October 21, 1986

Docket Nos. 50-275
and 50-323

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Mr. J. D. Shiffer, Vice President
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c/o Nuclear Power Generation, Licensing
Pacific Gas and Electric Company
77 Beale Street, Room 1451
San Francisco, California 94106

Dear Mr. Shiffer:

The Commission has issued the enclosed Amendment No. 10 to Facility Operating License No. DPR-80 and Amendment No. 8 to 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 June 10, 1986, as supplemented August 19, 1986.

These amendments revise the Technical Specifications to (1) redefine the moderator temperature coefficient limits, (2) revise the F^N-delta-H partial power multiplier, and (3) delete the design feature description of the total weight of uranium in a fuel rod. These changes will facilitate the operation of Unit 1 Cycle 2. Changes (1) and (3) apply equally to Units 1 and 2. Change (2) applies only to 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,

Hans Schierling, Senior Project Manager
Project Directorate #3
Division of PWR Licensing-A

Enclosures:
1. Amendment No. 10 to DPR-80
2. Amendment No. 8 to DPR-82
3. Safety Evaluation

cc: w/enclosures
See next page

* SEE PREVIOUS CONCURRENCE

PD#3	PD#3	OGC	PD#3
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10/07/86	10/ 7/86	10/ /86	10/21/86

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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. 10
License No. DPR-80

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

   A. The application for amendment dated June 10, 1986, as supplemented August 19, 1986, (LAR 86-06), by Pacific Gas & Electric Company (the licensee) 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 1;

   B. The facility will operate in conformity with the application, as amended, 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 a change to the combined Technical Specifications for Units 1 and 2 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, as revised through Amendment No. 10 , and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. PG&E shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Director
PWR Project Directorate #3
Division of PWR Licensing-A, NRR

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 21, 1986

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. 8
License No. DPR-82

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

   A. The application for amendment dated June 10, 1986, as supplemented August 19, 1986, (LAR 86-06), by Pacific Gas & Electric Company (the licensee) 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 1;

   B. The facility will operate in conformity with the application, as amended, 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 a change to the combined Technical Specifications for Units 1 and 2 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, as revised through Amendment No. 8 , and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. PG&E shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Director
PWR Project Directorate #3
Division of PWR Licensing-A, NRR

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 21, 1986

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

ATTACHMENT TO LICENSE AMENDMENT NOS. 10 AND 8

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

DOCKET NOS. 50-275 AND 50-323

Revise the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by Amendment Number and contain vertical lines indiciating the area of change.

Remove Pages	Insert Pages
2-1	2-1
2-2	2-2
---	2-2a
B2-1	B2-1
3/4 1-4	3/4 1-4
3/4 2-9	3/4 2-9
3/4 2-10	3/4 2-10
B3/4 1-1	B3/4 1-1
5-5	5-5

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1a for Unit 1 and Figure 2.1-1b for Unit 2.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.

MODES 3, 4 and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.

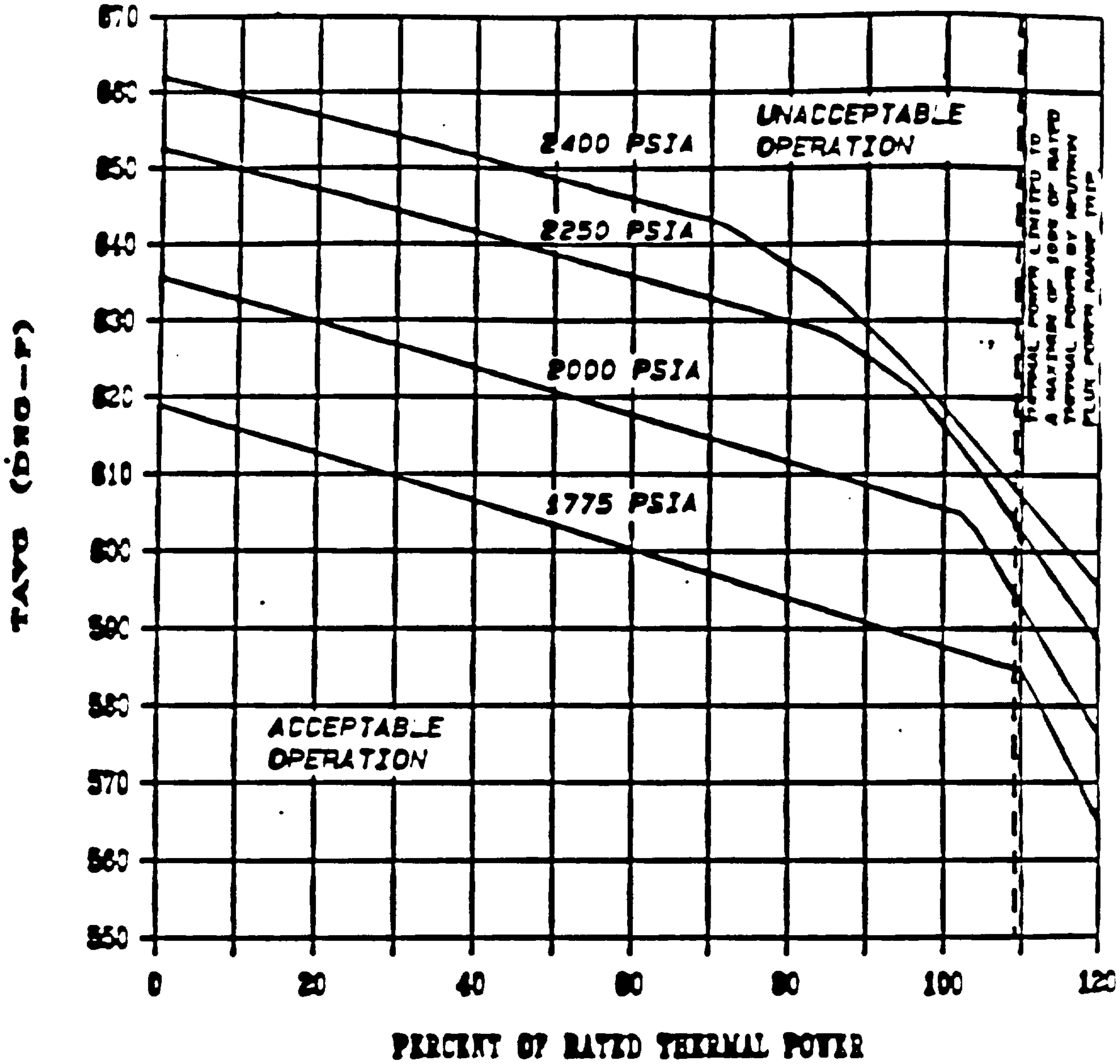

FIGURE 2.1-1a
REACTOR CORE SAFETY LIMIT (UNIT 1)

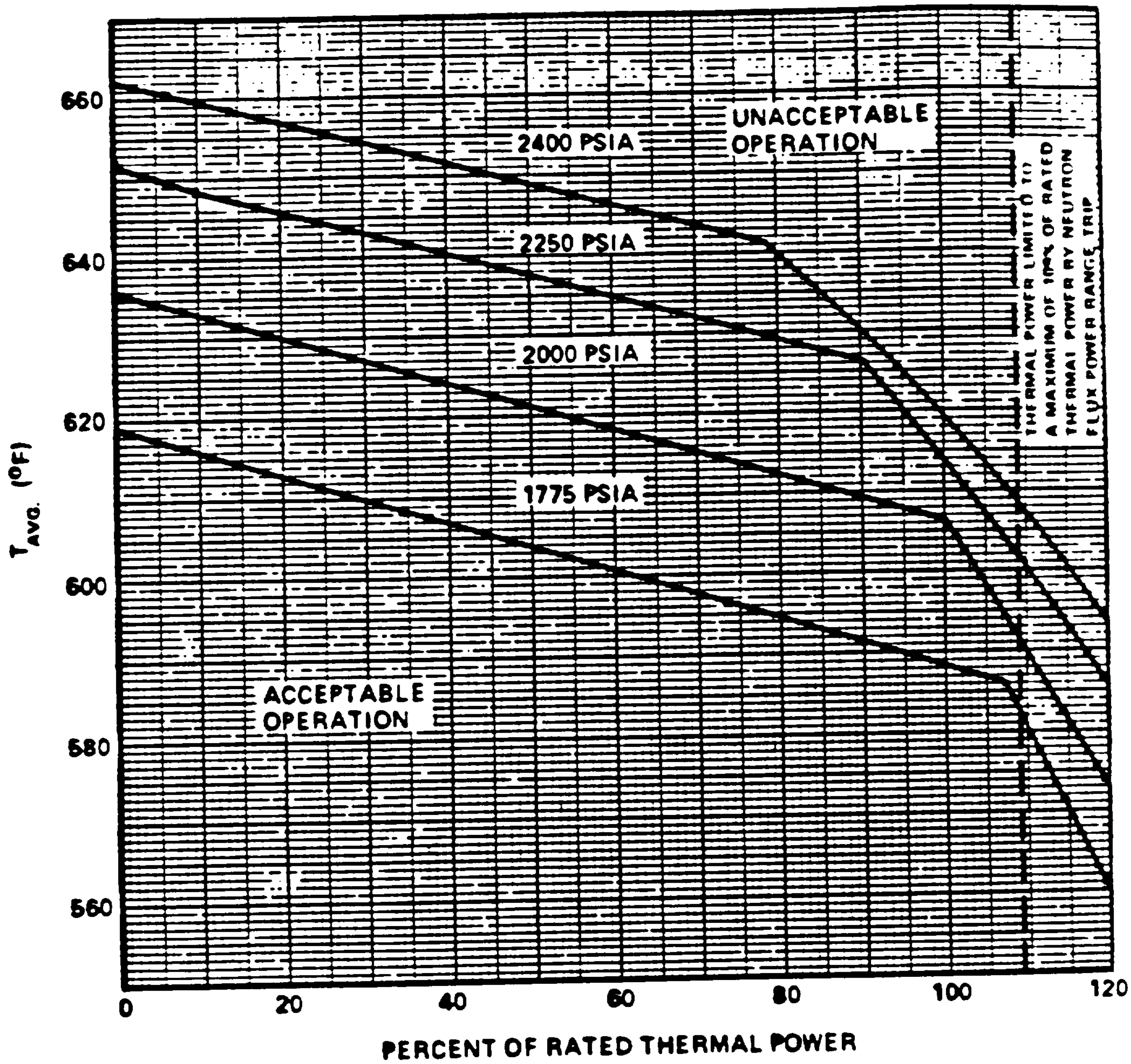

FIGURE 2.1-1b
REACTOR CORE SAFETY LIMIT (UNIT 2)

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:

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

$$F^N_{\Delta H} = 1.55\ [1+ 0.2\ (1-P)]\ \text{(Unit 2)}$$

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.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than $+5 \times 10^{-5}$ $\Delta k/k/°F$ for 0% to 70% RATED THERMAL POWER, and for > 70% to 100% RATED THERMAL POWER the MTC decreases linearly to 0 $\Delta K/\bar{K}/°F$ for the all rods withdrawn condition, beginning of cycle life (BOL); or

b. Less negative than -3.9×10^{-4} $\Delta k/k/°F$ for all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

APPLICABILITY: Specification 3.1.1.3a. - MODES 1 and 2* only#.
Specification 3.1.1.3b. - MODES 1, 2, and 3 only#.

ACTION:

a. With the MTC more positive than the limit of Specification 3.1.1.3a. above, operation in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than the limit of Specification 3.1.1.3a within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition; and

3. A Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the limit of Specification 3.1.1.3b. above, be in HOT SHUTDOWN within 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

---

*With K_{eff} greater than or equal to 1.

#See Special Test Exceptions Specification 3.10.3.

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. $$R = \frac{F^N_{\Delta H}}{1.49\ [1.0 + 0.3\ (1.0 - P)]}, \text{ (Unit 1)},$$

$$R = \frac{F^N_{\Delta H}}{1.49\ [1.0 + 0.2\ (1.0 - P)]}, \text{ (Unit 2)}$$

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

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.

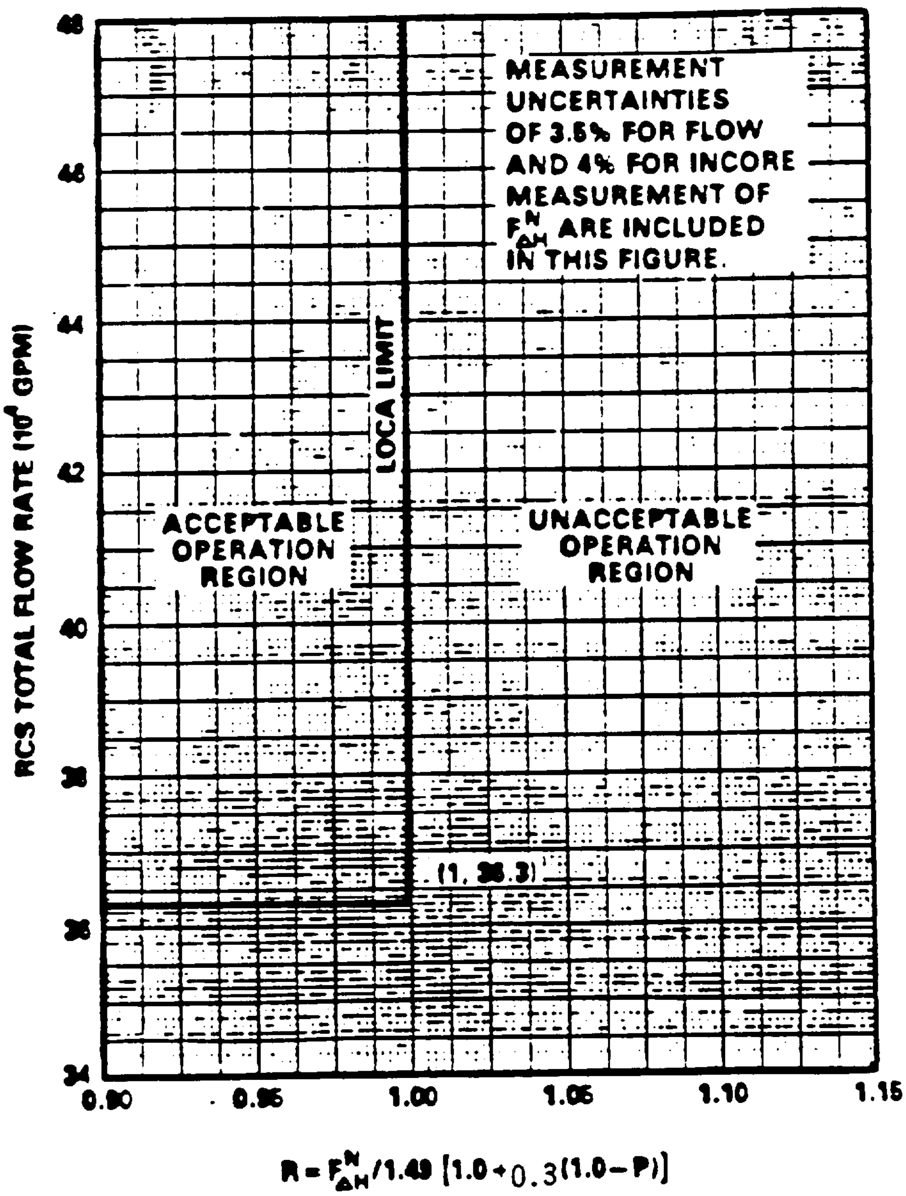

FIGURE 3.2-3a

RCS TOTAL FLOWRATE VERSUS R (UNIT 1)

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that: (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.6% $\Delta k/k$ is initially required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% $\Delta k/k$ shutdown margin provides adequate protection.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting conditions assumed in the FSAR accident and transient analysis.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC) was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition, and a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting MTC value -3.9×10^{-4} $\Delta k/k/°F$. The MTC value of -3.0×10^{-4} $\Delta k/k/°F$ represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting MTC value of -3.9×10^{-4} $\Delta k/k/°F$.

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of each fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. In addition, verification during startup testing at beginning of life hot zero power for each cycle validates that the MTC parameters are within the limits specified for all other power levels.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 Containment is designed and shall be maintained for a maximum internal pressure of 47 psig and a temperature of 271°F, coincident with a Double Design Earthquake.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 193 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 144 inches. The initial core loading shall have a maximum enrichment of 3.15 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 4.5 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 53 full length and no part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80% silver, 15% indium, and 5% cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 12,811 ± 100 cubic feet at a nominal T_{avg} of 576°F for Unit 1 and 12,903 ± 100 cubic feet at a nominal T_{avg} of 577°F for Unit 2.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 10 TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 8 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

INTRODUCTION

By letter dated June 10, 1986 Pacific Gas and Electric Company (the licensee) made application to amend Facility Operating Licenses DPR-80 and DPR-82 for Diablo Canyon Units 1 and 2 to reflect the Cycle 2 refueling and related Technical Specification changes. The Cycle 2 reload core will utilize 68 new Westinghouse fuel assemblies and 448 fresh burnable absorber rods. The new fuel assemblies are of the same mechanical, nuclear, and thermal hydraulic design as Standard Fuel Assemblies except for some minor mechanical design changes. The licensee references the approved Westinghouse reload methodology outlined in "Westinghouse Reload Safety Evaluation Methodology" (WCAP-9272 P-A, July 1985) for the Cycle 2 core analyses. We consider the Cycle 2 reload core acceptable.

The proposed Technical Specification changes include (1) redefining the moderator temperature coefficient limits, (2) revising the F^N-delta-H partial power multiplier, and (3) deleting the design feature description of the total weight of uranium in a fuel rod. Changes (1) and (3) apply to Units 1 and 2. Change (2) applies to Unit 1 only.

DISCUSSION AND EVALUATION

Moderate Temperature Coefficient (MTC)

The present Technical Specifications require the MTC to be zero or negative at all times while the reactor is critical. The licensee proposes to change Technical Specification Section 3/4.1.1.3 to allow a maximum positive MTC of 5 x 10-5Δk/k/°F below 70% power, with the maximum positive MTC value decreasing linearly to 0Δk/k/°F between 70% and 100% power during beginning of life (BOL) operation. The licensee also proposes to revise Bases 3/4.1.1.3 to clarify the BOL Surveillance Requirements. The revision adds a requirement to verify that the MTC parameters are within these limits during startup testing at BOL. We consider this addition to Bases 3/4.1.1.3 acceptable.

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The licensee assessed the impact of a positive MTC on the accident analyses presented in Chapter 15 of the Diablo Canyon Units 1 and 2 updated FSAR. Those incidents which were found to be sensitive to positive MTC were re-analyzed. These are: (A) Uncontrolled Boron Dilution, (B) Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition, (C) Uncontrolled RCCA Bank Withdrawal at Power, (D) Complete Loss of Forced Reactor Coolant Flow, (E) Single Reactor Coolant Pump Locked Rotor, (F) Loss of External Electrical Load and/or Turbine Trip, (G) Loss of Normal Feedwater/Loss of Offsite Power, (H) Rupture of a Main Feedwater Pipe, (I) Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection), and (J) Accidental RCS Depressurization. In general, these incidents cause the reactor coolant temperature to rise.

The licensee states that the re-analysis is based on the identical analysis methods, computer codes, and assumptions employed in the updated FSAR. The same safety criteria are used during re-analysis, e.g., DNBR limit, peak cladding temperature limit, and the 280 cal/gm fuel enthalpy limit. The results show that all the safety criteria are met for the proposed Technical Specification change of positive MTC, and no significant reduction in the safety margin is observed. We therefore conclude that the Technical Specification changes in Sections 3/4.1.1.3 and Bases 3/4.1.1.3 concerning a positive MTC are acceptable. This change did not affect GDC-11.

F^N-delta-H Partial Power Multiplier

The licensee proposes to change the F^N-delta-H partial power multiplier from 0.2 to 0.3 at low power. The change involves Technical Specification Section 3/4.2.3, Figure 3.2-3a, Bases 2.1.1, and Figure 2.1-1. The proposed revision would allow optimization of core loading patterns by minimizing the restriction on F^N-delta-H at lower power levels. The increase in the partial power multiplier from 0.2 to 0.3 has a direct impact on DNBR calculations. We have previously approved a 0.3 partial power multiplier for a number of operating plants including Turkey Point, Ginna, Trojan, Cook Unit 1, Zion, Indian Point Unit 3, Point Beach, and Surry.

The licensee assessed the impact of larger F^N-delta-H on thermal-hydraulic design, nuclear design, and accident conditions. The results showed that (1) the DNBR safety limit is not violated, (2) there is no impact on other nuclear design bases, and (3) overtemperature and overpower setpoints are not impacted by the proposed F^N-delta-H change for non-LOCA accidents and the F^N-delta-H increase has no effect on the LOCA analyses. We thus consider that the Technical Specification changes involving Section 3/4.2.3, Figure 3.2-3a, Bases 2.1.1, and Figure 2.1-1 of F^N-delta-H are acceptable.

Deleting The Design Feature of The Total Weight of Uranium in A Fuel Rod.

The licensee proposes to delete a design quantity describing the maximum total weight of uranium from Technical Specification Section 5.3. The licensee indicates that the total uranium weight is intended to be descriptive and has not been used as an input to any safety analysis. We agree with the licensee's statement. Therefore, we conclude that the deletion of total uranium weight from Technical Specification Section 5.3 is acceptable.

Revised ECCS Analysis

By letter dated August 19, 1986 from J. D. Shiffer (PG&E) to S. A. Varga (NRC), the licensee submitted a revised LOCA analysis for Unit 1 Cycle 2. The revised LOCA analysis uses the most up-to-date NRC-approved Westinghouse LOCA Evaluation Model (1981 Evaluation Model with BART). The BART model is an improved version of the 1981 Model and is documented in WCAP-9561-P and WCAP-10062. The BART model for this analysis has been modified to consider the effect of core thimbles, the hot assembly power correction, and the BART heat transfer model conservatisms. These modifications are documented in WCAP-9561-P Addendum 3, which is approved by NRC in a letter from C. E. Rossi (NRC) to E. P. Rahe (Westinghouse) dated August 25, 1986. The result shows that the calculated peak cladding temperature is well within the 2200°F limit. We therefore conclude that the revised LOCA analysis is acceptable for Diablo Canyon Unit 1 Cycle 2. This August 19, 1986 letter did not change the Technical Specifications and thus was not noticed.

We have reviewed the licensee's submittal regarding the Diablo Canyon Unit 1 Cycle 2 reload core, associated Technical Specification changes, and a revised LOCA analysis. We conclude that the reload core design, Technical Specification changes, and the revised LOCA analysis are acceptable for Diablo Canyon Unit 1 Cycle 2. We also conclude that the changes related to the moderator temperature coefficient limits and the total weight of uranium in a fuel rod are acceptable for Unit 2.

ENVIRONMENTAL CONSIDERATION

These amendments involve a change in the installation or use of the facilities components located within the restricted areas as defined in 10 CFR 20. The staff has determined that these 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 these amendments involve no significant hazards consideration and there has been no public comment on such finding. 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.

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 the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: October 21, 1986

PRINCIPAL CONTRIBUTOR:

S. L. Wu