12 JUN 1987

Docket No. 50-275
and 50-323

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Mr. J. D. Shiffer, Vice President
Nuclear Power Generation
c/o Nuclear Power Generation, Licensing
Pacific Gas and Electric Company
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Dear Mr. Shiffer:

SUBJECT: ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES DPR-80 AND DPR-82 - DIABLO CANYON NUCLEAR POWER PLANT (TAC NOS. 65011 AND 65012)

The Commission has issued the enclosed Amendment No. 14 to Facility Operating License No. DPR-80 and Amendment No. 13to Facility Operating License No. DPR-82 for the Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated March 25, 1987, as supplemented May 26, 1987.

These amendments revise the Diablo Canyon combined Technical Specifications for Units 1 and 2 to accommodate Cycle 2 and later operation of Unit 2, and Cycle 3 and later operation of Unit 1.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular bi-weekly Federal Register notice.

Sincerely,

Original signed by

Charles M. Trammell, Project Manager
Project Directorate V
Division of Reactor Projects - III/IV/V
& Special Projects

Enclosures:
1. Amendment No. 14 to DPR-80
2. Amendment No. 13 to DPR-82
3. Safety Evaluation

cc w/enclosures:
See next page

DRSP/PD5	DRSP/PD5	OGC	DRSP/D:PD5
JLee	CTrammell	BHVogler	GKnighton
6/10/87	6/10/87	6/10/87	6/12/87

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

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-275

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 14
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 March 25, 1987, as supplemented May 26, 1987 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and 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 rules and 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) of Facility Operating License No. DPR-80 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 14 , 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.

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

FOR THE NUCLEAR REGULATORY COMMISSION

George W. Knighton

George W. Knighton, Director
Project Directorate V
Division of Reactor Projects - III/IV/V
& Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 12, 1987

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

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON NUCLEAR POWER PLANT, UNIT 2

DOCKET NO. 50-323

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 13
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 March 25, 1987, as supplemented May 26, 1987 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and 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 rules and 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) of Facility Operating License No. DPR-82 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 13 , 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.

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

FOR THE NUCLEAR REGULATORY COMMISSION

George W. Knighton

George W. Knighton, Director
Project Directorate V
Division of Reactor Projects - III/IV/V
& Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 12, 1987

ATTACHMENT TO LICENSE AMENDMENT NOS.14 AND 13

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

DOCKET NOS. 50-275 AND 50-323

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
2-2a	2-2a
2-7	2-7
2-8	2-8
B 2-1	B 2-1
3/4 1-12	3/4 1-12
3/4 1-13	3/4 1-13
3/4 2-6	3/4 2-6
3/4 2-9	3/4 2-9
3/4 2-11	3/4 2-11
3/4 5-1	3/4 5-1
3/4 5-11	3/4 5-11
B3/4 1-2	B3/4 1-2
B3/4 1-3	B3/4 1-3
B3/4 5-3	B3/4 5-3
B3/4 6-3	B3/4 6-3

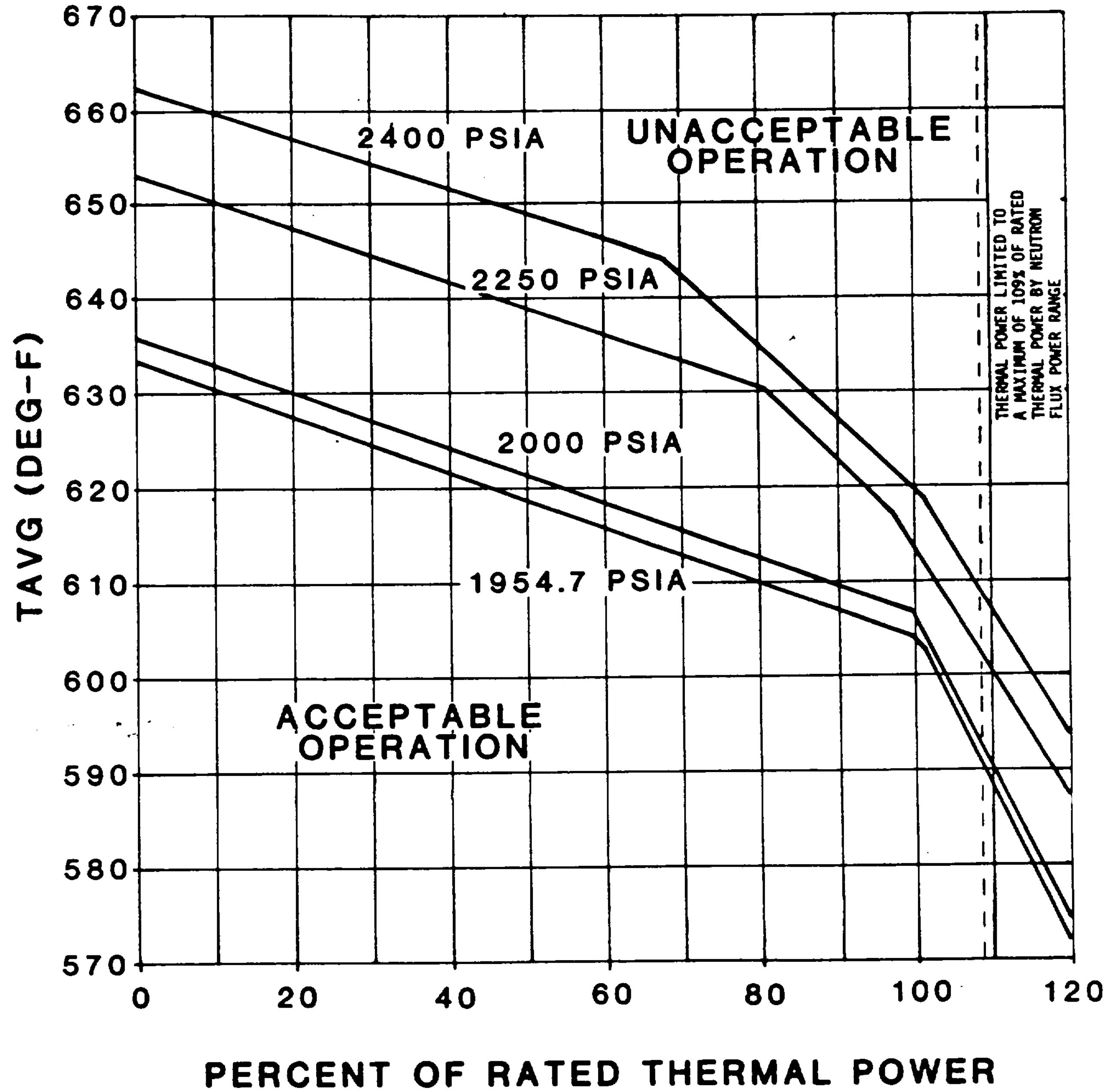


FIGURE 2.1-1b
REACTOR CORE SAFETY LIMIT (UNIT 2)

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \leq \Delta T_o \left[K_1 - K_2 \frac{1+\tau_1 S}{1+\tau_2 S} (T-T') + K_3(P-P') - f_1(\Delta I)\right]$$

Where: ΔT_o = Indicated ΔT at RATED THERMAL POWER;

T = Average temperature, °F;

T' = $\leq$ 576.6°F for Unit 1 and $\leq$ 577.6°F for Unit 2 Reference T_{avg} at RATED THERMAL POWER;

P = Pressurizer pressure, psig;

P' = 2235 psig (indicated RCS nominal operating pressure);

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation;

τ_1 & τ_2 = Time constants utilized in the lead-lag controller for T_{avg}, τ_1 = 30 s, τ_2 = 4 s;

S = Laplace transform operator, s^{-1};

K_1 = 1.174 (Unit 1 Cycle 2);

K_1 = 1.166 (Unit 1 Cycle 3 and after, Unit 2);

K_2 = 0.01358/°F (Unit 1 Cycle 2);

K_2 = 0.01149/°F (Unit 1 Cycle 3 and after, Unit 2);

K_3 = 0.000685/psig (Unit 1 Cycle 2);

K_3 = 0.000502/psig (Unit 1 Cycle 3 and after, Unit 2);

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

TABLE NOTATIONS (Continued)

NOTE 1 (Continued)

and f_1 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for q_t - q_b between - 32% and + 10% (Unit 1 Cycle 2) and -32% and +9% (Unit 1 Cycle 3 and after, Unit 2), f_1 (ΔI) = 0 (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and q_t + q_b is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) for each percent that the magnitude of (q_t - q_b) exceeds - 32%, the ΔT Trip Setpoint shall be automatically reduced by 2.11% (Unit 1 Cycle 2) and 2.02% (Unit 1 Cycle 3 and after, Unit 2) of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of (q_t - q_b) exceeds + 10%, (Unit 1 Cycle 2) and 9% (Unit 1 Cycle 3 and after, Unit 2) the ΔT Trip Setpoint shall be automatically reduced by 1.45% (Unit 1 Cycle 2) and 1.454% (Unit 1 Cycle 3 and after, Unit 2) of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 4%.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and reactor coolant temperature and pressure have been related to DNB through the R-Grid correlation. The R-Grid DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux and is indicative of the margin to DNB.

The minimum value of the DNBR during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30. This value corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than 1.30, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

The curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$ of 1.55 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.55\ [1+ 0.3\ (1-P)]$$

where P is the fraction of RATED THERMAL POWER

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trip will reduce the Setpoints to provide protection consistent with core Safety Limits.

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A Boric Acid Storage System and at least one associated heat tracing channel with:

   1) A minimum contained borated water volume of 835 gallons,

   2) A boron concentration between 20,000 and 22,500 ppm, and

   3) A minimum solution temperature of 145°F.

b. The Refueling Water Storage Tank (RWST) with:

   1) A minimum contained borated water volume of 50,000 gallons,

   2) A minimum boron concentration of 2000 ppm (Unit 1 Cycle 2), A minimum boron concentration of 2300 ppm (Unit 1 Cycle 3 and after, Unit 2), and

   3) A minimum solution temperature of 35°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

   1) Verifying the boron concentration of the water,

   2) Verifying the contained borated water volume, and

   3) Verifying the boric acid storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water and the outside ambient air temperature is less than 35°F.

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 Each of the following borated water source(s) shall be OPERABLE:

a. A Boric Acid Storage System and at least one associated heat tracing channel with:

   1) A minimum contained borated water volume of 5106 gallons,

   2) A boron concentration between 20,000 and 22,500 ppm, and

   3) A minimum solution temperature of 145°F.

b. The Refueling Water Storage Tank (RWST) with:

   1) A contained borated water volume of greater than or equal to 400,000 gallons,

   2) A boron concentration between 2000 and 2200 ppm (Unit 1 Cycle 2), A boron concentration between 2300 and 2500 ppm, (Unit 1 Cycle 3 and after, Unit 2), and

   3) A minimum solution temperature of 35°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the Boric Acid Storage System inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F; restore the Boric Acid Storage System to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

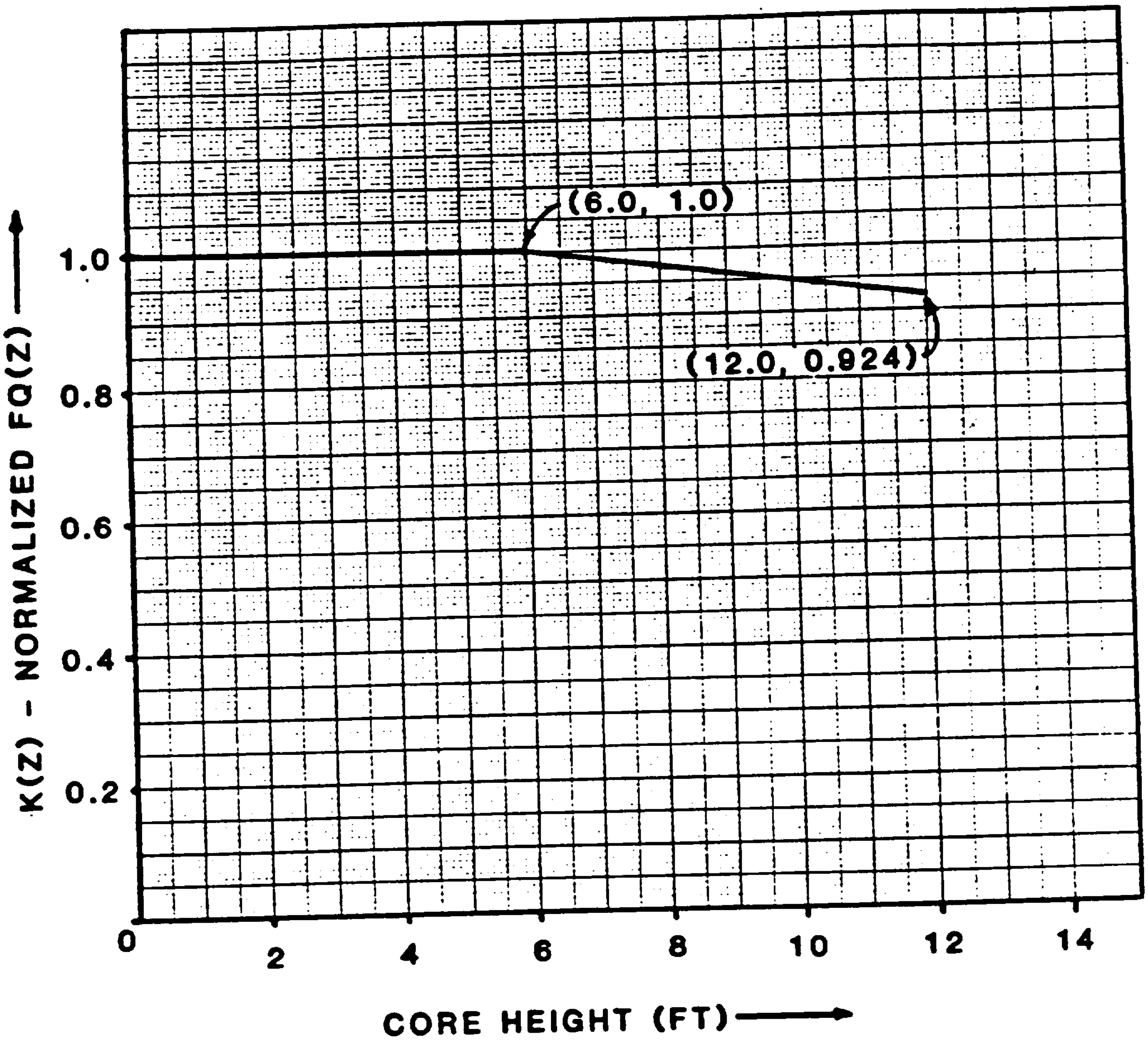


FIGURE 3.2-2
K(Z) - NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT

3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System (RCS) total flow rate and R shall be maintained within the region of allowable operation shown on Figure 3.2-3a for Unit 1 and Figure 3.2-3b for Unit 2 for four loop operation.

Where:

a. $R = \dfrac{F^N_{\Delta H}}{1.49\,[1.0 + 0.3\,(1.0 - P)]}$

b. $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

c. $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^N_{\Delta H}$ shall be used to calculate R since Figure 3.2-3a for Unit 1 and Figure 3.2-3b for Unit 2 include measurement uncertainties of 3.5% for flow and 4% for incore measurement of $F^N_{\Delta H}$.

APPLICABILITY: MODE 1.

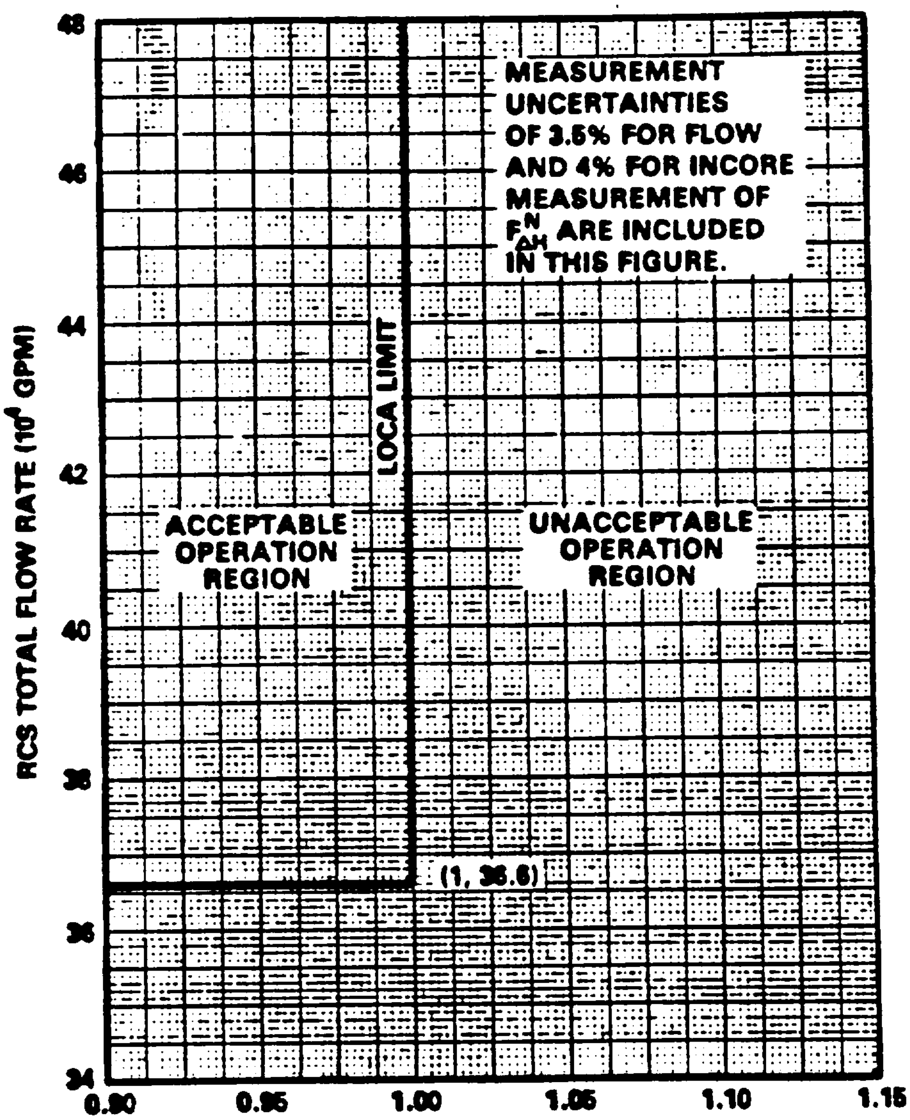
ACTION:

With the combination of RCS total flow rate and R outside the region of acceptable operation shown on Figure 3.2-3a for Unit 1 and Figure 3.2-3b for Unit 2:

a. Within 2 hours either:

1. Restore the combination of RCS total flow rate and R to within the above limits, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.



$$R = \frac{F^N_{\Delta H}}{1.49[1.0 + 0.3(1.0-P)]}$$

FIGURE 3.2-3b

RCS TOTAL FLOWRATE VERSUS R (UNIT 2)

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System accumulator shall be OPERABLE with:

a. The isolation valve open and power removed,

b. A contained borated water volume of between 836 and 864 cubic feet of borated water,

c. A boron concentration of between 1900 and 2200 ppm (Unit 1 Cycle 2), A boron concentration of between 2200 and 2500 ppm (Unit 1 Cycle 3 and after, Unit 2), and

d. A nitrogen cover-pressure of between 595.5 and 647.5 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying the contained borated water volume and nitrogen cover-pressure in the tanks, and

2) Verifying that each accumulator isolation valve is open.

*Pressurizer pressure above 1000 psig.

3/4.5.5 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The Refueling Water Storage Tank (RWST) shall be OPERABLE with:

a. A minimum contained borated water volume of 400,000 gallons,

b. A boron concentration of between 2000 and 2200 ppm (Unit 1 Cycle 2), A boron concentration of between 2300 and 2500 ppm (Unit 1 Cycle 3 and after, Unit 2), and

c. A minimum solution temperature of 35°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the contained borated water volume in the tank, and

2) Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature when the outside ambient air temperature is less than 35°F.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 541°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the protective instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, (5) associated heat tracing systems, and (6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.6% Δk/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 5106 gallons of 20,000 ppm borated water from the boric acid storage tanks or 75,000 gallons of 2000 ppm (Unit 1 Cycle 2) and 2300 ppm (Unit 1 Cycle 3 and after, Unit 2) borated water from the refueling water storage tank.

With the RCS temperature below 200°F, one Boron Injection System is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% Δk/k after xenon decay and cooldown from 200°F to 140°F. This condition requires either 835 gallons of 20,000 ppm borated water from the boric acid storage tanks or 9690 gallons of 2000 ppm (Unit 1 Cycle 2) and 2300 ppm (Unit 1 Cycle 3 and after, Unit 2) borated water from the refueling water storage tank.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.0 and 9.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

BORATION SYSTEMS (Continued)

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

The OPERABILITY of one Boron Injection System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all centrifugal charging pumps except the required OPERABLE pump to be inoperable below 323°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits. Group demand position can be determined from: (1) the group step counters, or (2) the plant computer, or (3) for control rods, the P to A converter at the rod control cabinet.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Continued operation of the Rod Control system is allowed with multiple immovable rods, that are still trippable and within alignment, for periods up to 72 hours to allow maintenance and/or testing of the Rod Control system (additional information is included in Attachment C of the Westinghouse letter to the NRC on Movable Assemblies, December 21, 1984.) Misalignment of a rod requires measurement of peaking factors and a restriction in THERMAL POWER. These restrictions provide assurance of fuel rod integrity during continued operation. In addition, those accident analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 541°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a Reactor trip at operating conditions.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the Refueling Water Storage Tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of either a LOCA or a steamline break. The limits on RWST minimum volume and boron concentration ensure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core; (2) the reactor will remain subcritical in the cold condition (68 to 212 degrees-F) following a small break LOCA assuming complete mixing of the RWST, RCS, spray additive tank, containment spray system piping and ECCS water volumes with all control rods inserted except the most reactive control rod assembly (ARI-1); (3) the reactor will remain subcritical in the cold condition following a large break LOCA (break flow area greater than 3 ft^2) assuming complete mixing of the RWST, RCS, ECCS water and other sources of water that may eventually reside in the sump post-LOCA with all control rods assumed to be out (ARO); and (4) long term subcriticality following a steamline break assuming ARI-1 and preclude fuel failure.

The maximum allowable value for the RWST boron concentration forms the basis for determining the time (post-LOCA) at which operator action is required to switch over the ECCS to hot leg recirculation in order to avoid precipitation of the soluble boron.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the Containment Spray System ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses.

The Containment Spray System and the Containment Cooling System are redundant to each other in providing post accident cooling of the containment atmosphere. However, the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable Spray System to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

3/4.6.2.2 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the Spray Additive System ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH minimum volume and concentration ensure a pH value of between 8.0 and 9.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the safety analyses.

3/4.6.2.3 CONTAINMENT COOLING SYSTEM

The OPERABILITY of the containment fan cooler units ensures that: (1) the containment air temperature will be maintained within limits during normal operation, (2) adequate heat removal capacity is available when operated in conjunction with the containment spray systems during post LOCA conditions, and (3) adequate mixing of the containment atmosphere following a LOCA to prevent localized accumulations of hydrogen from exceeding the flammable limit.

The Containment Cooling System and the Containment Spray System are redundant to each other in providing post accident cooling of the containment atmosphere. As a result of this redundancy in cooling capability, the allowable out of service time requirements for the Containment Cooling System have been appropriately adjusted. However, the allowable out of service time requirements for the Containment Spray System have been maintained consistent with that assigned other inoperable ESF equipment since the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 14 TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 13 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 March 25, 1987, as supplemented May 26, 1987, Pacific Gas and Electric Company (PG&E or the licensee) requested amendments to the Technical Specifications appended to Facility Operating License Nos. DPR-80 and DPR-82 for the Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2. The proposed amendments would revise the Diablo Canyon combined Technical Specifications for Units 1 and 2 to accommodate Cycle 2 and later operation of Unit 2, and Cycle 3 and later operation of Unit 1.

The requested amendment is in the form of proposed changes to the Technical Specifications as follows:

1. Increase the $F^N_{\Delta H}$ partial power multiplier.

2. Increase the refueling water storage tank and accumulator boron concentration.

3. Relax the third line segment of the K(z) figure.

2.0 EVALUATION

1. $F^N_{\Delta H}$ Multiplier

The Technical Specifications for Westinghouse reactors historically have allowed for an increase in $F^N_{\Delta H}$ with decreasing power level to compensate for power distribution changes with control rod insertion and decreasing reactivity feedback. This has been done with a 0.2 part power multiplier on $F^N_{\Delta H}$. The 0.2 multiplier on occasion has been restrictive at low power, and in recent years a 0.3 multiplier has been approved for a number of operating plants.

In general the restriction that the average enthalpy at the vessel exit be less than the enthalpy of saturated liquid is more limiting than DNB considerations, so the increase in the allowable $F^N_{\Delta H}$ at reduced power levels does not result in large changes to the reactor safety limits defined in Technical Specification Section 2. The exit enthalpy restriction is not impacted by the radial peaking factor. The licensee presented an analysis supporting this.

The analysis for Diablo Canyon Units 1 and 2 showed that slight modification of the reactor core safety limit curve, TS Figure 2.1-1b, was required as a result of the change in the $F^N_{\Delta H}$ multiplier. The licensee proposed a revised TS Figure 2.1-1(b) reflecting these changes. (The figure provided in the licensee's March 25, 1987 submittal contains an error which has been corrected in a submittal dated May 26, 1987.)

As a consequence of the change to the safety limits, the licensee therefore also proposed changes to the equation constants for the over-temperature ΔT trip setpoints. Reanalysis of affected non-LOCA accident events with the revised setpoints was performed and the conclusions for the non-LOCA accidents were found to remain valid as presented in the revised Diablo Canyon FSAR.

Because the proposed changes (with the corrected Fig 2.1-1b) and the accident evaluation are the result of analyses using the methods described in the approved report WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology", we find them acceptable.

2. Boron Concentration

The proposed license amendment request increases the Refueling Water Storage Tank (RWST) and accumulator boron concentration. These changes are required to have enough boron concentration to maintain the core subcritical following a large break LOCA. For Cycle 1 on both units and Cycle 2 on Unit 1, the present minimum boron concentrations of 2000 ppm for the RWST and 1900 ppm for the accumulators are capable of maintaining the core subcritical. For Unit 2, Cycle 2 the reload safety evaluation determined that, with the excess reactivity provided for a longer cycle, if the burnup of Cycle 1 was less than 15,250 MWD/MTU, higher boron concentration in the RWST and accumulator were needed to satisfy the post LOCA subcriticality requirement. Since the Cycle 1 burnup was less than the stated value, the licensee proposes to increase the boron concentrations for the RWST from between 2000 ppm and 2200 ppm to between 2300 ppm and 2500 ppm. The accumulator boron concentration is proposed to be increased from between 1900 ppm and 2200 ppm to between 2200 ppm and 2500 ppm. We examined the proposed changes to Technical Specifications 3.1.2.5, 3.1.2.6, 3.5.1 and Bases 3/4.1.2, 3/4.5.5 and 3/4.6.2.2 and find they correctly implemented these changes. The proposed increased boron concentrations will also support future core designs with longer fuel cycles and higher burnup.

The licensee's submittal presented the results of analyses of the proposed increases in RWST and accumulator boron concentrations on the following areas of the Diablo Canyon Units 1 and 2 design:

1. Non-LOCA Transient Analysis

2. LOCA Analysis
    a. Small Breaks
    b. Large Breaks
    c. Long-Term Core Cooling
    d. Boron Precipitation

3. LOCA Related Design Considerations
    a. Radiological Consequences
    b. Hydrogen Production
    c. Equipment Qualifications.

The non-LOCA safety analyses in which boron from the RWST or accumulators is involved are uncontrolled boron dilution, accidental depressurization of the main steam system, spurious operation of the safety injection system at power, minor secondary system pipe breaks, rupture of a main steam line, and rupture of a control rod drive mechanism housing. The analyses show no negative effect of the proposed boron concentration increases on the non-LOCA transients.

The proposed boron concentration changes do not have any effect on the small or large break LOCA analyses except for the long term cooling post LOCA shutdown analyses. For these analyses, the increased concentrations are required for Unit 2 Cycle 2 as discussed above. For future cycles of both units, confirmation that the proposed increases in boron concentration will provide enough margin to keep the core subcritical for long term cooling requirements is required. This will be accomplished through the normal cycle-specific reload safety evaluation process.

The analyses of boron precipitation and other LOCA related design considerations all showed acceptable results. We therefore conclude the proposed RWST and accumulator boron concentration increases are acceptable.

3. K(z) Third Line Segment

The K(z) curve provides the normalized heat flux hot channel factor as a function of core height. Near the top of the core, from 10.8 to 12.0 ft., the present K(z) curve decreases rapidly from 0.94 to 0.43. A LOCA reanalysis was performed by Westinghouse using a K(z) curve which continues the second line segment as a straight line. This eliminates the third line segment and causes 12 foot intercept of the K(z) curve to be 0.924. The third line segment of the K(z) curve is determined by the small break LOCA analysis. Westinghouse reanalyzed the small break LOCA using the approved NOTRUMP code (WCAP-10054-P-A). The three, four and six inch diameter break sizes were analyzed for Unit 2, as well as the limiting of these three break sizes for Unit 1. The worst case was the four inch diameter break size. This resulted in a peak cladding temperature of 1288° F for Unit 2 and 1244° F for Unit 1. These results are well below the peak cladding temperature limit of 2200° F specified in 10 CFR 50.46. The proposed modification of the K(z) curve is therefore acceptable for both units.

As discussed above, the Technical Specification changes proposed by PG&E in its letter of March 25, 1987, and as modified in a letter dated May 26, 1987 are acceptable. As requested in the March 25, 1987 letter, the changes should become effective immediately upon issuance of this license amendment for Unit 2 and by the completion of the second refueling outage for Unit 1.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve changes in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have 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. Accordingly, these amendments meets 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.

4.0 CONCLUSION

We have 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, and (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: June 12, 1987