May 3, 1999

Mr. Gregory M. Rueger
Senior Vice President and General Manager
Pacific Gas and Electric Company
Diablo Canyon Nuclear Power Plant
P. O. Box 3
Avila Beach, California 93424

SUBJECT: ISSUANCE OF AMENDMENTS FOR DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 1 (TAC NO. MA3573) AND UNIT NO. 2 (TAC NO. MA3574)

Dear Mr. Rueger:

The Commission has issued the enclosed Amendment No.133 to Facility Operating License No. DPR-80 and Amendment No.131 to Facility Operating License No. DPR-82 for the Diablo Canyon Nuclear Power Plant (DCNPP), Unit Nos. 1 and 2, respectively. The amendments consist of changes to the combined Technical Specifications (TS) in response to your application dated September 3, 1998, as supplemented by letters dated January 22, February 5, and March 17, 1999.

These amendments revise the combined Technical Specifications (TS) for the Diablo Canyon Power Plant (DCPP) Unit Nos. 1 and 2 to revise TS 3/4.4.9.1, "Reactor Coolant System - Pressure/Temperature Limits," Figure 3.4-2,"Reactor Coolant System Heatup Limitations - Applicable Up to 12 EFPY," and Figure 3.4-3, "Reactor Coolant System Cooldown Limitations - Applicable Up to 12 EFPY," to extend the applicability up to 16 effective full power years (EFPY). The affected TS Bases would also be appropriately revised.

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

Sincerely,
Original Signed By
Steven D. Bloom, Project Manager
Project Directorate IV-2
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

DF01

Docket Nos. 50-275
and 50-323

Enclosures: 1. Amendment No. 133 to DPR-80
2. Amendment No. 131 to DPR-82
3. Safety Evaluation

cc w/encl: See next page

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DATE	4/23/99	4/23/99	4/28/99	4/30/99

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001
May 3, 1999

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Senior Vice President and General Manager
Pacific Gas and Electric Company
Diablo Canyon Nuclear Power Plant
P. O. Box 3
Avila Beach, California 93424

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These amendments revise the combined Technical Specifications (TS) for the Diablo Canyon Power Plant (DCPP) Unit Nos. 1 and 2 to revise TS 3/4.4.9.1, "Reactor Coolant System - Pressure/Temperature Limits," Figure 3.4-2,"Reactor Coolant System Heatup Limitations - Applicable Up to 12 EFPY," and Figure 3.4-3, "Reactor Coolant System Cooldown Limitations - Applicable Up to 12 EFPY," to extend the applicability up to 16 effective full power years (EFPY). The affected TS Bases have also been revised.

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Sincerely,

Steven D. Bloom, Project Manager, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-275
and 50-323

Enclosures: 1. Amendment No. 133 to DPR-80
2. Amendment No. 131 to DPR-82
3. Safety Evaluation

cc w/encl: See next page

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PACIFIC GAS AND ELECTRIC COMPANY

DOCKET NO. 50-275

DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 133
License No. DPR-80

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

   A. The application for amendment by Pacific Gas and Electric Company (the licensee) dated September 3, 1998, as supplemented by letters dated January 22, 1999, February 5, 1999, and March 17, 1999, 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 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. 133, are hereby incorporated in the license. Pacific Gas and 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 is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Stephen Dembek, Chief, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: May 3, 1999

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

PACIFIC GAS AND ELECTRIC COMPANY

DOCKET NO. 50-323

DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 131
License No. DPR-82

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

   A. The application for amendment by Pacific Gas and Electric Company (the licensee) dated September 3, 1998, as supplemented by letters dated January 22, 1999, February 5, 1999, and March 17, 1999, 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 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) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 131, are hereby incorporated in the license. Pacific Gas and 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 is effective as of its date of issuance and shall be implemented within 30 days from date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Stephen Dembek, Chief, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: May 3, 1999

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 133 TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 131 TO FACILITY OPERATING LICENSE NO. DPR-82

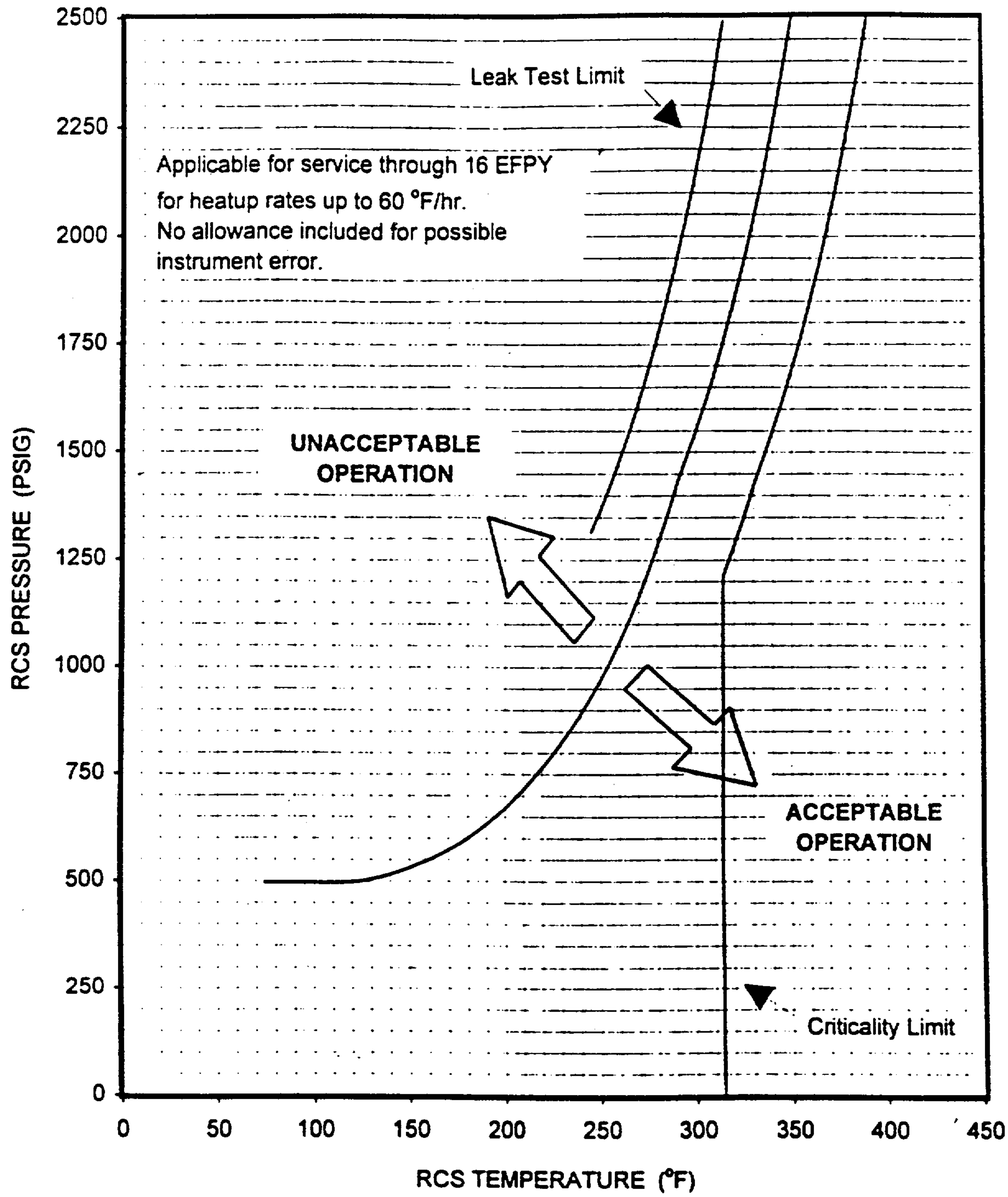
DOCKET NOS. 50-275 AND 50-323

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
viii	viii
3/4 4-31	3/4 4-31
3/4 4-32	3/4 4-32
B 3/4 4-7	B 3/4 4-7
B 3/4 4-12	B 3/4 4-12
B 3/4 4-15	B 3/4 4-15
B 3/4 4-16	B 3/4 4-16

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

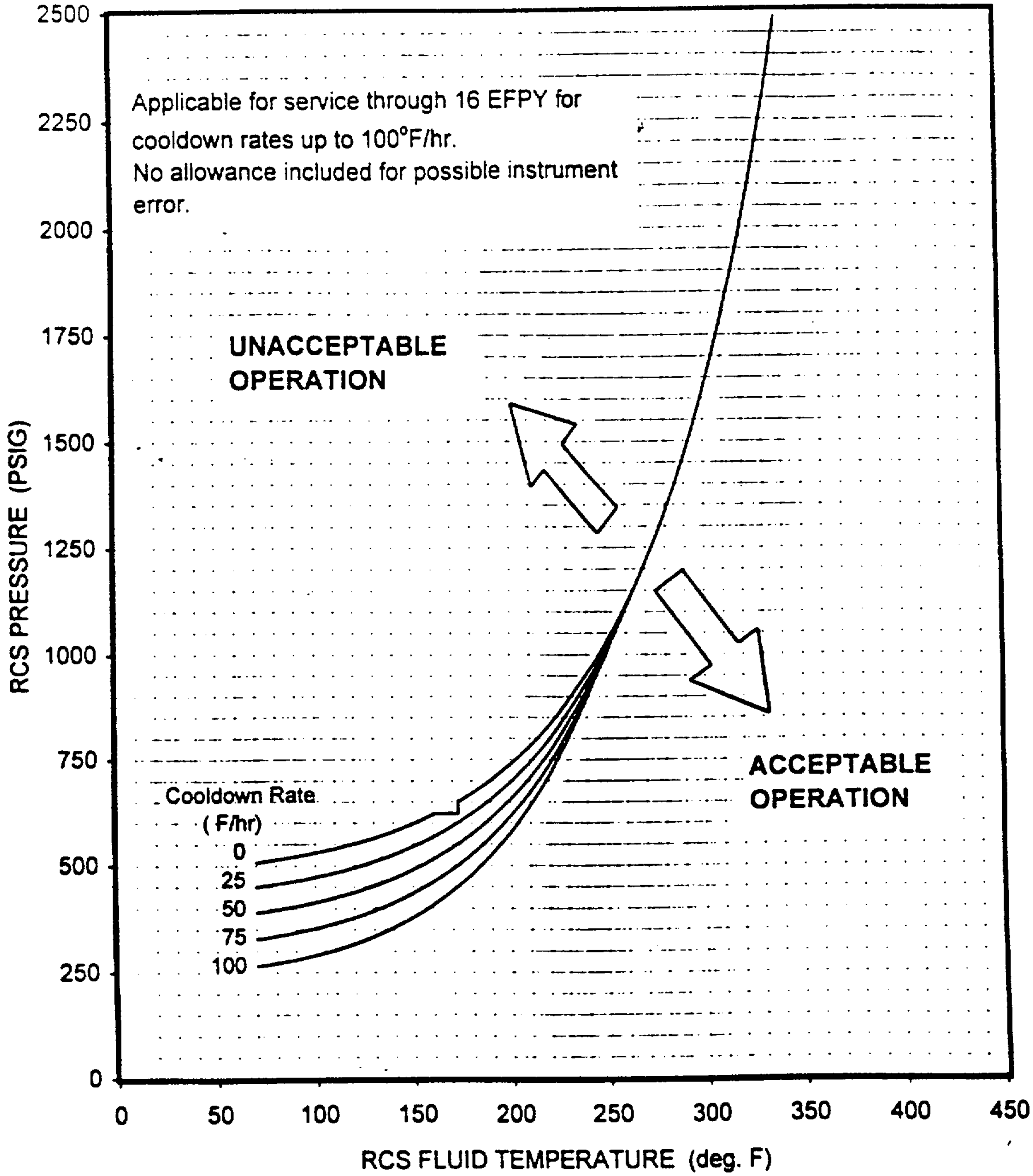
LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS



Controlling Material:

1/4T: Unit 1 Lower Shell Longitudinal Weld 3-442 C. RT_{ndt} @ 1/4T = 183.7°F.
3/4T: Unit 2 Intermediate Shell Plate B5454-2. RT_{ndt} @ 3/4T = 151.4°F.

FIGURE 3.4-2
REACTOR COOLANT SYSTEM HEATUP LIMITATIONS - APPLICABLE UP TO 16 EFPY



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3/4T: Unit 2 Intermediate Shell Plate B5454-2. RT_{ndt} @ 3/4T = 151.4°F.

FIGURE 3.4-3
REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS - APPLICABLE UP TO 16 EFPY

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G, and Section XI, Appendix G:

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 for the service period specified thereon:

    a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation; and

    b. Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. These limit lines shall be calculated periodically using methods provided below,

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4. Deleted

5. System preservice hydrotests and inservice leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI. Allowable pressures and temperatures for inservice leak and hydrostatic tests are given in Figure 3.4-2.

6. The criticality limit on Figure 3.4-2 is based on the minimum allowable temperature of 314°F for an inservice hydrostatic test of 110% of operating pressure.

The fracture toughness testing of the ferritic materials in the reactor vessel was performed in accordance with the 1966 Edition for Unit 1 and the 1968 Edition for Unit 2 of the ASME Boiler and Pressure Vessel Code, Section III. These properties are then evaluated in accordance with the NRC Standard Review Plan.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil ductility reference temperature, RT_{NDT}, at the end of 16 effective full power years (EFPY) of service life.

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PRESSURE/TEMPERATURE LIMITS (Continued)

The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in the FSAR Update. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, copper content and nickel content of the material in question, can be predicted using value of ΔRT_{NDT} computed by Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," for the maximum neutron fluence at the locations of interest. The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 16 EFPY.

Values of ΔRT_{NDT} determined in this manner will be used until the results from the material surveillance program, evaluated according to ASTM E185-82, can be used. Capsules will be removed in accordance with the requirements of ASTM E185 and 10 CFR Part 50, Appendix H. The surveillance specimen withdrawal schedule will be maintained in the FSAR Update. The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the reactor vessel. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section XI of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50 and these methods are discussed in detail in the following paragraphs.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside or outside surface of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section XI as the maximum postulated defect, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against non-ductile failure. To assure that the radiation embrittlement effects are accounted for in the

BASES

PRESSURE/TEMPERATURE LIMITS (Continued)

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

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

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

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

LOW TEMPERATURE OVERPRESSURE PROTECTION

The OPERABILITY of both Class 1 PORVs or an RCS vent opening of at least 2.07 square inches ensures that the RCS will be protected from pressure transients that could exceed 110% of the limits of Appendix G to 10 CFR Part 50 when operating at low temperatures. Low temperature is defined as less than or equal to the reactor coolant temperature corresponding to a reactor vessel wall temperature of RT_{NDT} + 50°F, where RT_{NDT} is evaluated at the beltline location (1/4T), which is controlling in the Appendix G Pressure-Temperature (60°F/hr heatup) limits. These pressure and temperature requirements are consistent with the guidelines and definitions in ASME Code Case N-514. The LTOP enable temperature applicable through 16 EFPY is 270°F.

LOW TEMPERATURE OVERPRESSURE PROTECTION (Continued)

OPERABILITY of the PORVs for LTOP use requires a lift setting of less than or equal to 435-psig. This setpoint ensures that either Class 1 PORV has adequate relieving capability to protect the RCS from overpressurization for all anticipated transients, concurrent with any single active failure. The limiting transient for LTOP is a mass injection event based on the combined ECCS injection line flow from one centrifugal charging pump and the positive displacement pump, into a water-sold RCS, with letdown isolated. The 435 psig setpoint was determined for this event based on a single, OPERABLE PORV, reactor service less than or equal to 16 EFPY, and administrative controls on RCP operation, charging pump operability, and the ECCS injection flow path. The instrument uncertainties are not included in the Technical Specification setpoints. Uncertainties associated with LTOP instrumentation were determined in accordance with the guidance provided in WCAP-14040-NP-A. An allowance for the pressure uncertainty is provided by administrative controls as discussed above.

The Maximum Allowed PORV Setpoint for the LTOPs will be modified, if required, based on the results of examinations of the reactor vessel material irradiation surveillance specimens performed as required by 10 CFR Part 50, Appendix H. The surveillance specimen withdrawal schedule is maintained in the FSAR Update.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 133 TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 131 TO FACILITY OPERATING LICENSE NO. DPR-82

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON NUCLEAR POWER PLANT, UNITS 1 AND 2

DOCKET NOS. 50-275 AND 50-323

1.0 INTRODUCTION

By letter dated September 3, 1998, as supplemented by letters dated January 22, 1999, February 5, 1999, and March 17, 1999, Pacific Gas and Electric Company (licensee) submitted a license amendment request to revise TS 3.4.9.1, Figures 3.4-2 and 3.4-3 regarding the Appendix G pressure temperature (P/T) limits to extend the applicability up to 16 effective full power years (EFPY). The licensee also updated the controlling materials for the P/T curves to reflect the current analysis for generating the new curves. The licensee also requested an exemption from 10 CFR 50.60 and Section IV.A.2 to Appendix G to 10 CFR Part 50, for use of the American Society of Mechanical Engineers (ASME) Code Case N-514 in determining the acceptable low temperature overpressure protection (LTOP) system setpoints. The licensee proposed revision to the TS Bases to reflect the above changes. The use of ASME Code Case N-514 would compensate for the more restrictive Appendix G P/T limits in the proposed TS affecting LTOP setpoints. The licensee has provided the justification to support its determination that the current LTOP setpoints are unchanged.

The January 22, 1999, February 5, 1999, and March 17, 1999, supplemental letters provided additional clarifying information and did not change the initial no significant hazards consideration determination published in the Federal Register on December 16, 1998 (63 FR 69345).

2.0 BACKGROUND

The LTOP system mitigates overpressure transients at low temperatures so that the integrity of the reactor coolant pressure boundary is not compromised by violating the 10 CFR Part 50, Appendix G, P/T limits under steady state operating conditions. Diablo Canyon Units 1 and 2 LTOP system use the pressurizer power operated relief valves (PORV) or a reactor coolant system (RCS) vent with the reactor depressurized to accomplish this function. The system is manually enabled by operators and uses a single lifting setpoint for the PORV. The design basis of Diablo Canyon Units 1 and 2 LTOP considers both mass-addition and heat-addition transients. The results of a licensee's evaluation indicated that the mass-addition transients are most limiting for the design of LTOP system. The limiting mass-addition analyses account for the injection from one centrifugal charging pump to the water solid RCS with letdown isolated. The heat-addition analyses accounts for heat input from the secondary side of the steam generators into the RCS upon starting a single reactor coolant pump (RCP) when the steam generator secondary water temperature is less than 50°F above the RCS cold leg temperature. The plant administrative controls in the current TSs and operating procedures provide restrictions in plant operation within the configuration assumed in the analysis for LTOP system design.

3.0 EVALUATION

3.1 Low Temperature Overpressure Protection

The current Limiting Condition for Operation (LCO) in TS 3.4.9.3 requires that an LTOP shall be operable with two operable PORVs with a lifting setting of less or equal to 435 psig, or the RCS depressurized with an RCS vent of ≥ 2.07 square inches. This LCO is applicable when any RCS cold leg temperature is ≤ 270°F when the head is on the reactor vessel. When any RCS cold leg temperature is below 107°F, a vent path greater than 2.07 square inches will be maintained per plant operating procedures. Also, the current TS 3.5.3 provides restrictions for a maximum of one operable centrifugal charging pump and no operable safety injection pump when any RCS cold leg temperature is less than or equal to 270°F. These TS restrictions in combination with other administrative controls in the plant operating procedures regarding RCP operation and ECCS flow path assure that the Diablo Canyon Units 1 and 2 will be operated within the configuration assumed in the analysis for LTOP system design. The method currently used for providing operational restriction in this area is consistent with NUREG-1431 and the licensing bases at Diablo Canyon Units 1 and 2.

Since the use of ASME Code Case N-514 compensates for the more restrictive Appendix G P/T limits in the proposed TS affecting LTOP setpoints, the licensee has proposed to maintain the current LTOP setpoints unchanged. The staff's evaluation of the licensee's justification regarding the LTOP setpoints are discussed below.

3.1.1 Enable Temperature

The LTOP system enable temperature is the temperature below which the LTOP system is required to be operable. The licensee has proposed to maintain the current LTOP enable temperature of 270°F. The licensee has performed an evaluation for the adequacy of the enable temperature of 270°F by using a methodology to: (1) account for instrument uncertainties associated with the instrumentation used to enable the LTOP system and, (2) implement ASME Code Case N-514 that uses an enable RCS water temperature corresponding to a metal temperature of at least RT_{NDT} + 50°F at the belt line location (1/4t or 3/4t). Therefore, the licensee calculates the enable temperature as RT_{NDT} + 50°F + temperature difference between RCS and metal + instrument uncertainties. Using the above equation, the calculated minimum enable temperature is 264°F. The licensee proposed an enable temperature of 270°F that includes an additional margin of 6°F.

The staff finds that this proposed LTOP enable temperature is conservative with respect to the enable temperature allowed by ASME Code Case N-514 and the methodology presented in WCAP-14040, Revision 1. Therefore, the staff finds it acceptable.

3.1.2 LTOP Actuation Setpoint

The LTOP system is designed to mitigate overpressure transients at low temperatures to prevent violating 10 CFR Part 50, Appendix G P/T limits. Additionally, since the licensee is using ASME Code Case N-514 to determine the acceptable LTOP system setpoints, the NRC staff has accepted the use of the P/T limits which are 10 percent above the steady-state Appendix G limits for the design of LTOP system. The LTOP system actuation setpoint is the pressure at which the PORVs will lift, when the LTOP system is enabled, to limit the peak RCS pressure within the acceptable limits during a pressurization transient.

Diablo Canyon Units 1 and 2 use PORVs to provide pressure relief capacity for LTOP system. The methodology used for determining the PORV actuation setpoint is consistent with the methodology presented in WCAP-14040, Revision 1.

The licensee has proposed that the current PORV actuation setpoint of 435 psig in TS 3.4.9.3 will remain unchanged to protect the proposed Appendix G P/T limits in Figures 3.4-2 and 3.4-3. In response to the staff's request, the licensee, in its letter dated January 2, 1999, provided a tabulation to list PORV setpoints, transient pressures overshoot, instrumentation uncertainties for temperature and pressure and corresponding P/T limit under various temperature conditions below the LTOP system enable temperature. The data presented in this tabulation confirms that the proposed PORV setpoints will provide adequate protection to the 10 CFR Part 50, Appendix G P/T limits under steady state conditions during a design basis overpressure transient (mass-addition or heat-addition) as described in Section 2.0 of this report. Based on the above discussion, the staff finds the proposed PORV setpoint acceptable.

3.1.3 RCS Vent Size

With the RCS depressurized, the results of the licensee's evaluation showed that a vent size of 2.07 square inches is capable of mitigating a most limiting low temperature overpressure transient. The vent size of 2.07 square inches is larger than the vent path area that a PORV, with a minimum throat diameter of 1.625 inches, would provide in its fully open position. The staff finds it acceptable.

3.1.4 LTOP Review

The staff has reviewed the licensee's justification for the unchanged LTOP system enable temperature and PORV actuation setpoint as discussed in Sections 3.1.1 and 3.1.2 above. The licensee has considered instrument uncertainties in its setpoint calculation using ISA S67.04-1994. The staff finds that the licensee's analyses were performed in a manner consistent with the approved methodology and that the results of the analyses conservatively demonstrated that the 10 CFR Part 50, Appendix G P/T limits up to 16 EFPYs will be adequately protected with these current LTOP setpoints, and, therefore, the staff finds the licensee's proposal acceptable.

3.2 Pressure-Temperature Limit Curves

3.2.1 Basis for the Staff's Assessment

The NRC has established requirements in 10 CFR Part 50 to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The staff evaluates the P-T limits based on the following NRC regulations and guidance: 10 CFR Part 50, Appendix G; Regulatory Guide 1.99, Revision 2; Standard Review Plan Section 5.3.2 (SRP 5.3.2); and Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code (Appendix G to the Code).

Appendix G of 10 CFR Part 50, requires that the P/T limits for an operating plant must be at least as conservative as those that would be generated if the methods of Appendix G to the Code were applied. The basic parameter in Appendix G to the Code for calculating P/T limit curves is the stress intensity factor K_I, which is a function of the stress state and flaw configuration. The methodology of Appendix G to the Code postulates the existence of a sharp surface flaw in the reactor pressure vessel (RPV) that is normal to the direction of the maximum stress. The maximum flaw size in the RPV is postulated to have a depth that is equal to one-fourth of the RPV beltline thickness and a length equal to 1.5 times the RPV beltline thickness. The methodology of Appendix G to the Code requires that the licensees determine the K_I factors, which vary as a function of temperature, from the RCS operating temperatures and from the adjusted reference temperatures (ARTs) for the limiting materials in the RPV beltline region. The critical locations in the RPV beltline region for calculating the ARTs used in the generation of the P/T limit curves are the 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations, which correspond to the depth of the maximum postulated flaw, if initiated and grown from the inside and outside surfaces of the RPV, respectively.

RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, provides an acceptable method for calculating the ARTs for ferritic RPV materials. The ART is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin (M) term. ΔRT_{NDT} is a product of a chemistry factor and a fluence factor. The chemistry factor (CF) is dependent upon the amount of copper and nickel in the material and may be determined from the tables in RG 1.99, Revision 2, or from surveillance data obtained from the plant's applicable reactor vessel material surveillance program. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depths. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the chemistry factor was determined using the tables in RG 1.99, Revision 2, or surveillance data. The margin term is used to account for uncertainties in the values of initial RT_{NDT}, copper and nickel contents, fluence and calculational procedures. RG 1.99, Revision 2, also describes the methodology to be used in calculating the M term.

Appendix G of 10 CFR Part 50 imposes the following restrictions on the calculation of P/T limits for an operating nuclear plant:

- During normal operation[1], at times when the reactor core is not critical or during hydrostatic pressure or leak rate testing, Appendix G of 10 CFR Part 50 requires that the P/T limits must be at least as conservative as those which would be generated by applying the methods of Appendix G to the Code;

- During normal operations when the reactor is critical, the requirement in Appendix G of 10 CFR Part 50 changes by adding 40°F to the values that would be obtained through application of the methods in Appendix G to the Code.

Appendix G of 10 CFR Part 50 also imposes certain minimum temperature requirements (MTRs) on a RPV at a nuclear power plant. The degree of conservatism in the MTRs specified in Appendix G of 10 CFR Part 50 are dependent upon mode of operation, criticality of the core, and on degree of pressurization of the RCS relative to the preservice hydrostatic test pressure (PHTP). The MTRs for PWRs are summarized below:

- For normal operating or pressure testing conditions of the RCS, when the RCS pressure is less than or equal to 20 percent of the PHTP and the reactor core is not critical, the MTR is equal to the limiting ART for the RPV closure flange.

- For pressure testing conditions of the RCS, when the RCS pressure is greater than 20 percent of the PHTP and the reactor core is not critical, the minimum temperature requirement is equal to the limiting ART for the RPV closure flange plus 90°F.

- For normal operations, when the RCS pressure is greater than 20 percent of the PHTP and the reactor core is not critical, the MTR is equal to the limiting ART for the RPV closure flange plus 120°F.

- For normal operations, when the RCS pressure is less than or equal to 20 percent of the PHTP and the reactor core is critical, the MTR is equal to the larger of either the minimum temperature for the inservice hydrostatic test or a temperature that is equal to the sum of the limiting ART for the RPV closure flange and 40°F.

- For normal operations, when the RCS pressure is greater than 20 percent of the PHTP and the reactor core is critical, the MTR is equal to the larger of either the minimum temperature for the inservice hydrostatic test or a temperature that is equal to the sum of the limiting ART for the RPV closure flange and 160°F.

Table 1 of Appendix G of 10 CFR Part 50 summarizes these requirements in slightly more detail. The composite P/T limit curves are generated by superimposing the appropriate minimum temperature requirements over the most limiting generated P/T limit curves for the units, and selecting the most conservative P/T data to establish the limiting composite curves

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[1] Appendix G to the Code considers normal operating conditions to include conditions of the plant during normal power operations of the reactor, during heatups and cooldowns of the reactor core either critical or not critical, and during anticipated operational transients.

for the plants. Exemptions from complying with requirements of 10 CFR Part 50, Appendix G, must be requested when it is determined that the P/T limits for an operating plant do not meet the criteria stated in the rule.

3.2.2 Evaluation

For the DCPP RPVs, PG&E provided the heatup, cooldown and pressure test curve figures for DCPP effective to 16 EFPY.[2] The staff determined that PG&E opted to use the technical methods provided in non-mandatory Appendix A to Section XI of the ASME Code as the methodology for generating the DCPP P/T limit curves. PG&E's P/T limits are based on the ART values for the most limiting materials in the DCPP RPVs at the 1/4T and 3/4T locations.

To test the validity of PG&E's proposed curves, the staff performed an independent assessment of the licensee's submittal. The staff applied the methodologies of Appendix G to the Code and Appendix G of 10 CFR Part 50, as the bases for its independent assessment. The assessment included an independent calculation of the 1/4T and 3/4T ART values effective to 16 EFPY for the limiting materials in the DCPP RPV beltlines, and independent generation of the P-T limit curves for the DCPP RPVs effective to 16 EFPY. For the evaluation of the limiting beltline materials, the calculations of the ART values were based on the methodology in RG 1.99, Revision 2.[3]

PG&E's proposed P/T limit curves for normal operating and pressure testing conditions, effective to 16 EFPY, are slightly more conservative than the P/T limit curves generated by the staff in accordance with the methods of Appendix G to the Code. The curves are in compliance with Appendix G of 10 CFR Part 50 and provide sufficient assurance that the DCPP reactors will be operated in a manner that will protect DCPP RPVs against brittle fracture. The staff confirmed that PG&E's P/T limit curves included the appropriate MTRs that were at least as conservative as those required by Appendix G of 10 CFR Part 50. Given these considerations, the staff therefore concludes that the proposed P/T limit curves, effective to 16 EFPY, are in compliance with Appendix G of 10 CFR Part 50 and acceptable for incorporation into the DCPP TS.

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[2] DCPP TS Figure 3.4-2 includes the proposed leak rate test curve, the heatup curve for the DCPP reactors when the cores are not critical, and heatup curve for the reactors when cores are critical. The heatup curves are based on heatup rates up to 60°F/Hr. DCPP TS Figure 3.4-3 includes the cooldown curves for DCPP reactors at cooling rates of 0°F/Hr, 25°F/Hr, 50°F/Hr, 75°F/Hr, and 100°F/Hr.

[3] For the limiting DCPP beltline material, the staff confirmed that at16 EFPY, DCPP Unit 1 Lower Shell Longitudinal Weld 3-442 C was the limiting beltline location for the assessment at the RPV 1/4T location, and that the ART value for this weld was appropriately calculated to be 183.7°F. This value is based on a fluence of 0.434E19 n/cm^2 and the latest weld chemistry provided in CEOG Task Report CE NPSD-1039, Rev. 2. The staff similarly confirmed DCPP Unit 2 Intermediate Shell Plate B5454-2 was the limiting beltline location for the assessment at the RPV 3/4T location, and that the ART value for this plate was appropriately calculated to be 151.4°F. This value is based on a fluence of 0.152E19 n/cm^2 and the latest chemistry for the plates provided by PG&E. All calculations were confirmed to be in accordance with the methods in RG 1.99, Rev.2.

4.0 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.

5.0 ENVIRONMENTAL CONSIDERATION

These amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC 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 (63 FR 69345). Accordingly, the 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 the amendments.

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

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Date: May 3, 1999