UNITED STATES OF AMERICA
NUCLEAR REGULATION 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 - OL 50-323 - OL

AFFIDAVIT OF WALTON L. JENSEN, JR. ON CONTENTION 10 AND 12

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

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

2. I am responsible for the technical analysis and evaluation of the public health and safety aspects of Reactor Systems.

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

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

8112230211 811221
PDR ADOCK 05000275
G PDR

4. The pressurizer heaters are part of the normal control system which regulates primary system pressure. When the pressurizer heaters are activated, boiling occurs within the pressurizer producing steam which acts to increase reactor system pressure. The reactor system pressure may be reduced by operation of the pressurizer sprays which condense the steam in the pressurizer. (FSAR Chapter 5)

5. Operation of the pressurizer heaters is required to maintain primary system pressure for power operation and hot standby. Tests at the Sequoyah Nuclear Power Plant have demonstrated that the effect of deenergizing the pressurizer heaters would be gradual depressurization of the primary system (100 psig/hour). Letter from L. Mills, TVA, to A. Schwencer, NRC, transmitting the Sequoyah Nuclear Plant Unit 1 special startup test report dated July 29, 1981. (See Attachment B.)

6. The tests at the Sequoyah Nuclear Power Plant demonstrated that natural circulation could be maintained without operation of the pressurizer heaters.

7. Failure of the pressurizer heaters to operate would allow the reactor system to gradually depressurize which, in the absence of any corrective operation action, would eventually cause automatic actuation of the ECCS. The capability to provide emergency power to the pressurizer heaters is available at Diablo Canyon to reduce the number of demands for ECCS to operate in accordance with Item II.E.3.1 of NUREG-0737 and Item 2.1.1 of NUREG-0578. SER Supplement 14.

8. Operation of the pressurizer heaters is not required to bring the plant to cold shutdown which is a safe and stable condition. One of the first actions in operating procedure L-5 for bringing the plant from hot standby to cold shutdown is to turn off the pressurizer heaters. Although the procedure requires that the reactor system pressure be temporary held at approximately 400 psig until the RHR system can be activiated (for which the pressurizer heaters may be utilized) this function could also be accomplished by control of flow from the charging system. The plant may be maintained at cold shutdown without pressurizer heaters.

9. Incorrect operator action following loss of all power to the pressurizer heaters could cause the reactor system to become saturated so that steam would form in the top of the reactor vessel and reactor coolant loops.

   For plants with U-tube steam generators, such as Diablo Canyon, the high points of the coolant loops are the U-bends of the steam generator tubes which are continually covered with secondary coolant supplied by the main or auxiliary feedwater system. Steam formed in the coolant loops of a plant of the Diablo Canyon design would be condensed by the steam generators with no interruption of natural circulation. If sufficient steam were present, the mode of natural circulation would change from single-phase natural covection to two-phase boiling-condensation. Tests at the LOFT and Semiscale facilities have demonstrated that continuous natural circulation (no interruption) will occur at plants equipped with U-tube steam generators in the presence of steam in the coolant loops as long as steam generator cooling remains available. Semiscale results are documented in Report No. EGG-SEMI-5507, "Quick Look Report for Semiscale Mod-2A Test S-NC-2," July 1981. (See Attachment C.) LOFT results are documented in report No. NUREG CR-1570 "Experimental Data Report for LOFT Nuclear Small Break Experiments L3-7," August 1980. (See Attachment D.)

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

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

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

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

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

(4) Only "safety-grade" structures, systems and components are <u>required</u> for the critical accident prevention, safe shutdown, and accident consequences mitigation safety functions identified in Section III.C of Appendix A to 10 CFR Part 100. (Testimony by J. Conran at TMI-1 Restart Hearing in response to UCS Contention 14, See Attachment E.)

12. The critical safety functions identified in 10 CFR 100 which must be provided with safety grade equipment are as follows:

(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 offsite exposure comparable to the guideline exposures of this part.

Operation of the pressurizer heaters and associated controls are not required to provide any of these critical safety functions and the NRC therefore does not require them to be safety grade.

13. <u>Contention 12 (as admitted by the Board in the September 30, 1981 Memorandum and Order)</u>

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

14. The function of the PORVs at Diablo Canyon is to open at the set pressure of 2350 psig and relieve pressurizer steam so as to preclude the necessity of the safety valves from being opened for mild transients. The set pressure for the safety valves is 2485 psig. The PORVs may also be manually opened by the operator at any pressure below their setpoint to provide a backup means of pressure control in accordance with the operating procedures. Manual opening of the PORVs would cause them to relieve steam at the pressure of the reactor system. Reliance is placed on the engineered safety features to mitigate design basis events rather than the PORV. The function of the block valve is to provide the operator a means to isolate a leaking or failed open PORV. (FSAR Chapter 5)

15. Emergency power has been provided to two of the three PORVs and to the three block valves to reduce the number of challenges to safety valves and ECCS during operation in accordance with Item II.G.1 of NUREG-0737 and Item 2.1.1 of NUREG-0578 (see Diablo Canyon SER Supplement 10).

16. The PORVs and associated block valves are not required to provide low temperature protection at Diablo Canyon for the first fuel cycle (see Diablo Canyon SER Supplement 6 pages 5-3 to 5-7).

17. Proper operation of the PORVs and block valves is not required to mitigate the consequences of any design basis accident. Analyses of design basis accidents are contained in Chapter 15 of the FSAR.

18. A failure of the PORV or associated instruments and controls which results in inability to isolate the flow path through the valve causes the equivalent of a small-break loss-of-coolant accident. The accident would be terminated by closure of the block valve. which is an immediate action to be taken by the operator in the event of a small-break LOCA. Even if the block valve

were not isolated, the capability of the Emergency Core Cooling System is sufficient to permit safe shutdown of the reactor with no core uncovery or core damage. This is demonstrated by analyses contained in Chapter 15 of the Diablo Canyon FSAR and in Section 3.3 of Volume III of WCAP-9600. (See Attachment F.)

19. Since, a stuck open PORV which is not isolated will not result in damage to the fuel element cladding, the fission products contained in the fuel elements would not escape from the core. The only releases to the public would be from radioactive materials already contained in the primary coolant. This material would include activated corrosion products contained in the primary coolant and fission products which might have leaked into the coolant during operation.

20. The offsite doses to the public for a stuck open unisolated PORV which did not cause fuel failure would be much less than the guidelines of 10 C.F.R. 100.

21. In the event that the PORV was to open inadvertently following a small break in the primary system piping, the effect on the reactor system would be equivalent to increasing the break size. The effect of an increase in break size would fall within the spectrum of small-break sizes already analyzed for Diablo Canyon. The small break spectrum is described in Chapter 15 of the FSAR. Analyses of open PORVs in conjunction with a small break loss of coolant accident are contained in Section 4.2 of Volume III of WCAP-9600. (See Attachment G). These analyses demonstrate that opening the PORVs improved core cooling.

22. The inadvertent opening of a PORV following a large break in the primary system piping would not significantly affect the consequences of a large break LOCA since the size of the flow path provided by an open PORV would not be significant in comparison with that of the break.

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

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

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

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

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

(4) Only "safety-grade" structures, systems and components are required for the critical accident prevention, safe shutdown, and accident consequences mitigation safety function identified in Section III.C of Appendix A to 10 C.F.R. Part 100. Testimony by J. Conran at TMI-1 restart hearing in response to USC Contention 14. (See Attachment E).

25. The Critical Safety functions identified in 10 CFR 100 which must be provided with safety grade equipment are as follows:

(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 offsite exposure comparable to the guideline exposures of this part.

Proper operation of the power operated relief valves, associated block valves and instruments and controls for these valves are not required to provide any of these critical safety functions and the NRC therefore does not require these valves to be safety grade.

Walton L. Jensen, Jr.
Walton L. Jensen, Jr.

Subcribed and sworn to before me
this 21st day of December, 1981

Marilyn Johnston
Notary Public

My Commission expires: July 1, 1982

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

CERTIFICATE OF SERVICE

I hereby certify that copies of NRC STAFF MOTION FOR SUMMARY DISPOSITION OF CONTENTIONS 10 and 12 in the above-captioned proceeding have been served on the following by deposit in the United States mail, first class or, as indicated by an asterisk, through deposit in the Nuclear Regulatory Commission's internal mail system, this 21st day of December, 1981.

John F. Wolf, Esq., Chairman
Administrative Judge
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555 *

Glenn O. Bright, Esq.
Administrative Judge
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555 *

Dr. Jerry Kline
Administrative Judge
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555 *

Elizabeth Apfelberg
1415 Cozadero
San Luis Obispo, California 93401

Philip A. Crane, Jr., Esq.
Pacific Gas and Electric Company
P.O. Box 7442
San Francisco, California 94120

Mr. Frederick Eissler
Scenic Shoreline Preservation
Conference, Inc.
4623 More Mesa Drive
Santa Barbara, California 93105

Mrs. Raye Fleming
1920 Mattie Road
Shell Beach, California 93449

Richard E. Blankenburg
Co-publisher
Wayne A. Soroyan, News Reporter
South County Publishing Company
P.O. Box 460
Arroyo Grande, California 93420

Mr. Gordon Silver
Mrs. Sandra A. Silver
1760 Alisal Street
San Luis Obispo, California 93401

Arthur C. Gehr, Esq.
Snell & Wilmer
3100 Valley Center
Phoenix, Arizona 95073

Paul C. Valentine, Esq.
321 Lytton Avenue
Palo Alto, California 94302

Bruce Norton, Esq.
3216 North 3rd Street
Suite 202
Phoenix, Arizona 85012

Andrew Baldwin, Esq.
124 Spear Street
San Francisco, California 94105

Joel R. Reynolds, Esq.
John R. Phillips, Esq.
Center for Law in the Public Interest
10951 West Pico Boulevard
Third Floor
Los Angeles, CA 90064

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

David S. Fleischaker, Esq.
P.O. Box 1178
Oklahoma City, Oklahoma 73101

Richard B. Hubbard
MHB Technical Associates
1723 Hamilton Avenue - Suite K
San Jose, California 95125

John Marrs, Managing Editor
San Luis Obispo County
Telegram-Tribune
1321 Johnson Avenue
P.O. Box 112
San Luis Obispo, California 93406

Herbert H. Brown
Hill, Christopher & Phillips, P.C.
1900 M Street, N.W.
Washington, D.C. 20036

Harry M. Willis
Seymour & Willis
601 California St., Suite 2100
San Francisco, California 94108

Janice E. Kerr, Esq.
Lawrence Q. Carcia, Esq.
350 McAllister Street
San Francisco, California 94102

Mr. James O. Schuyler
Nuclear Projects Engineer
Pacific Gas and Electric Company
77 Beale Street
San Francisco, California 94106

Atomic Safety and Licensing Appeal Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555 *

Atomic Safety and Licensing Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555 *

Docketing and Service Section
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555 *

Mark Gottlieb
California Energy Commission
MS-18
1111 Howe Avenue
Sacramento, California 95825

Bradley W. Jones

Bradley W. Jones
Counsel for NRC Staff