on how these components do or do not meet the specific criteria. This Interrogatory is therefore premature until sufficient detailed information is available to evaluate compliance. However, it is likely that non-compliances exist for the following reasons:

a. GDC 20 requires, among other things, that the protection system shall be designed "to initiate the operation of systems important to safety." Standard Review Plan Table 7-1 extends the applicability of GDC 20 to all instrumentation and control functions important to safety.* PG&E's January 26, 1981 response to Full Power License Requirements describes the manual procedure necessary for transferring the pressurizer heater power supply onto the ESF buses. This requires the dispatch of an operator to a location three floors down in the Auxiliary Building and verbal confirmation that such action has been taken.** This complex procedure does not meet the automatic initiation requirements of GDC 20.

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* NUREG 75/087, Section 7, Table 7-1.
** Philip A. Crane to Frank J. Miraglia, January 26, 1981, page II E-14.

b. None of the pressurizer heater system, other than the breakers, switches and portion of the bus connection cables identified in Response 1, has been qualified in accordance with EGC 2 (seismic and environmental qualification) GDC 22 (protection system independence, "separation") on GDC 3 (fire protection).

c. Since these components have not been classified as important to safety, the requirement of GDC 1 (Quality standards and records) does not appear to have been applied.[1/]

Response 14

See Response 13.

Response 15

The proposed arrangement addresses only the reliability of power supply to the pressurizer heaters. The heaters and associated controls are still subject to failures introduced through incomplete attention and lack of compliance with the applicable safety-grade criteria (See Responses 11, 12, 13 and 14).

1/ We note that the classification of pressurizer heaters and associated controls is currently the subject of Union of Concerned Scientists Contention 3 in the ongoing TMI restart hearings (NRC docket 50-289).

Response 16:

Documents which were utilized as the base of answers to Interrogatories 1 to 15 herein were identified at the point of reference in the specific interrogatory responses. The documents' description included sufficient information to identify the documents, including the identification of the specific page(s) of the document which relate to each interrogatory response.

Response 17:

The term "any pending" as related to contentions is ambiguous. Likewise, the term "these" lacks the necessary specific basis. Accordingly, we cannot identify any additional documents or exhibits as set forth in this request. However, assuming that this request is limited to the subjects identified as "Contention 10" and "Contention 12" in the current Diablo Canyon full power license proceeding, the documents or exhibits relied upon which Joint Intervenors may introduce into evidence are identified in the foregoing Responses 1 through 16. Additional documents and exhibits may be identified during the ongoing document discovery and as a result of NRC Staff and PG&E responses to Governor Brown and Joint Intervenor interrogatories. All parties have access to the documents provided during discovery. Further, such documents and exhibits will gneerally be referenced in the testimony of Joint Intervenors, witnesses which will be submitted to all parties in this full-power proceeding prior to any hearings.

Response 18:

Assuming that this request is limited to the subject identified as "Contention 10" and "Contention 12" in the current Diablo Canyon full power license proceeding, the identification of witnesses Joint Intervenors may call to testify was set forth in "Joint Intervenors' Identification of Witnesses for Full Power Proceeding" dated November 3, 1981. At that time, the following potential witness for the subject two contentions was identified: Robert Pollard. Joint Intervenors will identify other witnesses in the future once the decision is made to present other witnesses. At this time Joint Intervenors do not plan to subpoena any witnesses on "Contention 10" and "Contention 12". Further information in response to this interrogatory will be supplied when it becomes available to Joint Intervenors' counsel.

DATED" November 4, 1981 Respectfully submitted,

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NUREG-75/087

U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
OFFICE OF NUCLEAR REACTOR REGULATION

TABLE 7-1

ACCEPTANCE CRITERIA FOR INSTRUMENTATION AND CONTROLS

Table 7-1 contains the acceptance criteria for the SRP sections of Chapter 7. These acceptance criteria include the applicable General Design Criteria, IEEE standards, Regulatory Guides, and Branch Technical Positions (BTP) of the Instrumentation and Control Systems Branch (ICSB). The applicability of these criteria to specific sections of Chapter 7 is indicated by an X in the matrix listing of criteria and SAR sections. The BTP listed in Table 7-1 are contained in Appendix 7-A to the Chapter 7 SRP section.

ACCEPTANCE CRITERIA FOR INSTRUMENTATION AND CONTROL SYSTEMS - TABLE 7-1

CRITERIA	TITLE	APPLICABILITY 7.1	7.2	7.3	7.4	7.5	7.6	7.7	REMARKS
1. 10 CFR Part 50									
a. 10 CFR §50.34	Contents of Application: Technical Information	X	X	X	X	X	X	X	
b. 10 CFR §50.36	Technical Specifications	X	X	X	X	X	X		
c. 10 CFR §50.55a	Codes and Standards	X	X	X	X	X	X	X	
2. General Design Criteria (GDC), Appendix A to 10 CFR Part 50									
a. GDC 1	Quality Standards and Records	X	X	X	X	X	X		
b. GDC 2	Design Bases for Protection Against Natural Phenomena	X	X	X	X	X	X		
c. GDC 3	Fire Protection	X	X	X	X	X	X		
d. GDC 4	Environmental and Missile Design Bases	X	X	X	X	X	X		
e. GDC 5	Sharing of Structures, Systems, and Components	X	X	X	X	X	X		
f. DGC 10	Reactor Design	X	X	X	X	X	X		
g. GDC 12	Suppression of Reactor Power Oscillations	X	X			X		X	
h. GDC 13	Instrumentation and Control	X	X	X	X	X	X	X	
i. GDC 15	Reactor Coolant System Design	X	X			X	X	X	
j. GDC 19	Control Room	X	X	X	X	X	X	X	
k. GDC 20	Protection System Functions	X	X	X	X	X	X		
l. GDC 21	Protection Systems Reliability and Testability	X	X	X	X	X	X		
m. GDC 22	Protection System Independence	X	X	X	X	X	X		
n. GDC 23	Protection System Failure Modes	X	X	X	X	X	X		
o. GDC 24	Separation of Protection and Control Systems	X	X	X	X	X	X	X	
p. GDC 25	Protection System Requirements for Reactivity Control Malfunctions	X	X			X			

TABLE 7-1 (CONTINUED)

CRITERIA	TITLE	APPLICABILITY 7.1	7.2	7.3	7.4	7.5	7.6	7.7	REMARKS
q. GDC 26	Reactivity Control System Redundancy and Capability	X	X		X	X		X	
r. GDC 27	Combined Reactivity Control Systems Capability	X	X		X	X		X	
s. GDC 28	Reactivity Limits	X	X			X	X	X	7.6 Interlocks only
t. GDC 29	Protection Against Anticipated Operational Occurrences	X	X	X	X	X	X	X	
u. GDC 33	Reactor Coolant Makeup	X			X	X			
v. GDC 34	Residual Heat Removal	X		X	X	X	X		
w. GDC 35	Emergency Core Cooling	X	X	X		X	X		
x. GDC 37	Testing of Emergency Core Cooling System	X	X	X		X	X		
y. GDC 38	Containment Heat Removal	X		X		X	X		
z. GDC 40	Testing of Containment Heat Removal System	X		X		X	X		
aa. GDC 41	Containment Atmosphere Cleanup	X		X		X	X		
bb. GDC 43	Testing of Containment Atmosphere Cleanup Systems	X		X		X	X		
cc. GDC 44	Cooling Water	X		X		X	X		
dd. GDC 46	Testing of Cooling Water System	X		X		X	X		
ee. GDC 50	Containment Design Basis	X		X		X	X		
ff. GDC 54	Piping Systems Penetrating Containment	X		X		X	X		
gg. GDC 55	Reactor Coolant Pressure Boundary Penetrating Containment	X		X		X	X		
hh. GDC 56	Primary Containment Isolation	X		X		X	X		
ii. GDC 57	Closed Systems Isolation Valves	X		X		X	X		

TABLE 7-1 (CONTINUED)

CRITERIA	TITLE	APPLICABILITY 7.1	7.2	7.3	7.4	7.5	7.6	7.7	REMARKS
3. Institute of Electrical and Electronics Engineers (IEEE) Standards:									
a. IEEE Std. 279 (ANSI N42.7)	Criteria for Protection Systems for Nuclear Power Generating Stations	X	X	X	X	X	X	X	See 10 CFR §50.55a(h) and Reg. Guidr 1.62.
b. IEEE Std 308	Criteria for Class IE Electric Systems for Nuclear Power Generating Stations	X			X	X	X		See Reg. Guide 1.32.
c. IEEE Std 317	Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations	X	X	X	X	X	X	X	See Reg. Guide 1.63. SRP Section 3.11.
d. IEEE Std. 336 (ANSI N45.2.4)	Installation, Inspection and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations	X	X	X	X	X	X	X	See Reg. Guide 1.30.
e. IEEE Std 338	Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems	X	X	X	X	X	X		See Reg. Guide 1.118.
f. IEEE Std 344 (ANSI N41.7)	Guide for Seismic Qualification of Class I Electrical Equipment for Nuclear Power Generating Stations	X	X	X	X	X	X		See Reg. Guide 1.100 SRP Section 3.10.
g. IEEE Std 379 (ANSI N41.2)	Guide for the Application of the Single Failure Criterion to Nuclear Power Generating Station Protection Systems	X	X	X	X	X	X	X	See Reg. Guide 1.53.
h. IEEE Std 384 (ANSI N41.14)	Criteria for Separation of Class IE Equipment and Circuits	X	X	X	X	X	X	X	See Reg. Guide 1.75.

TABLE 7-1 (CONTINUED)

CRITERIA	TITLE	APPLICABILITY 7.1	7.2	7.3	7.4	7.5	7.6	7.7	REMARKS
4. Regulatory Guides (RG)									
a. RG 1.6	Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems	X			X	X	X		
b. RG 1.7	Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident	X		X		X			
c. RG 1.11	Instrument Lines Penetrating Primary Reactor Containment	X	X	X	X	X	X		
d. RG 1.22	Periodic Testing of Protection System Actuation Functions	X	X	X	X	X	X		
e. RG 1.29	Seismic Design Classification	X	X	X	X	X	X		SRP Section 3.10
f. RG 1.30	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment	X	X	X	X	X	X	X	
g. RG 1.32	Use of IEEE Std 308 "Criteria for Class IE Electric Systems for Nuclear Power Generating Stations"	X			X	X	X		
h. RG 1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems	X	X	X	X	X	X		Use in conjunction with Position 3, RG 1.17.
i. RG 1.53	Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems	X	X	X	X	X	X		
j. RG 1.62	Manual Initiation of Protection Actions	X	X	X	X		X		

TABLE 7-1 (CONTINUED)

CRITERIA		TITLE	APPLICABILITY 7.1	7.2	7.3	7.4	7.5	7.6	7.7	REMARKS
k.	RG 1.63	Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plant	X	X	X	X	X	X	X	
l.	RG 1.68	Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors	X	X	X	X	X	X	X	
m.	RG 1.70	Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, Rev. 2	X	X	X	X	X	X	X	
n.	RG 1.75	Physical Independence of Electric Systems	X	X	X		X			
o.	RG 1.78	Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release	X					X		
p.	RG 1.89	Qualification of Class IE Equipment for Nuclear Power Plants	X	X	X	X	X	X		SRP Section 3.11.
q.	RG 1.96	Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants	X		X					
r.	RG 1.12	Instrumentation for Earthquakes	X					X		
s.	RG 1.45	Reactor Coolant Pressure Boundary Leakage Detection Systems	X					X		
t.	RG 1.67	Installation of Overpressure Protection Devices	X					X		
u.	RG 1.80	Pre-operational Testing of Instrument Air	X		X	X		X		SRP Section 9.

TABLE 7-1 (CONTINUED)

CRITERIA	TITLE	APPLICABILITY							REMARKS
		7.1	7.2	7.3	7.4	7.5	7.6	7.7	
v. RG 1.95	Protection of Nuclear Power Plant Control Room Operators Against Accidental Chlorine Releases	X					X		
w. RG 1.97	Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and following an Accident	X				X			
x. RG 1.100	Seismic Qualification of Electrical Equipment for Nuclear Power Plants	X	X	X	X	X	X		SRP Section 3.10.
y. RG 1.105	Instrument Spans and Setpoints	X	X	X	X	X	X		
z. RG 1.118	Periodic Testing of Electric Power and Protection Systems	X	X	X	X	X	X		
aa. RG 1.120	Fire Protection Guidelines for Nuclear Power Plants	X	X	X	X	X	X	X	SRP Section 3.10.
5. Branch Technical Positions (BTP) ICSB									
a. BTP ICSB 1	Backfitting of the Protection and Emergency Power Systems of Nuclear Reactors	X	X	X	X		X		DOR Responsibility.
b. BTP ICSB 3	Isolation of Low Pressure Systems from the High Pressure Reactor Coolant System	X			X		X		
c. BTP ICSB 4 (PSB)	Requirements on Motor-Operated Valves in the ECCS Accumulator Lines	X			X		X		
d. BTP ICSB 5	Scram Breaker Test Requirements - Technical Specifications	X	X						
e. BTP ICSB 9	Definition and Use of "Channel-Calibration" - Technical Specifications	X	X		X	X	X		

TABLE 7-1 (CONTINUED)

CRITERIA	TITLE	APPLICABILITY 7.1	7.2	7.3	7.4	7.5	7.6	7.7	REMARKS
f. BTP ICSB 10	Electrical and Mechanical Equipment Seismic Qualification Program	X	X		X	X	X		Replaced by Reg. Guide 1.100
g. BTP ICSB 12	Protection System Trip Point Changes for Operation with Reactor Coolant Pumps Out of Service	X	X	X					
h. BTP ICSB 13	Design Criteria for Auxiliary Feedwater Systems	X		X					
i. BTP ICSB 14	Spurious Withdrawals of Single Control Rods in Pressurized Water Reactors	X	X					X	
j. BTP ICSB 15 (PSB)	Reactor Coolant Pump Breaker Qualification	X	X						
k. BTP ICSB 16	Control Element Assembly (CEA) Interlocks in Combustion Engineering Reactors	X	X						
l. BTP ICSB 18 (PSB)	Application of the Single Failure Criterion to Manually-Controlled Electrically-Operated Valves	X		X	X		X		
m. BTP ICSB 19	Acceptability of Design Criteria for Hydrogen Mixing and Drywell Vacuum Relief Systems	X		X			X		
n. BTP ICSB 20	Design of Instrumentation and Controls Provided to Accomplish Changeover from Injection to Recirculation Mode	X		X	X		X		
o. BTP ICSB 21	Guidance for Application of Reg. Guide 1.47	X	X	X	X	X	X		
p. BTP ICSB 22	Guidance for Application of Req. Guide .122	X	X	X	X	X	X		

TABLE 7-1 (CONTINUED)

CRITERIA	TITLE	APPLICABILITY 7.1	7.2	7.3	7.4	7.5	7.6	7.7	REMARKS
q. BTP ICSB 23	Qualification of Safety-Related Display Instrumentation for Post-Accident Condition Monitoring and Safe Shutdown	X				X			Replaced by Reg. Guide 1.97.
r. BTP ICSB 24	Testing of Reactor Trip System and Engineered Safety Feature Actuation System Sensor Response Times	X	X	X	X		X		Replaced by Reg. Guide 1.118.
s. BTP ICSB 25	Guidance for the Interpretation of General Design Criterion 37 for Testing the Operability of the Emergency Core Cooling System as a Whole	X		X	X				
t. BTP ICSB 26	Requirements for Reactor Protection System Anticipatory Trips	X	X						
u. BTP ICSB 27	Design Criteria for Thermal Overload Protection for Motors of Motor-Operated Valves	X		X	X		X		Replaced by Reg. Guide 1.106

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AFFIDAVIT OF

DALE G. BRIDENBAUGH, RICHARD B. HUBBARD, AND GREGORY C. MINOR

FOR JOINT INTERVENORS

DALE G. BRIDENBAUGH, RICHARD B. HUBBARD, AND GREGORY C. MINOR, being duly sworn, do say under oath that I, the undersigned have assisted in preparing and reviewing responses of Joint Intervenors to Pacific Gas and Electric Company's Third Set of Interrogatories Nos. 1-18. Said answers are true and correct to the best of my knowledge and belief.

Dale G Bridenbaugh
Dale G. Bridenbaugh

Richard B Hubbard
Richard B. Hubbard

Gregory C Minor
Gregory C. Minor

October 30, 1981

Subscribed and sworn to before me this 30th day of October, 1981.

Carlo F Caralli

NOTARY PUBLIC

My commission expires: 10/5/84

OFFICIAL SEAL
CARLO F. CARALLI
Notary Public California
Principal Office in
Santa Clara County
My commission expires Oct. 5, 1984

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, Units 1 and 2)	Docket Nos. 50-275 O.L. 50-323 O.L.

CERTIFICATE OF SERVICE

I hereby certify that on this 4th day of November, 1981, I have served copies of the foregoing JOINT INTERVENORS' RESPONSE TO NRC STAFF'S REQUEST FOR ADMISSIONS, RESPONSE OF JOINT INTERVENORS TO SECOND SET OF INTERROGATORIES OF NRC STAFF, and RESPONSE OF JOINT INTERVENORS TO APPLICANT PACIFIC GAS AND ELECTRIC COMPANY'S THIRD SET OF INTERROGATORIES, mailing them through the U. S. mails, first class, postage prepaid.

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Board
U. S. Nuclear Regulatory
Commission
Washington, D.C. 20555

Glenn O. Bright
Atomic Safety & Licensing
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