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

January 13, 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: ISSUANCE OF AMENDMENTS FOR DIABLO CANYON NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2 (TAC NOS. M79423 AND M79424)

The Commission has issued the enclosed Amendment No. 75 to Facility Operating License No. DPR-80 and Amendment No. 74 to Facility Operating License No. DPR-82 for the Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, respectively. The amendments respond to your amendment application dated November 15, 1990, as supplemented by letters dated May 19, 1992, and October 8, 1992 (reference License Amendment Request LAR 90-11).

These amendments change the combined technical specifications (TS) for the Diablo Canyon Nuclear Power Plant Unit Nos. 1 and 2 to: (1) revise and relocate the Fire Protection license conditions, (2) relocate the Fire Protection TS and associated bases to plant procedures and the Updated Final Safety Analysis Report, (3) augment the administrative TS to define the fire protection program controls, and (4) revise the condensate storage tank TS and associated bases to clarify the requirement for water sources used by the auxiliary feedwater system.

During our review of LAR 90-11, we noted that Diablo Canyon has no technical specifications for alternate shutdown equipment. This is contrary to the recommendations of NRC Generic Letter 81-12, "Fire Protection Rule (45 FR 76602, November 19, 1980)" and 88-12, "Removal of Fire Protection Requirements from Technical Specifications." Accordingly, we request that you address the TS for alternate shutdown equipment that is relied upon in the event of a control room evacuation in accordance with the guidance of GL 81-12 and 88-12.

9301220247 930113
PDR ADOCK 05000275
P PDR

A copy of the Safety Evaluation by the NRC staff, addressing LAR 90-11, is enclosed. Also enclosed is a related Technical Evaluation Report prepared by a staff consultant, Science Applications International Corporation. A notice of issuance of amendments will be included in the Commission's next regular biweekly Federal Register notice. If you have any questions regarding these issues, please contact me.

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:
1. Amendment No. 75 to License No. DPR-80
2. Amendment No. 74 to License No. DPR-82
3. Safety Evaluation
4. Technical Evaluation Report

cc w/enclosures:
See next page

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A copy of the Safety Evaluation by the NRC staff, addressing LAR 90-11, is enclosed. Also enclosed is a related Technical Evaluation Report prepared by a staff consultant, Science Applications International Corporation. A notice of issuance of amendments will be included in the Commission's next regular biweekly Federal Register notice. If you have any questions regarding these issues, please contact me.

Sincerely,

Sheri R. Peterson

Sheri R. Peterson, Project Manager
Project Directorate V
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3. Safety Evaluation
4. Technical Evaluation Report

cc w/enclosures:
See next page

Mr. Gregory M. Rueger
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 1

DOCKET NO. 50-275

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.75
License No. DPR-80

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Pacific Gas & Electric Company (the licensee) dated November 15, 1990, as supplemented by letters dated May 19, 1992, and October 8, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2), 2.C.(5), and 2.G of License No. DPR-80 are hereby amended to read as follows:

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PDR ADOCK 05000275
P PDR

2.C.(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 75 , are hereby incorporated in the license. Pacific Gas & Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

2.C.(5) Fire Protection

a. PG&E shall implement and maintain in effect all provisions of the approved fire protection plan as discussed in its Final Safety Analysis Report Update, in PG&E's December 6, 1984, Appendix R Report, and in the NRC staff's Fire Protection Evaluation in Supplements 8, 9, 13, 23, and 27 to the Diablo Canyon Safety Evaluation Report, subject to provision b below.

b. PG&E may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

c. Deleted.

2.G. Reporting

PG&E shall report any violations of the requirements contained in Sections 2.C.(3) and 2.C.(4), 2.C.(6) through 2.C.(10), 2.E, and 2.F, of this License within 24 hours. Initial notification shall be made in accordance with the provisions of 10 CFR 50.72 with written follow-up in accordance with the procedures described in 10 CFR 50.73 (b), (c), (d) and (e).

3. This license amendment becomes effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Theodore R. Quay

Theodore R. Quay, Director
Project Directorate V
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 13, 1993

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

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 2

DOCKET NO. 50-323

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 74
License No. DPR-82

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Pacific Gas & Electric Company (the licensee) dated November 15, 1990, as supplemented by letters dated May 19, 1992, and October 8, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2), 2.C.(4), and 2.G of License No. DPR-82 are hereby amended to read as follows:

2.C.(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 74 , are hereby incorporated in the license. Pacific Gas & Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

2.C.(4) Fire Protection (SSER 31, Section 9.6.1 and SSER 32, Section 10)

a. PG&E shall implement and maintain in effect all provisions of the approved fire protection program as discussed in its Final Safety Analysis Report Update, in PG&E's December 6, 1984 Appendix R Report, and in the NRC staff's Fire Protection Evaluation in Supplements 8, 9, 13, 23, 27 and 31 to the Diablo Canyon Safety Evaluation Report, subject to provision b below.

b. PG&E may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

c. Deleted.

2.G. Reporting

Except as otherwise provided in the Technical Specifications or Environmental Protection Plan, the licensee shall report any violations of the requirements contained in Sections 2.C.(3), 2.C.(5) through 2.C.(10), 2.E, and 2.F of this license in the following manner: initial notification shall be made within 24 hours to the NRC Operations Center via the Emergency Notification System with written followup within thirty days in accordance with the procedures described in 10 CFR 50.73(b), (c), and (e).

3. This license amendment becomes effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Theodore R. Quay

Theodore R. Quay, Director
Project Directorate V
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 13, 1993

ATTACHMENT TO LICENSE AMENDMENT NOS. 75 AND 74

FACILITY OPERATING LICENSE NOS. DPR-80 AND DPR-82

DOCKET NOS. 50-275 AND 50-323

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SURVEILLANCE REQUIREMENTS (Continued)

(2) Verifying that each non-automatic valve in the pump flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

(3) Verifying that each non-automatic valve in both steam supplies to the steam turbine-driven pump that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months by verifying that each auxiliary feedwater pump starts and valve opens* as designed automatically upon receipt of an Auxiliary Feedwater Actuation test signal.

*For the steam turbine-driven pump, when the secondary steam supply pressure is greater than 650 psig.

AUXILIARY FEEDWATER SOURCE

LIMITING CONDITION FOR OPERATION

3.7.1.3 The Condensate Storage Tank (CST) shall have a usable volume of at least 164,678 gallons of water with an open flow path to the Auxiliary Feedwater (AFW) pump suction, and the Fire Water Storage Tank (FWST) shall have a usable volume of at least 57,922 gallons of water for one Unit operation and 115,844 gallons of water for two Unit operation with a flow path capable of being aligned to the AFW pump suction.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With the CST usable volume less than 164,678 gallons, or with the CST flow path not open to the AFW pumps suction, within four hours restore the required conditions; or, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With an FWST usable volume less than 57,922 gallons for one Unit operation and 115,844 gallons for two Unit operation, or with the FWST flow path not capable of being aligned to the AFW pump suction, within seven days restore the required FWST conditions; or, be in at least HOT STANDBY, within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The CST volume shall be demonstrated at least once per 12 hours by verifying the usable volume is within its limits.

4.7.1.3.2 The FWST volume shall be demonstrated at least once per 12 hours by verifying the usable volume is within its limits.

4.7.1.3.3 Verify the FWST is capable of being aligned to the Auxiliary Feedwater System by cycling each FWST valve in the flow path necessary for realignment through at least one full cycle once per quarter.

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use; and

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in core components or the core and following repair or maintenance of the source.

4.7.8.1.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microcurie of removable contamination.

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3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix A of 10 CFR Part 100. The instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes."

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR Part 50.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. The normal plant instrument channels specified are suitable for use as post-accident instruments. This capability is consistent with the recommendations of Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," May 1983, and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.3.3.7 CHLORINE DETECTION SYSTEMS

The OPERABILITY of the Chlorine Detection System ensures that sufficient capability is available to promptly detect and initiate protective action in the event of an accidental chlorine release. This capability is required to protect control room personnel and is consistent with the recommendations of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," February 1975.

3/4.3.3.9 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Section relocated to RMCP.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to not more than 105% of its design pressure of 1065 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1968 Edition. The total relieving capacity for all valves on all of the steam lines is 1.645×10^7 lb/hr, which is 113% of the total secondary steam flow of 1.45×10^7 lb/hr at 100% RATED THERMAL POWER for Unit 1 and 110% of the total secondary steam flow of 1.49×10^7 lb/hr at 100% RATED THERMAL POWER for Unit 2. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in Secondary Coolant System steam flow and THERMAL POWER required by the reduced Reactor trip settings of the Power Range Neutron Flux channels. The Reactor Trip Setpoint reductions are derived on the following bases:

$$SP = \frac{(X) - (Y)(V)}{X} \times (109)$$

Where:

SP = Reduced Reactor Trip Setpoint in percent of RATED THERMAL POWER,

V = Maximum number of inoperable safety valves per steam line,

109 = Power Range Neutron Flux-High Trip Setpoint,

X = Total relieving capacity of all safety valves per steam line in lb/hour, and

Y = Maximum relieving capacity of any one safety valve in lb/hour.

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the Auxiliary Feedwater System ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss of off-site power.

Each electric motor-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 440 gpm at a pressure of 1135 psig to the entrance of the steam generators. The steam-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 880 gpm at a pressure of 1135 psig to the entrance of the steam generators. The capacity of one motor-driven AFW pump (440 gpm) delivered to at least two steam generators is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the Residual Heat Removal System may be placed into operation.

3/4.7.1.3 AUXILIARY FEEDWATER SOURCE

The principal function of the Auxiliary Feedwater (AFW) Source is to provide a qualified source of water to the steam generators via the AFW System for removal of decay and sensible heat from the Reactor Coolant System (RCS) through generation and release of steam.

The minimum usable water volume in the Condensate Storage Tank (CST) ensures the availability of sufficient water for cooldown of the RCS to less than 350°F in the event of a total loss of offsite power. This minimum volume is also sufficient to remove decay heat sufficient to maintain the RCS at HOT STANDBY conditions for 8 hours with steam discharge to the atmosphere.

An alternate plant cooldown scenario has been postulated for the loss of offsite power, which assumes a reduced Reactor Coolant System cooldown rate and provides credit only for seismically qualified water sources. The lower rate increases the cooldown time period until the Residual Heat Removal System can be used to remove further decay heat. The capacity of the seismically qualified portion of the CST is less than the total amount of water needed for the extended time period. The Fire Water Storage Tank (FWST) has been identified as the seismically qualified source of additional water in the event of an extended cooldown without offsite power.

With the CST less than the required volume, the volume must be restored to the limit. Four hours provides time to restore the required volume from the condenser, or other source, and is a reasonable time to limit the risk from accidents requiring the plant to cool down.

With the FWST unable to supply the required backup volume of cooling water to the AFW System, the operability of the supply must be restored within seven days. This is considered a reasonable time to limit the risk of an accident which would require the use of the backup volume in addition to the primary volume maintained in the CST. Alternate non-seismically qualified water sources are also available to supply water to supplement the CST volume.

3/4.7.1.4 SPECIFIC ACTIVITY

The limitations on Secondary Coolant System specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of a steam line rupture. This dose also includes the effects of a coincident 1 gpm reactor-to-secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to: (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the Surveillance Requirements is consistent with the assumptions used in the safety analyses.

3/4.7.1.6 STEAM GENERATOR 10% ATMOSPHERIC DUMP VALVES

The Limiting Condition for Operation requirement of four steam generator 10% atmospheric dump valves (ADVs) (PCV-19, PCV-20, PCV-21, and PCV-22) ensures that following a steam generator tube rupture accident subcooling can be achieved, consistent with assumptions used in the steam generator tube rupture analysis, to facilitate equalizing pressures between the Reactor Coolant System and the faulted steam generator. This eliminates further primary to secondary leakage and potential subsequent overfill of the affected steam generator. The analysis assumes that the 10% ADV on the ruptured steam generator is not used, and that the other three 10% ADVs are used for heat removal. The surveillance requirement for the 10% ADVs backup air bottles ensures that the 10% ADVs will be available to mitigate the consequences of a steam generator tube rupture accident concurrent with loss of offsite power.

A backup air bottle pressure of 260 psig provides adequate air to operate as assumed in the analysis. This provides sufficient margin to allow cooldown consistent with the analysis assumptions.

Concurrent with the requirement that a specific number of 10% ADVs be OPERABLE is the requirement that the associated 10% ADV block valves upstream be open. Should an associated 10% ADV block valve be closed, the 10% ADV downstream of that block valve should also be considered inoperable and the applicable ACTION statement shall be entered until such time that the block valve is opened.

Additionally, the requirements of Technical Specification 3.6.3, Containment Isolation Valves, apply to the 10% ADVs.

The Technical Specification is applicable in plant operational Modes 1, 2, and 3 because the 10% ADVs are required to provide the subcooling as necessary to permit primary system depressurization for SGTR accident mitigation.

SNUBBERS (Continued)

Any inspection whose results require a shorter inspection interval will override the previous schedule.

The acceptance criteria are to be used in the visual inspection to determine OPERABILITY of the snubbers. For example, if a fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be declared inoperable and shall not be determined OPERABLE via functional testing.

To provide assurance of snubber functional reliability, one of three functional testing methods is used with the stated acceptance criteria:

1. Functionally test 10% of a type of snubber with an additional 10% tested for each functional testing failure, or

2. Functionally test a sample size and determine sample acceptance or rejection using Figure 4.7-1, or

3. Functionally test a representative sample size and determine sample acceptance or rejection using the stated equation.

Figure 4.7-1 was developed using "Wald's Sequential Probability Ratio Plan" as described in "Quality Control and Industrial Statistics" by Acheson J. Duncan.

Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and, if applicable, snubber life destructive testing was performed to qualify the snubber for the applicable design conditions at either the completion of its fabrication or at a subsequent date. Snubbers so exempted shall be listed in the list of individual snubbers indicating the extent of the exemptions.

The service life of a snubber is established via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life.

3/4.7.8 SEALED SOURCE CONTAMINATION

The limitation on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(a)(3) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.7.11 AREA TEMPERATURE MONITORING

The area temperature limitations ensure that safety-related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause loss of its OPERABILITY. The temperature limits include allowance for an instrument error of 1°F.

3/4.7.12 ULTIMATE HEAT SINK

The OPERABILITY of the Component Cooling Water (CCW) System and the components that it cools is ensured if the CCW temperature remains equal to or less than 132°F during any condition assumed in the safety analysis. One CCW heat exchanger is required in service when the ocean temperature is 64°F or less. Two CCW heat exchangers are required in service when the ocean temperature is greater than 64°F. If the reactor coolant temperature is less than 350°F (MODE 4), one CCW heat exchanger in service is adequate even if the ocean temperature is greater than 64°F.

3/4.7.13 FLOOD PROTECTION

The breakwaters (east and west) provide flood protection to the safety-related auxiliary salt water (ASW) pumps located in the intake structure. The ASW pumps would be flood protected for the events up to and including the probable maximum tsunami and maximum credible wave, if the breakwaters are OPERABLE at a level of 0 feet mean lower low water (MLLW) or above. However, substantial degradation of the breakwaters below MLLW level may result in the ASW pump being flooded under the design basis flood and would require corrective actions to be taken. The ongoing surveillance of the breakwaters will ensure that proper flood protection is provided for the ASW pumps.

6.1 RESPONSIBILITY

6.1.1 The Vice President, Diablo Canyon Operations and Plant Manager, hereinafter called Plant Manager, shall be responsible for overall plant operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Foreman (or during his absence from the Control Room, a designated individual) shall be responsible for the Control Room Command function. A management directive to this effect signed by the Senior Vice President and General Manager, Nuclear Power Generation shall be reissued to all plant personnel on an annual basis.

6.2 ORGANIZATION

6.2.1 OFFSITE AND ONSITE ORGANIZATIONS

Onsite and offsite organizations shall be established for plant operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the FSAR Update.

b. The Plant Manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. The Senior Vice President and General Manager - Nuclear Power Generation shall have corporate responsibilities for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 PLANT STAFF

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1;

b. At least one licensed Operator shall be in the control room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3 or 4, at least one licensed Senior Operator shall be in the Control Room;

c. A Health Physics Technician* shall be on site when fuel is in the reactor;

*The Health Physics Technician position may be unfilled for a period of time not to exceed 2 hours in order to accommodate unexpected absence provided immediate action is taken to fill the position.

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PLANT STAFF (Continued)

d. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Operator or Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation;

e. Deleted.

f. Administrative procedures shall be developed and implemented to limit the working hours of plant staff who perform safety-related functions; e.g., licensed Senior Operators, licensed Operators, Health Physicists, auxiliary operators, and key maintenance personnel.

   Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a nominal 40-hour week while the unit is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance or major plant modifications, on a temporary basis, the following guidelines shall be followed:

   1) An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time;

   2) An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any 7-day period, all excluding shift turnover time;

   3) A break of at least 8 hours should be allowed between work periods, including shift turnover time; and

   4) Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

   Any deviation from the above guidelines shall be authorized by the Plant Manager or his designee, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Plant Manager or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized; and

g. The Operations Manager shall hold a senior reactor operator license.

6.2.3 ONSITE SAFETY REVIEW GROUP (OSRG)

FUNCTION

6.2.3.1 The OSRG shall function to examine plant operating characteristics, NRC issuances, industry advisories, REPORTABLE EVENTS and other sources of plant design and operating experience information, including plants of similar design, which may indicate areas for improving plant safety. The OSRG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities or other means of improving plant safety to the Manager, Nuclear Safety Assessment and Regulatory Affairs.

COMPOSITION

6.2.3.2 The OSRG shall be composed of at least five engineers located on site.

RESPONSIBILITIES

6.2.3.3 The OSRG shall be responsible for maintaining surveillance of plant activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4 The Shift Technical Advisor shall provide advisory technical support to the Shift Foreman in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the plant.

6.3 PLANT STAFF QUALIFICATIONS

6.3 Each member of the plant staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1978 for comparable positions, except for the Radiation Protection Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, Revision 2, April 1987 for Radiation Protection Manager. The licensed Operators and Senior Operators shall also meet or exceed the minimum qualifications of 10 CFR Part 55 and the supplemental requirements specified in Section A of Enclosure 1 of the March 28, 1980 NRC letter to all licensees.

*Not responsible for sign-off function.

6.4 TRAINING

6.4 A retraining and replacement training program for the plant staff shall be maintained under the direction of a designated member of the facility staff and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and 10 CFR Part 55.

6.5 REVIEW AND AUDIT

6.5.1 TECHNICAL REVIEW AND CONTROL ACTIVITIES

6.5.1.1 Each procedure and program required by Specification 6.8 and other procedures, tests, and experiments that affect nuclear safety, and changes thereto, shall be prepared by a qualified individual/group. Each such procedure, test, and experiment, and changes thereto, shall be reviewed by an individual/group other than the individual/group which prepared the procedure, test, or experiment, or changes thereto, but who may be from the same organization as the individual/group which prepared the procedure, test, or experiment, or changes thereto.

6.5.1.2 Individuals responsible for reviews performed in accordance with Specifications 6.5.1.1 shall be previously designated by the Plant Manager to perform such reviews. Such designation shall include the disciplines or procedure categories for which each individual is qualified. Each individual designated to perform these reviews shall have at least 3 years of related experience and a baccalaureate degree in engineering or a related field, or shall have at least 8 years of related experience.

6.5.2 PLANT STAFF REVIEW COMMITTEE (PSRC)

FUNCTION

6.5.2.1 The Plant Staff Review Committee shall function to advise the Plant Manager on all matters related to nuclear safety.

COMPOSITION

6.5.2.2 The PSRC shall be chaired by the Plant Manager and shall be composed of a minimum of 8 senior plant management individuals whose responsibilities include the functional areas of: operations, maintenance, radiation protection, support services, technical services, and quality control. All members shall be appointed in writing by the PSRC Chairman. The qualifications of each PSRC member shall meet or exceed the requirements and recommendations of Section 4.7 of ANSI/ANS 3.1-1978.

ALTERNATES

6.5.2.3 The Chairman may designate in writing other regular members who may serve as the Acting Chairman of PSRC meetings. All alternate members shall be appointed in writing by the PSRC Chairman. Alternates may be designated for specific PSRC members and shall have expertise in the same general area as the regular PSRC member they represent. No more than two alternates shall participate as voting members in PSRC activities at any one time.

RESPONSIBILITIES

6.5.2.6 The Plant Staff Review Committee shall be responsible for:

a. Review of: (1) Administrative procedures, Security Plan implementing procedures, Emergency Plan implementing procedures, and changes thereto; (2) the safety evaluations for: (a) changes to procedures and (b) tests or experiments completed under the provision of 10 CFR 50.59, to verify that such actions do not constitute an unreviewed safety question and (3) proposed procedures or changes thereto that have been initially determined to constitute an unreviewed safety question or require a change to the Technical Specifications;

b. Review of all proposed changes to Appendix "A" Technical Specifications;

c. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety;

d. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Senior Vice President and General Manager, Nuclear Power Generation and to the Chairman of the Nuclear Safety Oversight Committee (NSOC); the investigation shall include an assessment of the safety significance of each violation.

e. Review of all REPORTABLE EVENTS;

f. Review of plant operations to detect potential nuclear safety hazards;

g. Performance of special reviews, investigations or analyses and reports thereon as requested by the Chairman of NSOC;

h. Review of the Security Plan and implementing procedures and shall submit recommended changes to the Chairman of the NSOC or the Plant Manager, as appropriate;

i. Review of the Emergency Plan and implementing procedures and shall submit recommended changes to the Chairman of the NSOC or the Plant Manager, as appropriate;

j. Review of any accidental, unplanned, or uncontrolled radioactive release including the preparation and forwarding of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence to the Senior Vice President and General Manager, Nuclear Power Generation and to NSOC; and

k. Review of changes to the PROCESS CONTROL PROGRAM, ODCP, ERMP, and the Radwaste Treatment Systems.

l. Review of the Fire Protection Program and revisions thereto.

RESPONSIBILITIES (Continued)

6.5.2.7 The Plant Staff Review Committee shall:

a. Recommend to the Plant Manager written approval or disapproval of items considered under Specification 6.5.2.6a. through d. and l. above;

b. Render determinations in writing with regard to whether or not each item considered under Specification 6.5.2.6a. through e. above constitutes an unreviewed safety question; and

c. Provide written notification within 24 hours to the Senior Vice President and General Manager, Nuclear Power Generation and NSOC of disagreement between the PSRC and the Plant Manager; however, the Plant Manager shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1 above.

RECORDS

6.5.2.8 The Plant Staff Review Committee shall maintain written minutes of each PSRC meeting that, at a minimum, document the results of all PSRC activities performed under the responsibility and authority provisions of these Technical Specifications. Copies shall be provided to the Senior Vice President and General Manager, Nuclear Power Generation and NSOC.

6.5.3 NUCLEAR SAFETY OVERSIGHT COMMITTEE (NSOC)

FUNCTION

6.5.3.1 The NSOC shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations,

b. Nuclear engineering,

c. Chemistry and radiochemistry,

d. Metallurgy,

e. Instrumentation and control,

f. Radiological safety,

g. Mechanical and electrical engineering, and

h. Quality assurance practices.

NSOC shall report to and advise the Senior Vice President and General Manager, Nuclear Power Generation on those areas of responsibility specified in Specifications 6.5.3.7 and 6.5.3.8.

COMPOSITION

6.5.3.2 NSOC shall be composed of a chairman and a minimum of four members. The NSOC Chairman and members shall be appointed in writing by the Senior Vice President and General Manager, Nuclear Power Generation. The NSOC Chairman shall have a minimum of six years of professional level managerial experience in the power field and the NSOC members shall have a minimum of five years of professional level experience in the field of their specialty. The NSOC Chairman and all members shall have qualifications that meet or exceed the requirements and recommendations of Section 4.7 of ANSI/ANS 3.1-1978.

ALTERNATES

6.5.3.3 This section deleted.

CONSULTANTS

6.5.3.4 Consultants shall be utilized as determined by the NSOC Chairman to provide expert advice to NSOC.

MEETING FREQUENCY

6.5.3.5 NSOC shall meet at least once per 6 months.

QUORUM

6.5.3.6 A quorum of NSOC necessary for the performance of the NSOC functions of these Technical Specifications shall be a majority (one-half or more) of the members, but no less than four. No more than a minority of the quorum shall have line responsibility for operation of the plant.

REVIEW

6.5.3.7 NSOC shall review:

a. The safety evaluations for: (1) changes to procedures, equipment or systems, and (2) tests or experiments completed under the provision of 10 CFR 50.59, to verify that such actions did not constitute an unreviewed safety question;

REPORTABLE EVENT ACTION (Continued)

b. Each REPORTABLE EVENT shall be reviewed by the PSRC and the results of this review submitted to NSOC and the Senior Vice President and General Manager, Nuclear Power Generation.

6.7 SAFETY LIMIT VIOLATION

6.7 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Senior Vice President and General Manager, Nuclear Power Generation and NSOC shall be notified within 24 hours;

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PSRC. This report shall describe: (1) applicable circumstances preceding the violation, (2) effects of the violation upon unit components, systems or structures, and (3) corrective action taken to prevent recurrence;

c. The Safety Limit Violation Report shall be submitted to the Commission, NSOC and the Senior Vice President and General Manager, Nuclear Power Generation within 14 days of the violation; and

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Generic Letter No. 82-33;

c. Security Plan implementation;

d. Emergency Plan implementation;

e. PROCESS CONTROL PROGRAM implementation;

f. ODCP and ERMP implementation;

g. Quality Assurance Program for effluent and environmental monitoring; and

h. Fire Protection Program Implementation.

PROCEDURES AND PROGRAMS (Continued)

6.8.2 Each procedure of Specification 6.8.1 above, and changes thereto, and all proposed tests and experiments that affect nuclear safety shall be approved prior to implementation by the Plant Manager or by a technically qualified manager who reports directly to the Plant Manager, as previously designated by the Plant Manager. Prior to approval, the Plant Manager or designated manager shall ensure that all necessary reviews and cross-disciplinary reviews, if appropriate, have been completed. If deemed necessary, such cross-disciplinary reviews shall be performed by the appropriate designated plant review personnel. Administrative procedures, procedures implementing the Security Plan, Emergency Plan, and Process Control Program, the ODCP and ERMP, and changes thereto, shall be reviewed by the PSRC and approved by the Plant Manager prior to implementation. Each procedure of Specification 6.8.1 above shall be reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of Specification 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered;

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Operator license on the unit affected; and

c. The change is documented, reviewed and approved by the Plant Manager, or by a technically qualified manager who reports directly to the Plant Manager as previously designated by the Plant Manager, within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, and maintained:

a. Reactor Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include portions of the Recirculation Spray System, Safety Injection System, Chemical and Volume Control System, Residual Heat Removal System, RCS Sample System, and Liquid and Gaseous Radwaste Treatment Systems. The program shall include the following:

1) Preventive maintenance and periodic visual inspection requirements, and

2) Integrated leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1) Training of personnel,

2) Procedures for monitoring, and

3) Provisions for maintenance of sampling and analysis equipment.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 75 TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 74 TO FACILITY OPERATING LICENSE NO. DPR-82

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-275 AND 50-323

1.0 INTRODUCTION

By letter dated November 15, 1990, as supplemented by letters dated May 19, 1992 and October 8, 1992, Pacific Gas and Electric Company (PG&E or the licensee) requested amendments to Facility Operating Licenses DPR-80 and DPR-82 for Diablo Canyon Units 1 and 2, respectively. The amendment application is designated License Amendment Request LAR 90-11. The amendments change the combined Diablo Canyon technical specifications (TS) to: (1) revise the fire protection license condition (2.C.(5)) and the reporting license condition (2.G.), (2) relocate the fire protection TS and associated bases to plant procedures and the Updated Final Safety Analysis Report (UFSAR), (3) augment the administrative TS to define the fire protection program controls, and (4) revise the condensate storage tank TS and associated bases to clarify the requirement for water sources used by the auxiliary feedwater system.

The PG&E letter of May 19, 1992, provided additional information on the specific fire protection sections that will be included in the next FSAR update. This documents in the FSAR the major commitments for the basis of the Diablo Canyon fire protection program, in accordance with the recommendations of GL 86-10. The PG&E letter of October 8, 1992, proposed that the fire protection license conditions not be relocated in the licenses, contrary to what had been proposed in the PG&E letter of November 15, 1990. This is an administrative change. The information provided by the 1992 submittals is within the scope of the initial notice and did not affect the Commission's initial determination.

2.0 EVALUATION

Pacific Gas and Electric Company has proposed changes to the operating licenses and the technical specifications for Diablo Canyon Units 1 and 2. The changes proposed by the licensee are discussed below.

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A. Fire Protection Changes

The licensee has requested that the fire protection TS be relocated to plant procedures and the UFSAR in accordance with the guidance of the NRC's Generic Letter (GL) 86-10, "Implementation of Fire Protection Requirements," and GL 88-12, "Removal of Fire Protection Requirements from Technical Specifications." The staff's contractor, Science Applications International Corporation (SAIC), reviewed the licensee's submittals and found it acceptable. The SAIC Technical Evaluation Report (TER) documenting its review is enclosed. The NRC staff has reviewed the SAIC TER and agrees with its conclusions.

B. Clarification of Water Source Requirements for the Auxiliary Feedwater System (AFW)

In its November 15, 1990, letter, PG&E proposed that the TS for the condensate storage tank be modified to clarify the requirements for water sources for the auxiliary feedwater system (AFW). Specifically, the following changes to TS 3/4.7.1.3 were proposed:

(1) The title of TS 3/4.7.1.3 is changed from "Condensate Storage Tank" to "Auxiliary Feedwater Source."

(2) The Limiting Condition for Operation (LCO) is changed to require that a useable volume of water of 164,678 gallons be maintained in the condensate storage tank (CST) with an open flow path to the AFW pump suction. In addition, the LCO requires that a useable volume of water be maintained in the fire water storage tank (FWST) of 57,922 gallons when one unit is in operation and 115,844 gallons when two units are in operation, with a flow path capable of being aligned to the AFW pump suction.

The previous TS required that the CST have a contained volume of 178,000 gallons. The previous TS had no LCO on the FWST, but allowed the FWST to serve as a backup AFW supply source when the CST is inoperable, with a surveillance requirement that required demonstrating once per 12 hours that the FWST contained at least 270,000 gallons when it is the supply source for the AFW pumps.

(3) The Action Statements, Surveillance Requirements, and Bases for the revised TS 3/4.7.1.3 were changed to be consistent with the revised LCO.

The NRC staff has reviewed the proposed changes and finds them acceptable because they make no physical alteration to the water source or the AFW system, nor is there a change in the method by which the AFW system performs its functions. Further, the changes provide assurance that there will be a sufficient quantity of water supplied to the AFW pumps to remove heat from the reactor coolant system in the

event of a loss of normal feedwater to the steam generators. Specifically, the revised TS requires that 222,690 gallons (164,768 plus 57,922) be available when one unit is in operation, or 280,612 (164,768 plus 115,844) gallons of water be available when both units are in operation. These volumes are sufficient to allow plant cooldown under the accident conditions postulated by Generic Letter 81-21, "Natural Circulation Cooldown," which was committed to by PG&E by letter dated December 7, 1981. Based on the above, the NRC staff finds the proposed changes to TS 3/4.7.1.3 to be acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the California State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

These amendments involve changes with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding (56 FR 11784). Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Dated: January 13, 1993

Enclosure

SAIC-92/6698

TECHNICAL EVALUATION REPORT
FIRE PROTECTION LICENSE AMENDMENT
REQUEST DIABLO CANYON UNITS 1 AND 2
TAC Nos. M79423 and M79424

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An Employee-Owned Company
July 23, 1992

Prepared for:

U.S. Nuclear Regulatory Commission
Washington, DC 20555

Contract NRC-03-87-029
Task Order No. 97

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1.0 INTRODUCTION

This Technical Evaluation Report (TER) documents in independent review of License Amendment Request 90-11 submitted for Diablo Canyon Units 1 and 2 by Pacific Gas and Electric Company, the licensee. The request was submitted to the NRC by letter dated November 15, 1990. A plant visit was made on February 6 and 7, 1992 following a preliminary review of the submittal. As a result of the plant visit and a subsequent conference call with the licensee on February 27, 1992 it was determined that additional information was needed from the licensee. The licensee provided this information to the NRC by letter dated May 19, 1992.

2.0 DISCUSSION

The licensee has requested an amendment to the Unit 1 and Unit 2 Operating License which would revise the fire protection license condition and relocate fire protection technical specifications to plant operating procedures. This amendment was requested in accordance with guidance provided in NRC Generic Letter 86-10.

The proposed changes to the Technical specifications are as follows:

1. The requirements of TS 3/4.3.3.8 (Fire Detection Instrumentation), 3/4.7.9.1 (Fire Suppression Water System), 3/4.7.9.2 (Spray and/or Sprinkler systems), 3/4.7.9.3 (CO2 system), 3/4.7.10 (Fire Barrier Penetrations) would be relocated to plant administrative procedure controls and to the FSAR Update.

2. The requirements of TS 6.2.2.e (Plant Staff Fire Brigade Composition) are currently in the FSAR Update and also would be relocated to plant procedures.

3. TS 6.5.2.6 and 6.5.2.7 (Plant Safety Review Committee (PSRC) Responsibilities) would be revised to require PSRC review of the plant Fire Protection Program and its revisions and recommend approval to the Plant Manager.

4. TS 6.8.1 (Written Procedure Requirements) would be revised to include the Fire Protection Program Implementation.

The proposed license condition for Unit 1 reads as follows:

PG&E shall implement and maintain in effect all provisions of the approved fire protection plan as discussed in its Final Safety Analysis Report Update, in PG&E's December 6, 1984, Appendix R Report, and in the NRC staff's Fire Protection Evaluation in Supplement 8, 9, 13, 23, and 27 to the Diablo Canyon Safety Evaluation Report, subject to provision b below.

Provision b:

PG&E may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

The proposed Unit 2 license condition reads exactly the same as Unit 1 except Supplement 31 is added to the list of Safety Evaluation Reports.

3.0 EVALUATION

The license amendment request for Diablo Canyon was reviewed against guidance provided in NRC Generic Letters 86-10 and 88-12. A site visit was made on February 5-6, 1992 to review current and proposed plant implementing procedures. In addition, an assessment was made against the guidance in Generic Letter 81-12 which establishes the need to provide

technical specification requirements for alternate shutdown equipment which was not previously contained in plant technical specifications.

The plant proposes to relocate the current fire protection technical specification requirements identified in Section 2 of this TER, to plant Equipment Control Guidelines (ECGs). This type of procedure was established by the licensee to address those operating requirements that did not exist in plant technical specifications but existed in other licensing documents such as the FSAR. The proposed fire protection ECGs were reviewed to ensure that all aspects of the fire protection program which were being deleted from technical specifications, would be addressed in plant procedures. Although the fire protection ECGs were not approved at the time of review, the draft procedures presented, did cover all technical specification items.

The FSAR update was reviewed to verify that all aspects of the fire protection program had been incorporated as defined in Generic Letter 86-10. The licensee had modified the FSAR Update in Revision 3 to include most aspects of the fire protection program. This Revision was submitted by letter dated September 16, 1987 and contained the certification per 10 CFR 50.71(e) as required by Generic Letter 86-10. This revision did not contain however, the operating requirements and surveillance requirements for fire protection equipment which is currently contained in plant technical specifications. The licensee stated that they intend to include this information in the next FSAR Update submittal which is scheduled for September 1992. This is not in accordance with Generic Letter 86-10 which states that the licensee must first include major commitments which form the basis of the fire protection program in the FSAR directly or by reference prior to removing technical specification requirements. By letter dated May 19, 1992 the licensee submitted a revision to the FSAR which included the operability and surveillance requirements proposed to be removed from technical specifications. This FSAR revision was reviewed and found to provide the equivalent information that currently exists in the plant's technical specifications and is therefore acceptable.

This review determined that the licensee's proposal does properly modify the existing technical specifications to require the Plant Safety Review Committee to review changes to the fire protection program and that the technical specifications require that written procedures exist to govern the fire protection program implementation.

The proposed license conditions were also discussed with the licensee. Although they were determined to be consistent with Generic Letter 86-10, a concern arose regarding the reference to Supplement 31 to the Safety Evaluation Report, only in the Unit 2 condition. Upon initial review, it was not clear whether this supplement had any impact on the Unit 1 fire protection program. Because of this, the licensee agreed to review this supplement to determine if it should be added to the Unit 1 license condition. As a result of this review, the licensee informed the NRC by telephone call on February 27, 1992 that the proposed license conditions did reference the appropriate safety evaluations and therefore did not need to be changed.

The existence of technical specifications for alternate shutdown equipment was discussed with the licensee. The licensee stated that as a result of findings in NRC Region V Inspection Report 91-01, a review of safe shutdown equipment was being conducted to determine if operability procedures were necessary. The licensee stated they did not intend to include any new operability requirements in technical specifications, however, they did intend to modify plant ECGs as appropriate. The licensee stated in their May 19, 1992 letter that ECGs for alternate shutdown equipment not covered by existing technical specifications will be implemented by May 5, 1993. The lack of technical specifications for alternate shutdown equipment does not conform to NRC guidance in Generic Letter 81-12.