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

August 5, 1993

Docket Nos. 50-275
and 50-323

Mr. Gregory M. Rueger
Nuclear Power Generation, B14A
Pacific Gas and Electric Company
77 Beale Street, Room 1451
P.O. Box 770000
San Francisco, California 94177

Dear Mr. Rueger:

SUBJECT: CORRECTED COPIES OF ISSUED AMENDMENTS FOR DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 1 (TAC NOS. M77347 AND M77417) AND UNIT NO. 2 (TAC NOS. M77348 AND M77418)

The Commission issued on July 23, 1993, Amendment Nos. 81 and 80 to Facility Operating License Nos. DPR-80 and DPR-82, for the Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, respectively.

These amendments revised TS 3/4.4.4, "Relief Valves," and 3/4.4.9.3, "Overpressure Protection Systems," and their associated bases to implement the recommendations of Generic Letter 90-06. These amendments included additional provisions for power operated relief valves (PORV) and block valve reliability and low temperature overpressure protection.

We have found that there were errors made in these amendments. Pages 3/4 4-9, B 3/4 4-1, and B 3/4 4-16 had the words "Amendment Nos. 81 & 80" mistakenly typed on them in the lower right corner. Those pages were only overleaf pages and were not modified by the amendment request. Please replace the above

DF01

pages with the attached corrected pages which are blank in the lower right corner.

We are sorry if this error has inconvenienced you in any way.

Sincerely,

Original signed by

Sheri R. Peterson, Project Manager
Project Directorate V
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
Pages 3/4 4-9, B 3/4 4-1, and B 3/4 4-16 to Amendment Nos. 81 & 80

cc w/enclosures:
See next page

DISTRIBUTION
Docket Files
NRC & Local PDRs
PDV Reading File
EAdensam
JRoe
TQuay
OGC, 15B18
CGrimes, 11E22
ACRS (10), P-315
DFoster-Curseen
DHagan, 3206
GHill (4), P1-37
OPA, 2G5
OC/LFDCB, 4503
Region V (8)
SPeterson
KPerkins, RV

OFFICE	PDV/LA	PDV/PM	PDV/D
NAME	DFoster-Curseen	SPeterson	TQuay for
DATE	8/4/93	8/5/93	8/5/93

OFFICIAL RECORD COPY DOCUMENT NAME: DC77347.COR pages with the attached corrected pages which are blank in the lower right corner.

We are sorry if this error has inconvenienced you in any way.

Sincerely,

Sheri R. Peterson

Sheri R. Peterson, Project Manager
Project Directorate V
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
Pages 3/4 4-9, B 3/4 4-1,
and B 3/4 4-16 to Amendment
Nos. 81 & 80

cc w/enclosures:
See next page

Mr. Gregory M. Rueger
Pacific Gas and Electric Company

Diablo Canyon

cc:
NRC Resident Inspector
Diablo Canyon Nuclear Power Plant
c/o U.S. Nuclear Regulatory Commission
P. O. Box 369
Avila Beach, California 93424

Dr. Richard Ferguson, Energy Chair
Sierra Club California
6715 Rocky Canyon
Creston, California 93432

Ms. Nancy Culver
San Luis Obispo
Mothers for Peace
P. O. Box 164
Pismo Beach, California 93448

Ms. Jacquelyn C. Wheeler
3303 Barranca Court
San Luis Obispo, California 93401

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

Chairman
San Luis Obispo County Board of
Supervisors
Room 370
County Government Center
San Luis Obispo, California 93408

Mr. Truman Burns
Mr. Robert Kinosian
California Public Utilities Commission
505 Van Ness, Rm. 4102
San Francisco, California 94102

Diablo Canyon Independent Safety Committee
ATTN: Robert R. Wellington, Esq.
Legal Counsel
857 Cass Street, Suite D
Monterey, California 93940

Mr. Steve Hsu
Radiologic Health Branch
State Department of Health Services
Post Office Box 942732
Sacramento, California 94234

Regional Administrator, Region V
U.S. Nuclear Regulatory Commission
1450 Maria Lane, Suite 210
Walnut Creek, California 94596

Mr. Peter H. Kaufman
Deputy Attorney General
State of California
110 West A Street, Suite 700
San Diego, California 92101

Christopher J. Warner, Esq.
Pacific Gas & Electric Company
Post Office Box 7442
San Francisco, California 94120

Mr. John Townsend
Vice President and Plant Manager
Diablo Canyon Power Plant
P. O. Box 56
Avila Beach, California 93424

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a water volume of less than or equal to 1600 cubic feet and two groups of pressurizer heaters each having a capacity of at least 150 kW.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one group of pressurizer heaters inoperable, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the Reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The pressurizer water volume shall be determined to be within its limit at least once per 12 hours.

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by measuring heater group power at least once per 92 days.

4.4.3.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by transferring power from the normal to the emergency power supply and energizing the heaters.

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 All power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or more PORV(s) inoperable because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and maintain power to the block valve(s), otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one or more PORV(s) inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve(s), and

  1. With only one Class 1 PORV OPERABLE, restore at least a total of two Class 1 PORVs to OPERABLE status within the following 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

  2. With no Class 1 PORVs OPERABLE, restore at least one Class 1 PORV to OPERABLE status within 1 hour and follow ACTION b.1, above, with the time requirement of that ACTION statement based on the time of initial loss of the remaining inoperable Class 1 PORV or be in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

c. With one or more block valve(s) inoperable, within 1 hour

  1. Restore the block valve(s) to OPERABLE status, or

  2. Close the PORV(s) and remove power from its associated solenoid.

  Also, comply with ACTION b, as appropriate, for the isolated PORV(s).

d. The provisions of Specification 3.0.4 are not applicable.

BASES

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation, and maintain DNBR above 1.30 during all normal operations and anticipated transients.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing core decay heat even in the event of a bank withdrawal accident; however, a single reactor coolant loop provides sufficient heat removal if a bank withdrawal accident can be prevented, i.e., by opening the Reactor Trip System breakers. Single failure considerations require that two loops be OPERABLE at all times.

In MODE 4, and MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR train provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either RHR or RCS) be OPERABLE.

In MODE 5, with reactor coolant loops not filled, a single RHR train provides sufficient heat removal capability for removing decay heat; but single failure considerations and the unavailability of the steam generator as a heat removing component require that at least two RHR trains be OPERABLE.

The operation of one reactor coolant pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a reactor coolant pump with one or more RCS cold legs less than or equal to 323°F are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by: (1) restricting the water volume in the pressurizer and thereby providing a volume for the reactor coolant to expand into, or (2) restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at 110% of the valve's Setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown.

SAFETY VALVES (Continued)

In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System (relief valves) provides a diverse means of protection against RCS overpressurization at low temperatures.

During operation, all pressurizer Code safety valves must be OPERABLE to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no Reactor trip until the first Reactor Trip System Trip Setpoint is reached (i.e., no credit is taken for a direct Reactor trip on the loss-of-load) and also assuming no operation of the power-operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.3 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady-state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12-hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

3/4.4.4 RELIEF VALVES

In MODES 1, 2, and 3 the power-operated relief valves (PORVs) provide an RCS pressure boundary, manual RCS pressure control for mitigation of accidents, and automatic RCS pressure relief to minimize challenges to the safety valves.

The functions of providing an RCS pressure boundary and manual RCS pressure control for mitigation of accidents such as steam generator tube rupture are the safety-related function of the PORVs in MODES 1, 2, and 3. The capability of the PORV to perform its function of providing an RCS pressure boundary requires that the PORV or its associated block valve is closed. The capability of the PORVs to perform manual RCS pressure control for mitigation of accidents is based on manual actuation and does not require the automatic RCS pressure control function. The automatic RCS pressure control function of the PORVs is not a safety-related function in MODES 1, 2, and 3. The automatic pressure control function limits the number of challenges to the safety valves, but the safety valves perform the safety function of RCS overpressure

PRESSURE/TEMPERATURE LIMITS (Continued)

heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of non-ductile failure, operation limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

LOW TEMPERATURE OVERPRESSURE PROTECTION

The OPERABILITY of both Class 1 PORVs or an RCS vent opening of at least 2.07 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 323°F. Either Class 1 PORV has adequate relieving capability to protect the RCS from overpressurization for all anticipated transients.

LOW TEMPERATURE OVERPRESSURE PROTECTION (Continued)

The Maximum Allowed PORV Setpoint for the LTOPs will be modified, if required, based on the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR Part 50, Appendix H, and in accordance with the schedule in Table 4.4-5.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for the ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. To the extent applicable, the inspection program for these components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.11 REACTOR VESSEL HEAD VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the Reactor Coolant System that could inhibit natural circulation core cooling. The OPERABILITY of a reactor vessel head vent path ensures the capability exists to perform this function.

The valve redundancy of the Reactor Coolant System vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure vent valve power supply or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System Vent Systems are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.