EXHIBIT 2

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 DALE G. BRIDENBAUGH
AND GREGORY C. MINOR

STATE OF CALIFORNIA)
) ss.
COUNTY OF SANTA CLARA)

DALE G. BRIDENBAUGH and GREGORY C. MINOR depose and say under oath as follows:

My name is Dale G. Bridenbaugh. A statement of my qualifications and experience has previously been provided to this Board as part of my testimony on Contention 10 and in Attachment A to that testimony.

My name is Gregory C. Minor. A statement of my qualifications and experience has previously been provided to this Board as part of my testimony on Contention 1 and in Attachment B to that testimony.

This affidavit relates to Joint Intervenors Contentions 10 and 12 as set forth in the ASLB Prehearing Conference Order of February 13, 1981 and specifically responds to the "Statement of Material Facts as to Which There Is No Genuine Issue" contained in the December 21, 1981 NRC Staff Motion for Summary Disposition of Contentions 10 and 12.

We attest that:

1. Classification of the pressurizer heaters at Diablo Canyon as "components important to safety" has not been established. While the NRC Staff claims that they have been, Affidavits by the Applicant's personnel state that:

> "There are no requirements for the pressurizer heaters and associated controls to be classified as 'components important to safety.'" 1/ and:
>
> "....the pressurizer heaters and associated controls are not classified as 'components important to safety.'" 2/

It therefore appears that the NRC Staff considers the pressurizer heaters as "components important to safety" but the Applicant has not treated them as such.

2. The pressurizer heaters do perform critical functions identified in 10 CFR 100, Appendix A, Section III (c). Physical

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1/ Pacific Gas and Electric Company's Motion for Summary Disposition, Affidavit of John B. Hoch, p. 2, paragraph 6.

2/ Ibid 1, paragraph 7.

integrity of the heaters is required to preserve the "integrity of the reactor coolant pressure boundary." Post-accident decay heat removal via the natural circulation mode is a function required and is normally achieved via and is specified in the Emergency Operating Procedures to be performed by the pressurizer heater system.

3. Pressurizer heaters are normally utilized in controlling reactor pressure while bringing the plant to cold shutdown.

4. Failure of the pressurizer heaters to operate would allow the reactor system to depressurize at essentially an uncontrolled rate unless additional equipment is brought into operation in a mode not normally utilized and which has not been clearly defined in the Emergency Operating Procedures.

5. The pressurizer heater system is the normal system utilized to control the primary reactor pressure. The pressurizer heaters have been designated as "important to safety" by the NRC Staff,[3/] they have been recommended to be upgraded in numerous NRC studies and reports, and are recognized as being of importance to the reduction of challenges to the other safety systems.

6. Two manual transfer switches with associated safety-related protective devices have been provided to connect the

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3/ NRC Staff Motion for Summary Disposition, 12/21/81, p. 6.

pressurizer heaters to on-site standby power supplies. They ar not, however, operable from the control room as was recommended by the TMI Action Plan.[4/] In addition, the safety-related devices serve only to protect the on-site power system rather tha also protecting the pressure control function.

7. Diablo Canyon has U-tube steam generators. The Applicant has not yet, however, demonstrated the adequacy of natural circulation through these steam generators at Diablo Canyon under adverse pressure control conditions.

8. The fact of the high points of the coolant loops being normally covered with secondary coolant supplied by main or auxiliary feedwater systems does not, of itself, assure adequate cooling of the core. Other systems must be operable, operator actions must not interfere with the system's necessary function, and conditions conducive to maintenance of natural circulation must be present. This has not been demonstrated at Diablo Canyon nor have the Emergency Operating Procedures been fully and adequately prepared.

9. The condensation of steam in the coolant loops with no loss of natural circulation has not been demonstrated at Diablo Canyon.

10. If sufficient steam were present, reactor coolant conditions would change from single phase natural circulation

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4/ NUREG-0737, Clarification of TMI Action Plan Requirements, p. 3-86.

to some two-phase mixture. If adequate cooling is provided, it would achieve a two-phase boiling condensation condition.

11. Loss of natural circulation could be blocked in U-tube steam generators if secondary cooling to the steam generators is inadequate.

12. Natural circulation tests performed at the LOFT and Semiscale facilities have not been shown to be directly applicable at Diablo Canyon through actual demonstrations at that plant.

13. The safety classification of PORV's and block valves and their associated instruments and controls is not clearly defined in the FSAR for Diablo Canyon, nor is it clear what the Applicant means by his use of the term "important to safety" in responses to interrogatories on valve classification. Thus, there is no assurance the valves are properly classified or qualified for their function.

14. The PORV's and block valves are called upon in Emergency Operating Procedures to perform functions related to insuring the integrity of the reactor coolant pressure boundary (both for low temperature over-pressure conditions and operating and accident conditions). However, contrary to our belief, the Applicant and Staff do not consider this as a safety function.

15. The functions of the PORV and block valves include the following:

a. Maintain integrity of the primary pressure boundary.

b. Provide pressure relief for Low Temperature Overpressurization conditions.

c. Reduce the number of challenges to the safety valves.

d. Reduce the number of challenges to the ECCS.

e. Provide a bleed capability during the feed-and-bleed mode of operation to remove decay heat from the core (as, for example, was done during the TMI-2 accident).

Several of these functions are consistent with the functions in 10 CFR 100, Appendix A, Section III.C, which was used by the NRC to define criteria for "safety-related" classification. However, the Applicant contends PORV's and block valves are not relied upon for safety functions.

16. The block valves are used to isolate a PORV and may also be used to provide throttling capability for back-up reactor coolant pressure control and for control of the bleed capability in the bleed-and-feed mode of heat removal following an accident. The Applicant does not consider these as safety functions; we disagree with this position.

17. As the accident at TMI-2 demonstrated, proper operation of PORV's and block valves can be important in mitigating the effects of an accident. They are also called upon in the

Emergency Operating Procedures to provide a means for depressurizing the reactor coolant so that back-up boration techniques may be applied. The EOP's also assume the PORV's will automatically open in an ATWS event, an event which could lead to a major accident although not presently recognized as a design basis event. Block valves are also used to mitigate and control a small LOCA resulting from a failed PORV. Despite these facts, the Applicant and Staff contend the block valves are not required to mitigate the consequences of a DBA.

18. If a PORV failed it would cause a small LOCA. If two or more failed due to a common-mode failure or systems interaction, the effects would be more severe. If the failure should occur simultaneously with a LOCA of other origin it would produce confusing symptoms and indications to the operator, release additional contaminanted coolant to the containment and could result in more severe consequences than a LOCA would otherwise produce. The Staff contends that the simultaneous LOCA and failure of a PORV would not significantly alter the consequences. We believe the impact could be significant.

19. An unisolated stuck-open PORV was the fundamental cause of coolant loss leading to core damage in the TMI-2 accident. Thus, it is impossible to assure that stuck-open PORV's at Diablo Canyon could not lead to core damage. Only under the most ideal conditions (i.e., ignoring systems interaction,

common-mode failures, operator error, and other system failures) can the Staff and Applicant assume no fuel damage will result from a stuck-open PORV. We feel this is an unreasonable assumption.

20. The pressure trip settings for the PORV's is slightly lower than that of the safety valves in order to reduce the number of challenges to the code safety valves. We consider this to be a safety-related function but Staff apparently disagrees.

21. Several Emergency Operating procedures include descriptions of how the operators should go about searching for possible sources of coolant loss; specifically, they instruct operators, in several EOP's, to check for indications of leaking or open PORV's. The operator would then take corrective action, such as closing the block valve. If corrective action is not taken and there was continued leakage without make-up, the coolant pressure and level would drop and core cooling should be automatically initiated. If ECCS is not initiated or if operator action precludes the continuted operation of a coolant source (as occurred at TMI-2), there is no assurance that proper core cooling will occur.

22. Although the operator can isolate a stuck-open PORV by utilizing the block valve, he must first recognize the necessary symptoms, properly diagnose the problem, and then take the proper

action. As has been shown by the experience at TMI, these steps can not be assured when the operators have only partial or misleading information or are predisposed to looking for a different causal event.

Gregory C. Minor

Gregory C. Minor

Dale G. Bridenbaugh

January 11, 1982

Dale G. Bridenbaugh

Subscribed and sworn before me this 11th day of January, 1982.

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

RELATED CORRESPONDENCE

EXHIBIT 3

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.

PREPARED DIRECT TESTIMONY OF

DALE G. BRIDENBAUGH AND GREGORY C. MINOR

ON BEHALF OF GOVERNOR EDMUND G. BROWN JR.

REGARDING

CONTENTION 10

January 11, 1982

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PREPARED DIRECT TESTIMONY

OF DALE G. BRIDENBAUGH AND GREGORY C. MINOR

REGARDING CONTENTION 10

I. INTRODUCTION

1. My name is Dale G. Bridenbaugh. I am a Professional Nuclear Engineer, licensed by the State of California, technical consultant, co-founder and president of MHB Technical Associates, technical consultants on energy and environment, with offices at 1723 Hamilton Avenue, Suite K, San Jose, California. I have participated as an expert witness in licensing proceedings before the U.S. Nuclear Regulatory Commission (NRC); have served as a consultant to the NRC; have testified at the request of the Advisory Committee on Reactor Safeguards; have appeared before various committees of the U.S. Congress; and testified in various state licensing and regulatory proceedings. I received a Bachelor of Science in Mechanical Engineering from the South Dakota School of Mines and Technology in 1953. From June, 1953, until February, 1976, I worked as an engineer and manager with the General Electric Company on a wide variety of most of the aspects of power generation equipment design, manufacture and operation. During the last 10 of those 22 years, I was in management positions in the General Electric Nuclear Energy Division where I had the responsibility for managing the monitoring of operation of nuclear

power plants, for the implementation of solutions to nuclear plant operational problems, and for the development of a master performance improvement plan aimed at bringing about the long term improvement of power reactor performance.

2. In my capacity as technical consultant with MHB Technical Associates, I have provided technical advice to various governmental bodies and individual groups on subjects related to the design and operation of commercial nuclear power plants. As examples of this work, in 1978 I served as a consultant to the United States Nuclear Regulatory Commission to review the NRC plan for research to improve the safety of light water nuclear power plants, and have served in various consulting capacities to the United States General Accounting Office, the states of California, Illinois, Massachusetts, New Jersey, Pennsylvania, to Suffolk County, New York, and to the governments of Sweden and Norway, all in the evaluation of nuclear plants or programs. A statement of my qualifications and professional experience is appended to this testimony as Attachment A.

3. My name is Gregory C. Minor. A statement of my qualifications and experience has previously been provided to this Board as part of my testimony on Contention 1 and in Attachment B to that testimony.

II. STATEMENT OF CONTENTION

4. The purpose of our testimony is to respond to Contention 10 as admitted by the Board as follows:1/

> 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 emergency power supplies does not provide an equivalent or acceptable level of protection.

The results of our review of some of the important matters encompassed by this Contention are summarized in the following paragraphs.

III. DISCUSSION OF ISSUES

III.A.: Background and Summary of Position

5. The essence of Contention 10 is that the pressurizer heaters, including the associated heater controls, should be

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1/ ASLB Memorandum and Order, September 30, 1981. On September 21, 1981, the Commission directed the Licensing Board to include in the full power proceeding Joint Intervenors' low power Contention 10.

formally classified as "components important to safety" and, accordingly, be designed, manufactured, and constructed with all the care that should be afforded such components.

6. The origin of this Contention is the experience of the Three Mile Island accident and the subsequent reviews performed to consider its significance. This accident, along with its extended recovery period, demonstrated the need to reconsider the safety classifications and design practices for nuclear systems and components. In particular, the inoperability of the reactor coolant pumps and the low pressure decay heat removal systems emphasized the importance of the ability to remove heat from the reactor via natural circulation and required associated systems. Thus, the NRC Lessons Learned Task Force found that "maintenance of natural circulation capability is important to safety."[2/] The pressurizer heater system is the normal and preferred system for this capability. In addition, the pressurizer heaters must also maintain physical integrity for the reactor coolant pressure boundary to be maintained. While it may be possible to maintain natural circulation at hot standby conditions without use of the pressurizer heater and associated controls, such operation may be difficult to control and is contrary to the normal and emergency plant operating procedures. In this regard, PG&E's response No. 45, dated October 26, 1981,

2/ NUREG-0578, p. A-2.

to Joint Intervenors' Second Set of Interrogatories provided a list of emergency operating procedures that include the use of pressurizer heaters. We have reviewed these procedures and find that "alternate" (to the use of the pressurizer heaters) pressure control methods are not specified for the operators' use. These procedures thus appear to place total reliance on automatic or manual operation of the heaters. We therefore conclude that the heater system has been improperly classified, or the procedures have been inadequately prepared in failing to provide safety-related backup systems, or both may be at fault. 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 controlability of the systems used in the mitigation or control of abnormal events. The logical response to the information gained from the TMI-2 accident, in our opinion, is to classify the pressurizer heater system as important to safety (safety-related) so as to ensure its operability for response to accidents or transient conditions.

7. It is important to place in proper context the intended meaning of the phrase "components important to safety." Contention 10 was formally accepted by the ASLB on September 30, 1981. On November 20, 1981, Harold R. Denton, Director of the NRC's Office of Nuclear Reactor Regulation, issued a Memorandum to clarify the

use of safety classification terms.[3/] This Memorandum stated that safety classification terms had not been consistently applied by the NRC Staff, and that three terms, "important to safety," "safety-grade," and "safety-related," have been used inconsistently or interchangeably. Mr. Denton's Memorandum goes on to identify the recommended usage of these terms. This should serve to make these terms more definitive when used in future licensing; however, our understanding of the usage intended in the Contention 10 language is that the pressurizer heaters and controls should be classified as "safety-related" (as defined in the Denton Memorandum) and should, therefore, be subject to the general requirements of the General Design Criteria (GCD) and that applicability of various GDC's should be judged by the guidance of 10 C.F.R. Part 100, Appendix A.[4/]

3/ Memorandum from H. R. Denton to All NRR Personnel, November 20, 1981, Subject: "Standard Definitions for Commonly-Used Safety Classification Terms," Attachment B hereto.

4/ Our review of the Diablo Canyon pressurizer heater documentation affirms our view that precise safety classification terminology is necessary and significant. The NRC Staff believes the pressurizer heaters are considered "components important to safety" with respect to their pressure control function. NRC Staff Motion for Summary Disposition of Contentions 10 and 12, p. 6. The Applicant believes these components are not required to be classified as "important to safety." Pacific Gas and Electric Company's Motion for Summary Disposition, December 21, 1981, p. 4. The Staff's position would lead to the belief that at least some of the General Design Criteria have been applied, whereas the Applicant's position would indicate that none of the GDC apply (other than those that admittedly apply to the RCPB pressure retaining capability and to the breakers which can be used to connect the heaters to the onsite emergency power system). Applicant's response to NUREG-0578 states that

(Cont'd on next page)

III.B.: Importance of Pressurizer Heaters

8. The pressurizer heater system used at the Diablo Canyon plant provides an important function, namely, the ability to control primary coolant pressure under various conditions. Not only is the system used during normal power operation, but is especially needed for control of pressure and of natural circulation capability in the hot standby mode. The NRC Staff's recommendations emanating from the TMI reviews recognize that maintenance of safe plant conditions depends on maintenance of pressure control in the primary system for the associated maintenance of natural circulation capability. The Staff, therefore, recommended upgrading the pressurizer heaters and associated

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4/ (Cont'd)

equipment identified as non-safety-grade will not be qualified for the Hosgri event, implying that the heaters, therefore, are not seismically qualified. Pacific Gas and Electric Company Response to NUREG-0578, April 21, 1980, p. III-B-5. The Westinghouse specification under which the pressurizer heaters were procured seems to confirm that only the coolant boundary GDC's were applied. Furnished with Applicant Pacific Gas and Electric Company's Supplemental Response to Joint Intervenors' Second Set of Interrogatories, December 23, 1981, Immersion Heater Spec. 393A701. The specification provides no design requirement on the radiation exposure the unit must withstand (the specific concern is the insulating boot at the electrical connection), nor does it address seismic loadings. No information is given on heater sheath supports along the length of the heater (the heater rods are approximately eight feet long and are 7/8" in diameter). These omissions provide little assurance that these important aspects have been adequately considered so as to produce a reliable source of pressure control.

controls to achieve greater reliability.[5/] The NRC Staff's Motion for Summary Disposition states that pressurizer heaters are required to maintain system pressure at the hot standby condition.[6/] PG&E claims that heaters are not required for hot standby pressure control and natural circulation.[7/] We agree with the Staff that the heaters should be used for this function. The basis of this position is that this is the normal control mode, that the procedures specify this mode, and that it is difficult for the operators to follow a different and infrequently used procedure under stressful conditions.

9. PG&E's intended reliance on the pressurizer heaters is indicated by frequent mention of them in the Diablo Canyon Emergency Operating Procedures. No less than nine such procedures call for the use of the pressurizer heater system.[8/] PG&E claims

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5/ NUREG-0578, NRR Lessons Learned Task Force Short-Term Recommendations, page A-2.

6/ NRC Staff Motion for Summary Disposition of Contentions 10 and 12, page 5.

7/ Affidavit of John B. Hoch, page 1, a part of Pacific Gas and Electric Company's Motion for Summary Disposition, December 21, 1981.

8/ Applicant Pacific Gas and Electric Company's Answers to Governor Edmund G. Brown Jr.'s Second Set of Interrogatories, page 47.

that alternate means (to the heater system) for pressure control are available; however, none of the cited emergency operating procedures specifically direct the operator how to proceed with alternatives if the heater system becomes unavailable. (See Paragraph 11 for further discussion of procedural inadequacies.)

10. The NRC Staff states that primary system pressure control is not a prerequisite for natural circulation as the Westinghouse design will provide natural circulation as long as adequate water is provided to the secondary side of the steam generators, even if the primary coolant pressure decays to bring the system to a saturated condition. Applicant and NRC Staff also cite test data obtained at the Sequoyah Nuclear Plant that supports the claim that the Diablo Canyon primary system pressure will decay at about 100 psig per hour if the pressurizer heaters are lost. It has not yet been demonstrated, however, that these characteristics are true at Diablo Canyon. Further, the Applicant has provided no directions in the Emergency Operating Procedures as to how the characteristics would be utilized to assure proper operation. If it is the Applicant's intent to rely upon these claimed reactor characteristics, they should be demonstrated and necessary operator action(s) should be fully described in the procedures.

III.C.: Deficiencies of Present Pressurizer Heater System

11. The purpose of the pressurizer heater system upgrading

required by NUREG-0578 (and 0737) is to assure that primary coolant pressure control will be available when needed. The time when this need is the greatest is during or following transient and/or accident conditions. Emergency Operating Procedure OP-13, Malfunction of Reactor Pressure Control System, is intended to provide guidance on how to maintain primary pressure control when the pressure control devices malfunction. This procedure only assumes control channel failure or failure to deenergize and therefore provides corrective action by placing the system in manual control. No guidance is given as to how to proceed to "feed and bleed" or the other "alternate control methods" claimed by the Applicant. Similarly, EP OP-23, Natural Circulation of Reactor Coolant, has as a basic assumption that offsite power and the heaters are available, making it incomplete for certain accident sequences.

12. PG&E appears to be in a paradoxical situation. On the one hand, PG&E has argued that the pressurizer heaters are not required for natural circulation; rather, other methods are available to ensure that this important cooling mechanism occurs. However, in the Diablo Canyon Emergency Procedures (OP-13 and 23), no other methods are provided for the operators' use. Thus, in our opinion, at a minimum, either the heaters should be upgraded to safety grade or the other methods which presumably rely on safety grade systems should be specified. Since the

other methods are not specified in the procedures at this time, there can be no assurance that Diablo Canyon operators would, in fact, utilize such other systems if the non-safety-grade heaters were unavailable. Thus, the procedures are inadequate or the heaters' classification is inadequate, or both.[9/]

13. Another deficiency affecting pressurizer heater reliability during emergency conditions is the method required to transfer some of the heaters to the onsite emergency power system. The NRC Staff claims that the dispatching of an operator to a remote (the 100 foot level of the Auxiliary Building) location to perform electrical breaker manipulations is an "acceptable alternative" to actually meeting clarification item 4 of TMI requirement II.E.3-1, which specifies that transfer is to be accomplished in the control room.[10/] The Staff does not state how this conclusion was reached, whether or not area radiation monitors will be available to assure immediate access to these areas under accident conditions, or whether they have independently verified the operator radiation exposure of 10 mRem claimed by

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9/ Use of the pressurizer heaters is clearly the preferable method of maintaining natural circulation. Thus, even assuming other methods may exist and assuming they may subsequently be identified in the procedures, we believe the heaters should be upgraded to safety grade to ensure to the extent feasible that this most useful equipment is available.

10/ NUREG-0675, Supplement 14, Safety Evaluation Report, p. 2-21.

Applicant. It also is not clear that the Staff has adequately assessed the potential delays and disruption to area accessibility inherent in a confusing post-accident situation.

III.D.: Impact of Upgrading the Safety Clarification

14. The possibility of upgrading all of the pressurizer heater system components to a "safety-related" classification has been considered in the past and was, in fact, recommended by one of the major NRC groups assembled to review the TMI accident. The recommendations presented included:

> The pressurizer heater system should be classified as safety grade which would assure emergency power availability and protection from failures due to environmental conditions. 11/

This recommendation, if followed, would have required full adherence to all applicable safety requirements and qualification of the components to appropriate seismic and environmental conditions. There are no reasons to believe that such upgrading could not be done (from a "state-of-the-art" standpoint).

15. If safety classification upgrading were to be required, the pressurizer heater system should become more reliable. Plant safety would be improved by the minimization of the number of

11/ Memorandum for J. M. Allan, NRC, from R. D. Martin, NRC, "Operations Team Recommendations," October 10, 1979, p. 23 (emphasis added).

challenges to the system and by the optimization of 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 the preferred components for this capability. It is our opinion that such upgrading would impose more of the safety design criteria on this system and its operability. GDC 20 requires, for example, 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.[12/] 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.[13/] This procedure does not meet the automatic initiation requirements of GDC 20. 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

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12/ NUREG 75/087, Section 7, Table 7-1.

13/ *See* Philip A. Crane to Frank J. Miraglia, January 26, 1981, p. II.3-14.

accordance with GDC 2 (seismic and environmental qualification), GDC 22 (protection system independence, "separation"), or GDC 3 (fire protection). Since these components have not been classified as important to safety, the requirements of GDC 1 (Quality standards and records) does not appear to have been applied.

IV. CONCLUSION

16. The discussion in Part III above indicates a number of reasons why the pressurizer heater system components should be classified as safety-related components. It also indicates some of the benefits to be obtained by such classification. We therefore conclude that this action should be taken at the Diablo Canyon plant.

ATTACHMENT A

EXPERIENCE AND QUALIFICATIONS OF DALE G. BRIDENBAUGH

Dale G. Bridenbaugh is a Professional Nuclear Engineer licensed by the State of California (license NU 973), president of MHB Technical Associates, and a member of the American Nuclear Society. Bridenbaugh received a B.S. in Mechanical Engineering from the South Dakota School of Mines and Technology in 1953. He has been intimately involved with the commercial nuclear power program since 1958, when he was first assigned in a supervisory capacity for the General Electric Company in the construction of the Dresden Nuclear Power Station near Chicago, Illinois. Subsequent to that assignment, he has accumulated over 20 years of nuclear experience, including responsible management of positions in construction, startup, operation, maintenance, and product improvement planning with General Electric's nuclear program. Included in his background experience is the management of the design, construction, and checkout of the first mobile test facility assembled by the General Electric Company for the on-site testing, under simulated environmental operating conditions, of various nuclear system safety and relief valves.

Bridenbaugh has been involved with Pacific Gas and Electric's (PG&E) nuclear plant programs since 1966, when he was assigned responsibility for liason with utilities on all operating nuclear plant matters. This included the ongoing engineering effort by General Electric in support of PG&E's Humboldt Bay No. 3 nuclear unit. After the formation of MHB Technical Associates in 1976, he has participated in the review and licensing process of the Diablo Canyon plant, presenting testimony before the ASLB in 1976 on expected plant capacity factors.

Bridenbaugh has analyzed the operations of numerous nuclear plants in his previous and present positions. He evaluated the response of the Sacramento Municipal Utility District Rancho Seco Plant to equipment and operating procedure recommendations made as a result of the TMI accident. Results of this evaluation were presented in direct testimony on behalf of the California Energy Commission in a hearing before the ASLB on Rancho Seco in 1980. He has testified on similar matters before the ASLB on the Black Fox (Oklahoma) case. He has also served as a consultant to the NRC on the review of the NRC safety improvement program and on the safety goals assessment program.

Bridenbaugh has testified on nuclear safety, reliability, and economic matters before the NRC (Commission and ASLB), before the Joint Committee on Atomic Energy of the United States Congress, and before the energy and utility commissions of Ohio, New York, New Jersey, Massachusetts, and California. He has also served as consultant to private and governmental bodies in Pennsylvania, Massachussetts, New York, Illinois, Texas, Oklahoma, and Oregon, as well as in Sweden, Italy, and Australia. Additional information on the professional qualification of Dale G. Bridenbaugh is set forth in the following:

PROFESSIONAL QUALIFICATIONS OF DALE G. BRIDENBAUGH

DALE G. BRIDENBAUGH
1723 Hamilton Avenue
Suite K
San Jose, CA 95125
(408) 266-2716

EXPERIENCE:

1976 - PRESENT

President - MHB Technical Associates, San Jose, California.
Co-founder and partner of technical consulting firm. Specialists in energy consulting to governmental and other groups interested in evaluation of nuclear plant safety and licensing. Consultant in this capacity to state agencies in California, New York, Illinois, New Jersey, Pennsylvania, Oklahoma and Minnesota and to the Norwegian Nuclear Power Committee, Swedish Nuclear Inspectorate, and various other organizations and environmental groups. Performed extensive safety analysis for Swedish Energy Commission and contributed to the Union of Concerned Scientist's Review of WASH-1400. Consultant to the U.S. NRC - LWR Safety Improvement Program, performed Cost Analysis of Spent Fuel Disposal for the Natural Resources Defense Council, and contributed to the Department of Energy LWR Safety Improvement Program for Sandia Laboratories. Served as expert witness in NRC and state utility commission hearings.

1976 - (FEBRUARY - AUGUST)

Consultant, Project Survival, Palo Alto, California.

Volunteer work on Nuclear Safeguards Initiative campaigns in California, Oregon, Washington, Arizona, and Colorado. Numerous presentations on nuclear power and alternative energy options to civic, government, and college groups. Also resource person for public service presentations on radio and television.

1973 - 1976

Manager, Performance Evaluation and Improvement, General Electric Company - Nuclear Energy Division, San Jose, California.

Managed seventeen technical and seven clerical personnel with responsibility for establishment and management of systems to monitor and measure Boiling Water Reactor equipment and system operational performance. Integrated General Electric resources in customer plant modifications, coordinated correction of causes of forced outages and of efforts to improve reliability and performance of BWR systems.

1973 - 1976 (Cont'd)

Responsible for development of Division Master Performance Improvement Plan as well as for numerous Staff special assignments on long-range studies. Was on special assignment for the management of two different ad hoc projects formed to resolve unique technical problems.

1972 - 1973

Manager, Product Service, General Electric Company - Nuclear Energy Division, San Jose, California.

Managed group of twenty-one technical and four clerical personnel. Prime responsibility was to direct interface and liaison personnel involved in corrective actions required under contract warranties. Also in charge of refueling and service planning, performance analysis, and service communication functions supporting all completed commercial nuclear power reactors supplied by General Electric, both domestic and overseas (Spain, Germany, Italy, Japan, India, and Switzerland).

1968 - 1972

Manager, Product Service, General Electric Company - Nuclear Energy Division, San Jose, California.

Managed sixteen technical and six clerical personnel with the responsibility for all customer contact, planning and execution of work required after the customer acceptance of department-supplied plants and/or equipment. This included quotation, sale and delivery of spare and renewal parts. Sales volume of parts increased from $1,000,000 in 1968 to over $3,000,000 in 1972.

1966 - 1968

Manager, Complaint and Warranty Service, General Electric Company - Nuclear Energy Division, San Jose, California.

Managed group of six persons with the responsibility for customer contacts, planning and execution of work required after customer acceptance of department-supplied plants and/or equipment--both domestic and overseas.

1963 - 1966

Field Engineering Supervisor, General Electric Company, Installation and Service Engineering Department, Los Angeles, California.

Supervised approximately eight field representatives with responsibility for General Electric steam and gas turbine installation and maintenance work in Southern California, Arizona, and Southern Nevada. During this period was responsible for the installation of eight different central station steam turbine generator units, plus much maintenance activity. Work included customer contact, preparation of quotations, and contract negotiations.

1956 - 1963

Field Engineer, General Electric Company, Installation and Service Engineering Department, Chicago, Illinois.

Supervised installation and maintenance of steam turbines of all sizes. Supervised crews of from ten to more than one hundred men, depending on the job. Worked primarily with large utilities but had significant work with steel, petroleum and other process industries. Had four years of experience at construction, startup, trouble-shooting and refueling of the first large-scale commercial nuclear power unit.

1955 - 1956

Engineering Training Program, General Electric Company, Erie, Pennsylvania, and Schenectady, New York.

Training assignments in plant facilities design and in steam turbine testing at two General Electric Factory locations.

1953 - 1955

United States Army - Ordnance School, Aberdeen, Maryland.

Instructor - Heavy Artillery Repair. Taught classroom and shop disassembly of artillery pieces.

1953

Engineering Training Program, General Electric Company, Evendale, Ohio.

Training assignment with Aircraft Gas Turbine Department.

EDUCATION & AFFILIATIONS:

BSME - 1953, South Dakota School of Mines and Technology, Rapid City, South Dakota, Upper ¼ of class.

Professional Nuclear Engineer - California. Certificate No. 0973.

Member - American Nuclear Society.

Various Company Training Courses during career including Professional Business Management, Kepner Tregoe Decision Making, Effective Presentation, and numerous technical seminars.

HONORS & AWARDS:

Sigma Tau - Honorary Engineering Fraternity.

General Managers Award, General Electric Company.

PERSONAL DATA:

Born November 20, 1931, Miller, South Dakota.
Married, three children
6'2", 190 lbs., health - excellent
Honorable discharge from United States Army
Hobbies: Skiing, hiking, work with Cub and Boy Scout Groups.

PUBLICATIONS & TESTIMONY:

1. Operating and Maintenance Experience, presented at Twelfth Annual Seminar for Electric Utility Executives, Pebble Beach, California, October 1972, published in General Electric NEDC-10697, December 1972.

2. Maintenance and In-Service Inspection, presented at IAEA Symposium on Experience From Operating and Fueling of Nuclear Power Plants, Bridenbaugh, Lloyd & Turner, Vienna, Austria, October, 1973.

3. Operating and Maintenance Experience, presented at Thirteenth Annual Seminar for Electric Utility Executives, Pebble Beach, California, November, 1973, published in General Electric NEDO-20222, January. 1974.

4. Improving Plant Availability, presented at Thirteenth Annual Seminar for Electric Utility Executives, Pebble Beach, California, November 1973, published in General Electric NEDO-20222, January, 1974.

5. Application of Plant Outage Experience to Improve Plant Performance, Bridenbaugh and Burdsall, American Power Conference, Chicago, Illinois, April 14, 1974.

6. Nuclear Valve Testing Cuts Cost, Time, Electrical World, October, 15, 1974.

7. The Risks of Nuclear Power Reactors: A Review of the NRC Reactor Safety Study WASH-1400, Kendall, Hubbard, Minor & Bridenbaugh, et al, for the Union of Concerned Scientists, August, 1977.

8. Swedish Reactor Safety Study: Barsebäck Risk Assessment, MHB Technical Associates, January, 1978. (Published by the Swedish Department of Industry as Document DsI 1978:1)

9. Testimony of D.G. Bridenbaugh, R.B. Hubbard, G.C. Minor to the California State Assembly Committee on Resources, Land Use, and Energy, March 8, 1976.

10. Testimony of D.G. Bridenbaugh, R.B. Hubbard, and G.C. Minor before the United States Congress, Joint Committee on Atomic Energy, February 18, 1976, Washington, DC (Published by the Union of Concerned Scientists, Cambridge, Massachusetts.)

11. Testimony by D.G. Bridenbaugh before the California Energy Commission, entitled, Initiation of Catastrophic Accidents at Diablo Canyon, Hearings on Emergency Planning, Avila Beach, California, November 4, 1976.

12. Testimony by D.G. Bridenbaugh before the U.S. Nuclear Regulatory Commission, subject: Diablo Canyon Nuclear Plant Performance, Atomic Safety and Licensing Board Hearings, December, 1976.

13. Testimony by D.G. Bridenbaugh before the California Energy Commission, subject: Interim Spent Fuel Storage Considerations, March 10, 1977.

14. Testimony by D.G. Bridenbaugh before the New York State Public Service Commission Siting Board Hearings concerning the Jamesport Nuclear Power Station, subject: Effect of Technical and Safety Deficiencies on Nuclear Plant Cost and Reliability, April, 1977.

15. Testimony by D.G. Bridenbaugh before the California State Energy Commission, subject: Decommissioning of Pressurized Water Reactors, Sundesert Nuclear Plant Hearings, June 9, 1977.

16. Testimony by D.G. Bridenbaugh before the California State Energy Commission, subject: Economic Relationships of Decommissioning, Sundesert Nuclear Plant, for the Natural Resources Defense Council, July 15, 1977.

17. Testimony by D.G. Bridenbaugh before the Vermont State Board of Health, subject: Operation of Vermont Yankee Nuclear Plant and Its Impact on Public Health and Safety, October 6, 1977.

18. Testimony by D.G. Bridenbaugh before the U.S. Nuclear Regulatory Commission, Atomic Safety and Licensing Board, subject: Deficiencies in Safety Evaluation of Non-Seismic Issues, Lack of a Definitive Finding of Safety, Diablo Canyon Nuclear Units October 18, 1977, Avila Beach, California.

19. Testimony by D.G. Bridenbaugh before the Norwegian Commission on Nuclear Power, subject: Reactor Safety/Risk, October 26, 1977.

20. Testimony by D.G. Bridenbaugh before the Louisiana State Legislature Committee on Natural Resources, subject: Nuclear Power Plant Deficiencies Impacting on Safety & Reliability, Baton Rouge, Louisiana, February 13, 1978.

21. Spent Fuel Disposal Costs, report prepared by D.G. Bridenbaugh for the Natural Resources Defense Council (NRDC), August 31, 1978.

22. Testimony by D.G. Bridenbaugh, G.C. Minor, and R.B. Hubbard before the Atomic Safety and Licensing Board, in the matter of the Black Fox Nuclear Power Station Construction Permit Hearings, September 25, 1978, Tulsa, Oklahoma.

23. Testimony of D.G. Bridenbaugh and R.B. Hubbard before the Louisiana Public Service Commission, Nuclear Plant and Power Generation Costs, November 19, 1978, Baton Rouge, Louisiana.

24. Testimony by D.G. Bridenbaugh before the City Council and Electric Utility Commission of Austin, Texas, Design, Construction, and Operating Experience of Nuclear Generating Facilities, December 5, 1978, Austin, Texas.

25. Testimony by D.G. Bridenbaugh for the Commonwealth of Massachusetts, Department of Public Utilities, Impact of Unresolved Safety Issues, Generic Deficiencies, and Three Mile Island-Initiated Modifications on Power Generation Cost at the Proposed Pilgrim-2 Nuclear Plant, June 8, 1979.

26. Improving the Safety of LWR Power Plants, MHB Technical Associates, prepared for U.S. Dept. of Energy, Sandia Laboratories, September 28, 1979.

27. BWR Pipe and Nozzle Cracks, MHB Technical Associates, for the Swedish Nuclear Power Inspectorate (SKI), October, 1979.

28. Testimony of D.G. Bridenbaugh and G.C. Minor before the Atomic Safety and Licensing Board, in the matter of Sacramento Municipal Utility District, Rancho Seco Nuclear Generating Station following TMI-2 accident, subject: Operator Training and Human Factors Engineering, for the California Energy Commission, February 11, 1980.

29. Italian Reactor Safety Study: Caorso Risk Assessment, MHB Technical Associates, for Friends of the Earth, Italy, March, 1980.

30. Decontamination of Krypton-85 from Three Mile Island Nuclear Plant, H. Kendall, R. Pollard, & D.G. Bridenbaugh, et al, The Union of Concerned Scientists, delivered to the Governor of Pennsylvania, May 15, 1980.

31. Testimony by D.G. Bridenbaugh before the New Jersey Board of Public Utilities, on behalf of New Jersey Public Advocate's Office, Division of Rate Counsel, Analysis of 1979 Salem-1 Refueling Outage, August, 1980.

32. Position Statement, Proposed Rulemaking on the Storage and Disposal of Nuclear Waste, Joint Cross-Statement of Position of the New England Coalition on Nuclear Pollution and the Natural Resources Defense Council, September, 1980.

33. Testimony by D.G. Bridenbaugh and Gregory C. Minor, before the New York State Public Service Commission, In the Matter of Long Island Lighting Company Temporary Rate Case, prepared for the Shoreham Opponents Coalition, September 22, 1980, Shoreham Nuclear Plant Construction Schedule.

34. Supplemental Testimony by D.G. Bridenbaugh before the New Jersey Boad of Public Utilities, on behalf of New Jersey Public Advocate's Office, Division of Rate Counsel, Analysis of 1979 Salem-1 Refueling Outage, December, 1980.

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

NOV 2 0 1981

MEMORANDUM FOR: All NRR Personnel

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

SUBJECT: STANDARD DEFINITIONS FOR COMMONLY-USED SAFETY CLASSIFICATION TERMS

Litigation of one of the principal issues in the TMI-1 Restart Hearing brought to light the fact that there is not complete consistency among all elements of the NRR staff in the application of safety classification terms used frequently in the conduct of NRR's safety review and licensing activities. More specifically, it appears that terms "important to safety," "safety grade," and "safety-related" have been used at times interchangeably, or in ways not completely consistent with the definitions and usage of such terms in the regulations, and which do not fully reflect the intent of the regulations or current licensing practice.

Efforts have been underway for some months now to develop guidance for the consistent usage of these terms. These efforts have included: (a) review of a large number of Reg Guides and SRP's, in conjunction with parts of the regulations upon which they are based, for consistency in the application of safety classification terminology, (2) extensive discussions among cognizant NRR, RES (Stds. Devel.) and ELD representatives regarding proper interpretation and application of such terms, including consideration of alternative "standard" definitions and (3) consultation with the cognizant ACRS Subcommittee regarding these matters, and consideration by the full ACRS as well.

As a result of these efforts, I am endorsing and prescribing for use by all NRR personnel the standard definitions set forth in the enclosure to this letter. It should be noted that in connection with long-term efforts to develop means for ranking reactor plant systems with respect to degree of importance to safety, and in connection with related efforts to develop a graded Q.A. approach in reactor licensing, the general question of safety classifications and safety classification terminologies will be reexamined; and this could result in changes to the definitions set forth in the enclosure or perhaps in development of a completely new scheme in this regard. For the time being, however, the definitions in the enclosure should be considered "standard" and should be applied consistently by all NRR personnel in all aspects of our safety review and licensing activities and should be appropriately reflected in our regulatory guidance documents.

It is expected that minor editorial revisions will have to be made to some existing Reg Guides and SRP's in order to make their wording consistent with these definitions. You should review the regulatory guidance documents within your purview in this regard and recommend the necessary changes; it is not expected that this will involve extensive revision efforts. I want to make clear that my interest here is only in establishing consistency in the language used by all cognizant groups within NRR in expressing our technical requirements. It is not my intention by this action to dictate new technical requirements, to modify existing technical requirements, or to broaden the existing scope of NRR licensing review.

Harold Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosure:
Definition of Terms

DEFINITION OF TERMS

Important to Safety

- Definition - From 10 CFR 50, Appendix A (General Design Criteria) - see first paragraph of "Introduction."

  "Those structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public."

- Encompasses the broad class of plant features, covered (not necessarily explicitly) in the General Design Criteria, that contribute in important way to safe operation and protection of the public in all phases and aspects of facility operation (i.e., normal operation and transient control as well as accident mitigation).

- Includes Safety-Grade (or Safety-Related) as a subset.

Safety-Related

- Definition - From 10 CFR 100, Appendix A - see sections III.(c), VI.a.(1), and VI.b.(3).

  Those structure, systems, or components designed to remain functional for the SSE (also termed 'safety features') necessary to assure required safety functions, i.e.:

  (1) the integrity of the reactor coolant pressure boundary;

  (2) the capability to shut down the reactor and maintain it in a safe shutdown condition; or

  (3) the capability to prevent or mitigate the consequences of accidents which could result in potential off-site exposures comparable to the guideline exposures of this part.

- Subset of "Important to Safety"

- Regulatory Guide 1.29 providesan LWR-generic, function-oriented listing of "safety-related" structures, systems, and components needed to provide or perform required safety functions. Additional information (e.g., NSSS type, BOP design A-E, etc.) is needed to generate the complete listing of safety-related SSC's for any specific facility.

  Note: The term "safety-related" also appears in 10 CFR 50, Appendix B (Q.A. Program Requirements); however, in that context it is framed in somewhat different language than its definition in 10 CFR 100, Appendix A. That difference in language between the two appendices has contributed to confusion and misunderstanding regarding the exact meaning of "safety-related" and its relationship to "important to safety" and "safety-grade." A revision to the language of Appendix B has been proposed to clarify this situation and remove any ambiquity in the meaning of these terms.

Safety-Grade

- Term not used explicitly in regulations but widely used/applied by staff and industry in safety review process.

- Equivalent to "Safety-Related," i.e., both terms apply to the same subset of the broad class "Important to Safety."

RELATED CORRESPONDENCE

EXHIBIT 4

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.

PREPARED DIRECT TESTIMONY OF

DALE G. BRIDENBAUGH AND GREGORY C. MINOR

ON BEHALF OF GOVERNOR EDMUND G. BROWN JR.

REGARDING

CONTENTION 12

January 11, 1982

~~8201200325~~

PREPARED DIRECT TESTIMONY

OF DALE G. BRIDENBAUGH AND GREGORY C. MINOR

REGARDING CONTENTION 12

I. INTRODUCTION

1. My name is Dale G. Bridenbaugh. A statement of my qualifications and experience has previously been provided to this Board as part of my testimony on Contention 10 and in Attachment A to that testimony.

2. My name is Gregory C. Minor. A statement of my qualifications and experience has previously been provided to this Board as part of my testimony on Contention 1 and in Attachment B to that testimony.

II. STATEMENT OF CONTENTION

3. The purpose of our testimony is to respond to Contention 12 as admitted by the Board as follows:[1/]

> 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,

---

1/ ASLB Memorandum and Order, September 30, 1981. On September 21, the Commission directed the Licensing Board to include in the full power proceeding Joint Intervenors' low power Contention 12.

> these valves must be classified as components important to safety and required to meet all safety-grade design criteria.

Further, the Appeal Board's order of December 11, 1981, expands Contention 12 to include "the testing and verification of these same components" since "testing and verification of these components is an integral part of the qualification process."[2/] Thus, the adequacy of the qualification process, including the adequacy of the EPRI testing program, is included in the expanded scope of Contention 12. The results of our review of some of the important matters encompassed by this Contention are summarized in the following paragraphs.

III. DISCUSSION OF ISSUES

III.A.: The NRC's Criteria for Equipment Classification are Confused

4. There is confusion as to the meaning of terms used to describe the safety significance of structures, systems, and components in nuclear power plants. The NRC issued a memorandum[3/]

2/ ASLAB Order, December 11, 1981, p. 3.

3/ Memorandum from H. R. Denton to All NRC Personnel, November 20, 1981, Subject: "Standard Definitions for Commonly-used Safety Classification Terms."

which provided definitions of the most often used safety classification terms as follows (see Attachment A for the full text):

> Important to Safety:
>
> Those structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.
>
> Safety-Related:
>
> Those structures, systems, or components designed to remain functional for the SSE (also termed 'safety features') necessary to assure required safety functions, i.e.,:
>
> (1) the integrity of the reactor coolant pressure boundary;
>
> (2) the capability to shut down the reactor and maintain it in a safe shutdown condition; or
>
> (3) the capability to prevent or mitigate the consequences of accidents which could result in potential off-site exposures comparable to the guideline exposures of this part.
>
> Safety-Grade:
>
> Term not used explicity in regulations but widely used/applied by staff and industry in safety review process.
>
> Equivalent to "Safety-Related," i.e., both terms apply to the same subset of the broad class "Important to Safety."

The writing of Contention 12 preceded the issuance of the clarification document. If Contention 12 had been written using the definitions of the Denton memo, the term "safety-related" would have been used instead of "important to safety." From our review of the Applicant's submissions, it is unclear to us how the Applicant is using the safety classification terms and how it defines "important to safety."

III.B.: <u>The Safety Significance of the PORVs' and Block Valves' Functions Justifies Safety-Related Classification</u>

5. The design of Diablo Canyon includes 3 PORV's and 3 associated block valves. Two of the relief valves are described by PG&E as "important to safety" and, the third is not, having been added to provide capability for 100% load rejection without reactor trip.[4/] The three block valves are also described as "important to safety."

6. The PORV's and/or Block Valves perform several functions which have safety significance along the lines of one or more of the definitions in paragraph 4. These functions are:

a. Maintain integrity of the primary pressure boundary.

b. Provide pressure relief for Low Temperature Overpressurization conditions.

c. Reduce the number of challenges to the safety valves.

d. Reduce the number of challenges to the ECCS.

e. Provide a bleed capability during the feed-and-bleed mode of operation to remove decay heat from the core. 5/

4/ PG&E response to Interrogatory No. 46. (PG&E's response includes the term "local rejection" which is interpreted as a typographical error for "load rejection".)

5/ As used in the TMI-2 accident and as referred to in NUREG-0578, Sec. 2.2.1 and page A-1.

Each of these functions is consistent with the definitions of "important-to-safety," and the first two functions are also consistent with the definitions of "safety-related."

7. The FSAR is vague as to the safety classification of the PORV's, Block valves, and their circuits and controls. The Applicant has stated that the qualification level of the three PORV's and their circuits are not all identical. However, documents which the operator relies on for guidance in operating the plant during emergency conditions (Emergency Operating Procedures) and deciding on an acceptable plant configuration (Diablo Canyon Technical Specifications) provide no evidence of differentiation between the greater or less "qualified valves or associated equipment."

8. The Block valves and/or PORV's are called upon to be operated or checked for misoperation in several of the Emergency Operating Procedures. For example, EOP-20 calls for checking the PORV's as a possible source of excessive leakage from the coolant system (i.e., a small LOCA). EOP-38 (ATWT) describes the need for automatic opening of the PORV's and checking later to see that they are not stuck open in the event of a pressure decay and coolant loss. EOP-2 describes the actions to prevent challenges to the pressurizer safety valves in the case of loss of secondary coolant. It too mentions that the transient may cause the PORV's to open and requires that their resetting be checked, thus insuring against a small LOCA in the primary coolant.

9. The emergency operating procedure for Emergency Shutdown (OP-22) describes conditions where the use of a PORV/BV combination may be needed to depressurize the primary loop so the safety injection pumps may be used for boration. The PORV would be opened and the block valve used for throttling the flow. The procedure does not restrict the operator to any particular PORV nor does it identify a safety-grade alternative component to accomplish the task. Thus any of the PORV/BV combinations should be able to accomplish this safety-related task.

10. Emergency Procedure OP-13 on Malfunction of Reactor Pressure Control System calls for use of a PORV/BV combination as a back-up pressure control technique. The same procedure identifies techniques for finding stuck open PORV's which may be leaking coolant and exacerbating normal pressure control methods.[6]

11. Section 3.4.9.3 of the Diablo Canyon Technical Specifications requires that two PORV's be operable during Hot Shutdown (Mode 4) conditions for overpressure protection. There is no guidance, however, to the operator to choose the more qualified

---

6/ It also notes that a stuck open PORV is designated as an "UNUSUAL EVENT" and requires notification of offsite personnel per the emergency procedures (EOP General Appendix 2 - Notification of Offsite Personnel in the Event of an Emergency).

PORV's. Thus, all PORV's should be qualified to the same level or the operators' EOP's should restrict which two valves are to be used.

12. During operating modes 1, 2 and 3 (Power Operation, Startup, and Hot Standby), Tech Spec Section 3.4.4 requires that each PORV must be operable or isolated by an operable block valve which is then deenergized. For these cases either the PORV's or their associated block valves are relied upon to protect the integrity of the primary pressure boundary. However, according to the Technical Specification, it is possible to block the two higher-qualified valves and rely only on the lesser-qualified valve and its associated controls. We feel there should be instructions to the operator to prevent this situation if the difference in valve classification continues to exist.

III.C.: The Fact that Diablo Canyon Has More PORV's than Other Plants is Not Necessarily an Advantage

13. Since it is permissible to operate with one or more of the PORV's isolated by their block valves, and there are no restrictions on which valves are isolated, it is possible that the PORV with the lesser classified components would be the only valve operable at a time when PORV operation was called upon by a transient or accident.

14. The fact that the Diablo Canyon design has more valves than most plants is commendable but it is not always a virtue. The addition of the third valve may help the reactor ride through a load rejection transient, thus preventing a challenge to the protection system, but it also creates additional failure points which could result in a small LOCA, additional common mode failure mechanisms, and the possibility of systems interaction which could impact other safety-related functions.

III.D.: Qualification of PORV's and BV's is Incomplete

15. Proper safety classification of the PORV/BV and their controls and instruments should insure proper design and qualification for worst case conditions and plant-specific evaluations.[7/]

16. However, the qualification of the Diablo Canyon PORV's and Block Valves is incomplete. The BV's have not been fully tested, and there apparently are no plans for further testing.[8/] The full range of conditions, including ATWS, has not

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7/ See June 16, 1981 PG&E Memorandum relating to EPRI safety valve testing, raising a potential issue regarding Diablo Canyon safety valves. Such testing designed to ensure qualification of valves will increase reliability of and confidence in Diablo Canyon systems.

8/ See NRC Staff Response to Joint Intervenors' Second Set of Interrogatories, No. 61(e).

been tested and the plant-specific analysis has not been prepared to cover Diablo Canyon's design of PORV/BV's and their components, systems, and structures. Thus there can be no assurance that the configuration meets GDC 2 and 14. Also, the scheduled completion of the valve tests and the plant-specific analyses have been delayed until July 1, 1982.[9] This may not be soon enough to satisfy the terms of the Low Power Testing License, Section 1, p. 6.

IV. CONCLUSIONS

17. Based on the functions and required operations of the PORV's and Block valves, as described above, and according to the NRC definitions of safety terms, the PORV's/BV's and their instruments, controls and structures, should all be classed as "safety-related."[10]

18. There are insufficient test data and plant-specific analyses to show that the Diablo Canyon PORV/BV's and associated equipment and structures have been properly qualified.

9/ Ibid, No. 61(d).

10/ "Safety-grade" is also appropriate since it is defined as equivalent to "safety-related" by the NRC.

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

NOV 2 0 1981

MEMORANDUM FOR: All NRR Personnel

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

SUBJECT: STANDARD DEFINITIONS FOR COMMONLY-USED SAFETY CLASSIFICATION TERMS

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Harold R. Denton, Director
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Definition of Terms

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EXHIBIT 5

RELATED CORRESPONDENCE

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. (Low Power Test Proceeding

OPPOSITION OF GOVERNOR EDMUND G. BROWN JR.
TO THE NRC STAFF AND
PACIFIC GAS AND ELECTRIC COMPANY MOTIONS
FOR RECONSIDERATION AND SUMMARY DISPOSITION

JAN 15 1982
Office of the Secretary
Docketing & Service
Branch

Byron S. Georgiou
Legal Affairs Secretary
Governor's Office
State Capitol
Sacramento, California 95814

Herbert H. Brown
Lawrence Coe Lanpher
HILL, CHRISTOPHER AND PHILLIPS, P.C.
1900 M Street, N.W.
Washington, D.C. 20036

Attorneys for Governor Edmund G. Brown, Jr. of the State of Californ:

April 24, 1981

INCOMPLETE

IV. The NRC Staff and PG&E Motions for Summary Disposition Must Be Denied.

The Staff and PG&E motions request summary disposition of all of Governor Brown's Subjects: emergency planning,[7] water level indicators,[8] and relief and block valves.[9] The Governor demonstrates below that the Staff and PG&E motions must be denied.

Well-settled law and NRC administrative practice require that the PG&E motion be denied summarily. Section 2.749(a) of the NRC's regulations requires that the movant annex to its summary disposition motion "a separate, short and concise statement of the material facts as to which the moving party contends that there is no genuine issue to be heard." This is a mandatory provision of law. See Houston Lighting and Power Co. (Allens Creek Nuclear Generating Station), ALAB-629, CCH Nuc. Reg. Rpt. ¶ 30,562 (1981). PG&E completely ignores this mandatory requirement.

PG&E cannot complain about this Board's summary denial of PG&E's motion. In the Stanislaus case, where PG&E similarly was the applicant and the movant for summary disposition, PG&E was chided for failing to comply with the very same summary disposition requirement. In Stanislaus, the Board stated:

7/ Governor Brown Subject 3 and Joint Intervenors' Contentions 4 and 5.

8/ Governor Brown Subject 13 and Joint Intervenors' Contention 13.

9/ Governor Brown Subject 14 and Joint Intervenors' Contention 24.

> Subsection (a) [of Section 2.749] clearly requires that "There shall be annexed to the motion a separate short and concise statement of the material facts as to which the moving party contends that there is no genuine issue to be heard." PG&E has failed to file this required statement of material facts. Such a requirement is not merely a procedural technicality, but it is of substantive significance. This statement is necessary in order to impose upon other parties a duty to file a statement of material facts as to which it is contended there exists a genuine issue to be heard under penalty of having uncontroverted material facts deemed to be admitted. It is necessary for the Board to have this information in a readily available form in order to evaluate the merits of a motion for summary disposition. PG&E's lengthy (77 pages plus numerous exhibits) and argumentative motion for summary disposition wholly fails to comply with the requirement of a concise statement of material facts as to which there is no genuine issue. In re Pacific Gas and Electric Co. (Stanislaus Nuclear Project), CCH Nuc. Reg. Rpt. ¶ 30,211 (LBP 1977) (emphasis supplied).

The same situation exists in this case. PG&E has again filed a lengthy and argumentative motion with numerous exhibits that, as the Board stated in Stanislaus, "wholly fails to comply with the requirement of a concise statement of material facts as to which there is no genuine issue." PG&E, like other participants, must adhere to the NRC's regulations. For this blatant violation of regulatory requirements, the Governor submits that PG&E's motion should be summarily denied.

The Staff has also failed to follow strictly the NRC's summary disposition regulations, although the Staff's violation is not so sweeping as that of PG&E. Section 2.749(b) requires that "[a]ffidavits shall set forth such facts as would be admissible in evidence ...." The NRC Staff has violated this requirement. For example, paragraphs 8-10, 12-13, 16-17, and 23 of the Staff's emergency planning facts, as to which the Staff alleges there is