May 11, 1990

Docket Nos. 50-275
and 50-323

DISTRIBUTION:

~~RegionV (4)~~	EJordan
Docket Files	ACRS (10)
NRC & LPDRs	GHill (8)
PDV Plant File	WandaJones
JLarkins	OGC
JZwolinski	DHagan
PShea	JCalvo
HRood	JTsao
OC/LFMB	BElliot
GPA/PA	CYCheng

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

Dear Mr. Shiffer:

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. 74100 AND 74101)

The Commission has issued the enclosed Amendment No. 54 to Facility Operating License No. DPR-80 and Amendment No. 53 to Facility Operating License No. DPR-82 for the Diablo Canyon Power Plant (DCPP), Units 1 and 2, respectively. The amendments revise the Diablo Canyon combined Technical Specifications (TS) in response to your application for license amendments dated July 7, 1989 (Reference LAR 89-07). The amendments revise the TS to change the heatup and cooldown curves and delete the surveillance capsule withdrawal schedule. The revised heatup and cooldown curves were calculated using methods described in Revision 2 to NRC Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," as recommended by Generic Letter 88-11, to predict the effect of neutron radiation on reactor vessel materials while continuing to meet the requirements of 10 CFR Part 50, Appendix G. The Pressure/Temperature (P/T) limit curves in the previous TS were applicable through 6 effective full power years (EFPY). The revised P/T limits are applicable through 8 EFPY.

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

Sincerely,

ORIGINAL SIGNED BY H. ROOD

Harry Rood, Senior Project Manager
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects Office of
Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 54 to DPR-80
2. Amendment No. 53 to DPR-82
3. Safety Evaluation

cc w/enclosures:
See next page

			HR for JL
DRSP/PD5	DRSP/PD	OGC	DRSP/(A)D:PD5
PShea*	HRood*	BMB*	JLarkins
03/12/90	03/08/90	03/20/90	05/11/90

*See Previous Concurrence

DF01
1/1

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

May 11, 1990

Docket Nos. 50-275
and 50-323

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

Dear Mr. Shiffer:

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. 74100 AND 74101)

The Commission has issued the enclosed Amendment No. 54 to Facility Operating License No. DPR-80 and Amendment No. 53 to Facility Operating License No. DPR-82 for the Diablo Canyon Power Plant (DCPP), Units 1 and 2, respectively. The amendments revise the Diablo Canyon combined Technical Specifications (TS) in response to your application for license amendments dated July 7, 1989 (Reference LAR 89-07). The amendments revise the TS to change the heatup and cooldown curves and delete the surveillance capsule withdrawal schedule. The revised heatup and cooldown curves were calculated using methods described in Revision 2 to NRC Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," as recommended by Generic Letter 88-11, to predict the effect of neutron radiation on reactor vessel materials while continuing to meet the requirements of 10 CFR Part 50, Appendix G. The Pressure/Temperature (P/T) limit curves in the previous TS were applicable through 6 effective full power years (EFPY). The revised P/T limits are applicable through 8 EFPY.

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

Sincerely,

Harry Rood

Harry Rood, Senior Project Manager
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects Office of
Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 54 to DPR-80
2. Amendment No. 53 to DPR-82
3. Safety Evaluation

cc w/enclosures:
See next page

Mr. J. D. Shiffer
Pacific Gas and Electric Company

Diablo Canyon

cc:
Richard F. Locke, Esq.
Pacific Gas & Electric Company
Post Office Box 7442
San Francisco, California 94120

Ms. Sandra A. Silver
660 Granite Creek Road
Santa Cruz, California 95065

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

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

Ms. Nancy Culver
192 Luneta Street
San Luis Obispo, California 93401

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

Mr. John Hickman
Senior Health Physicist
Environmental Radioactive Mgmt. Unit
Environmental Management Branch
State Department of Health Services
714 P Street, Room 616
Sacramento, California 95814

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

Bruce Norton, Esq.
c/o Richard F. Locke, Esq.
Pacific Gas and Electric Company
Post Office Box 7442
San Francisco, California 94120

Dr. R. B. Ferguson
Sierra Club - Santa Lucia Chapter
Rocky Canyon Star Route
Creston, California 93432

Chairman
San Luis Obispo County Board of
Supervisors
Room 270
County Government Center
San Luis Obispo, California 93408

Michael M. Strumwasser, Esq.
Special Assistant Attorney General
State of California
Department of Justice
3580 Wilshire Boulevard, Room 800
Los Angeles, California 90010

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. 54
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 July 7, 1989, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) 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. 54, 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

H. Rood for

John T. Larkins, Acting Director
Project Directorate V
Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: May 11, 1990

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. 53
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 July 7, 1989, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) 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. 53, 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

H. Rood for

John T. Larkins, Acting Director
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 11, 1990

ATTACHMENT TO LICENSE AMENDMENT NOS. 54 AND 53

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 areas of change. Overleaf pages are also included, as appropriate.

Remove Page	Insert Page
viii	viii
xv	xv
3/4 4-30	3/4 4-30
3/4 4-31	3/4 4-31
3/4 4-32	3/4 4-32
3/4 4-33	3/4 4-33
B 3/4 4-7	B 3/4 4-7
B 3/4 4-8	B 3/4 4-8
B 3/4 4-9	B 3/4 4-9
B 3/4 4-10	B 3/4 4-10
B 3/4 4-11	B 3/4 4-11
B 3/4 4-12	B 3/4 4-12
B 3/4 4-13	B 3/4 4-13

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

INDEX

BASES

SECTION

PAGE

TABLE 4.4-4 (Continued)

TABLE NOTATIONS

#Until the specific activity of the Reactor Coolant System is restored within its limits.

*Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

**A gross radioactivity analysis shall consist of the quantitative measurement of the total specific activity of the reactor coolant except for radionuclides with half-lives less than 10 minutes and all radioiodines. The total specific activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities in the sample within 2 hours after the sample is taken and extrapolated back to when the sample was taken. Determination of the contributors to the gross specific activity shall be based upon those energy peaks identifiable with a 95% confidence level. The latest available data may be used for pure beta-emitting radionuclides.

***A radiochemical analysis for $\bar{E}$ shall consist of the quantitative measurement of the specific activity for each radionuclide, except for radionuclides with half-lives less than 10 minutes and all radioiodines, which is identified in the reactor coolant. The specific activities for these individual radionuclides shall be used in the determination of $\bar{E}$ for the reactor coolant sample. Determination of the contributors to $\bar{E}$ shall be based upon those energy peaks identifiable with a 95% confidence level.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 100°F in any 1-hour period.

b. A maximum cooldown of 100°F in any 1-hour period.

c. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

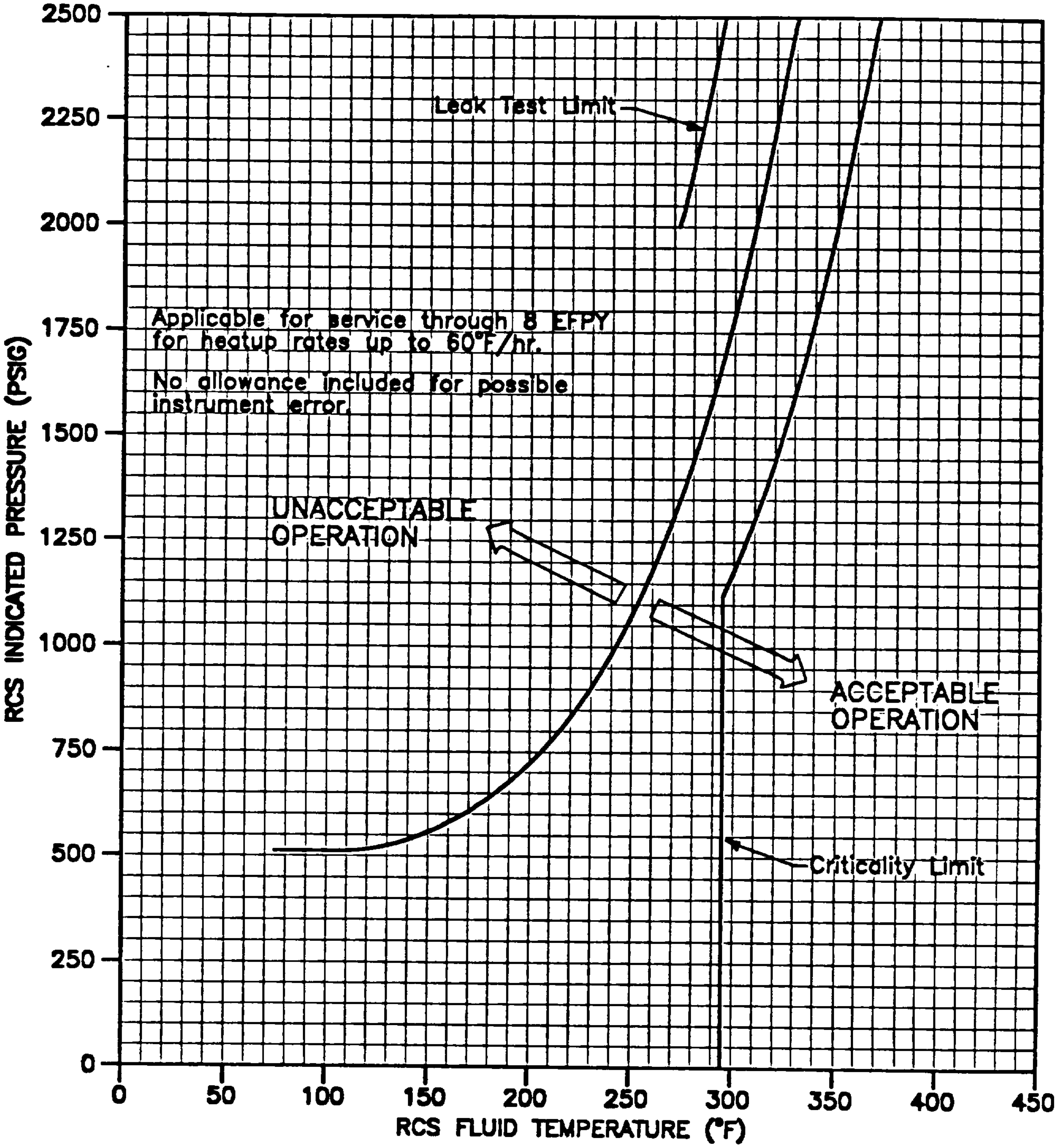
APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per hour during system heatup, cooldown, and inservice leak and hydrostatic testing operations.



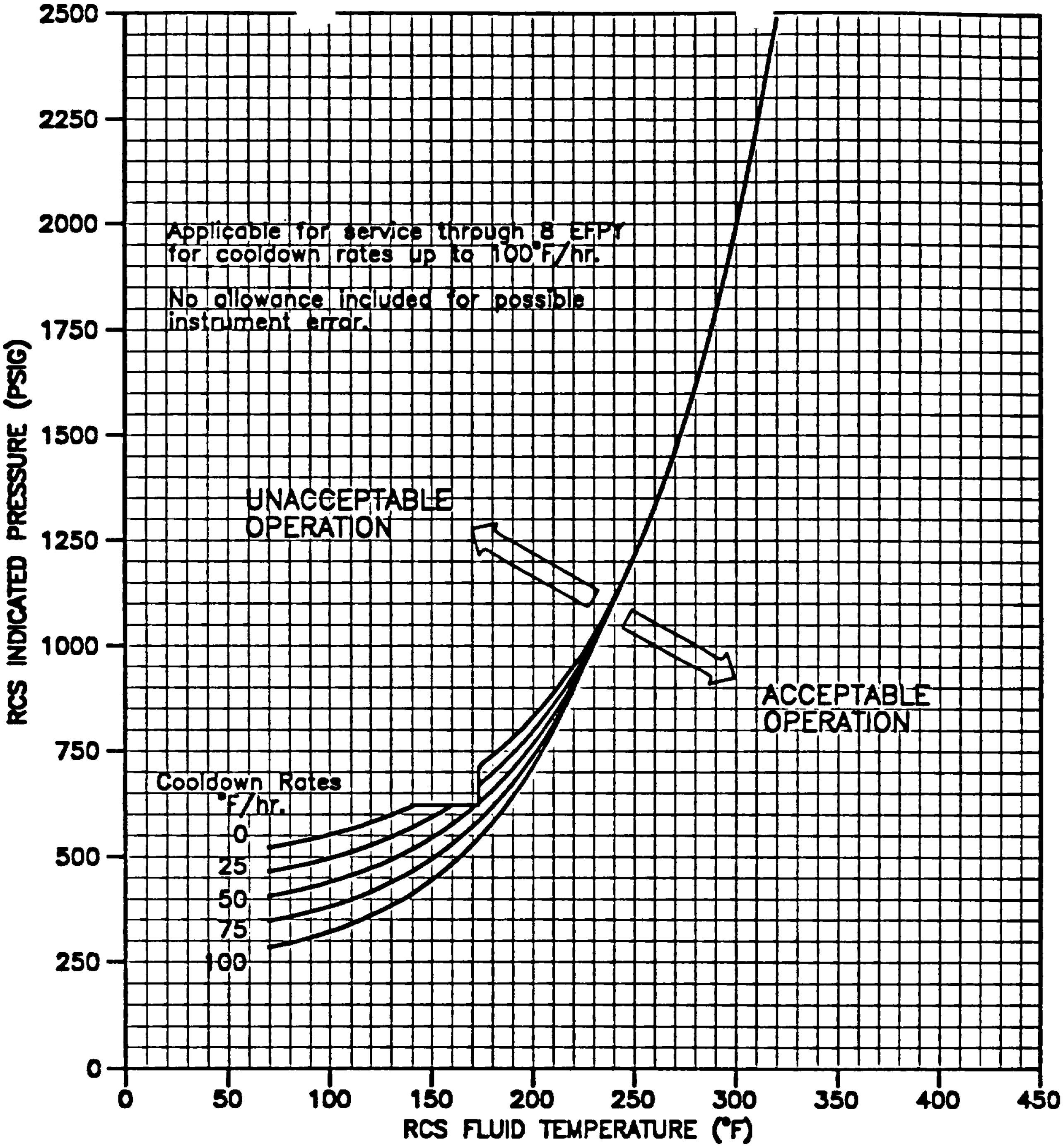
Controlling Material:

Unit 2 Intermediate Shell Plate B5454-2 0.14wt.% Cu 0.59wt.% Ni

Initial RT_{NDT}=67°F Projected RT_{NDT} 1/4T = 164°F 3/4T = 141°F

FIGURE 3.4-2

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS - APPLICABLE UP TO 8 EFPY



Controlling Material:

Unit 2 Intermediate Shell Plate B5454-2 0.14wt% Cu 0.59wt% Ni

Initial RT_{NDT}=67°F Projected RT_{NDT} 1/4T = 164°F 3/4T = 141°F

FIGURE 3.4-3

REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS - APPLICABLE UP TO 8 EFPY

THIS PAGE INTENTIONALLY DELETED.

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 200°F in any 1-hour period, and

c. A maximum spray water temperature differential of 560°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per hour during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during auxiliary spray operation.

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 Presure Vessel Code, Section III, 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. The pressurizer heatup and cooldown rates shall not exceed 100°F/hr and 200°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 560°F, and

5. System preservice hydrotests and in-service 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 295°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 8 effective full power years (EFPY) of service life. The 8 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region

THIS PAGE INTENTIONALLY DELETED.

THIS PAGE INTENTIONALLY DELETED.

THIS PAGE INTENTIONALLY DELETED.

THIS PAGE INTENTIONALLY DELETED.

PRESSURE/TEMPERATURE LIMITS (Continued)

is greater than the RT_{NDT} of the limiting unirradiated material. 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, "Effects of Residual Elements on Predicted Radiation Damage to 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 8 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 III 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 of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, 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

PRESSURE/TEMPERATURE LIMITS (Continued)

calculation of the limit curves, the most limiting value of the nil ductility reference temperature, RT_{NDT}, is used and this includes the radiation induced shift, ΔRT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T-RT_{NDT} + 160)] \quad (1)$$

Where: K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\, K_{IM} + K_{It} \leq K_{IR} \quad (2)$$

Where, K_{IM} = the stress intensity factor caused by membrane (pressure) stress,

K_{It} = the stress intensity factor caused by the thermal gradients,

K_{IR} = reference stress intensity factor provided by the Code as a function of temperature relative to the RT_{NDT} of the material,

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature of the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factors, K_{It}, for the reference flaw are computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

PRESSURE/TEMPERATURE LIMITS (Continued)

COOLDOWN

For the calculation of the allowable pressure versus coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IR} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

HEATUP

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stresses at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4T crack during heatup is lower than the K_{IR} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and different K_{IR}'s for steady-state and finite heatup rates do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 54 TO FACILITY OPERATING LICENSE NO. DPR-80
AND AMENDMENT NO. 53 TO FACILITY OPERATING LICENSE NO. DPR-82
PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2
DOCKET NO. 50-275 AND 50-323

1.0 INTRODUCTION

By letter dated July 7, 1989 (Reference LAR 89-07), Pacific Gas and Electric Company (PG&E or the licensee) requested amendments to the combined Technical Specifications (TS) appended to Facility Operating License Nos. DPR-80 and DPR-82 for the Diablo Canyon Power Plant (DCPP) Unit Nos. 1 and 2, respectively. The amendments change the TS to revise the heatup and cooldown curves and delete the surveillance capsule withdrawal schedule.

The revised heatup and cooldown curves were calculated using methods described in Revision 2 to NRC Regulatory Guide (RG) 1.99, "Radiation Embrittlement of Reactor Vessel Materials," as recommended by Generic Letter (GL) 88-11, to predict the effect of neutron radiation on reactor vessel materials while continuing to meet the requirements of 10 CFR Part 50, Appendix G. The previous pressure-temperature (P/T) limit curves were applicable through 6 effective full power years (EFPY) of plant operation. The revised TS provide up-to-date P/T limits for the operation of the reactor coolant system during heatup, cooldown, criticality, and hydrotest. The revised TS use one set of P/T limits for both units, and are applicable through 8 EFPY.

To evaluate the Diablo Canyon P/T limits, the staff used the following NRC regulations and guidance:

A. Appendix G of 10 CFR Part 50, "Fracture Toughness Requirements."

B. Appendix H of 10 CFR Part 50, "Reactor Vessel Material Surveillance Program Requirements."

C. The ASTM Standards and the ASME Code, which are referenced in Appendices G and H to 10 CFR Part 50.

D. Standard Review Plan (SRP) Section 5.3.2, "Pressure-Temperature Limits" (NUREG-0800, July 1981).

E. Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, May 1988.

F. Generic Letter 88-11, "NRC position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," July 12, 1988.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide technical specifications for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the technical specifications. The P/T limits are among the limiting conditions of operation in the technical specifications for all commercial nuclear plants in the United States. Appendices G and H of 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits. An acceptable method for constructing the P/T limits is described in SRP Section 5.3.2.

Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and requires that the beltline materials in the surveillance capsules be tested in accordance with Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE). Generic Letter 88-11 requested that licensees and permittees use the methods in RG 1.99, Rev. 2, to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Appendix H of 10 CFR Part 50 requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and that they contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline.

The staff evaluation of the changes proposed in the licensee's letter of July 7, 1989 is given below. The staff's proposed determination of no significant hazards consideration for this TS change was published in the Federal Register on August 9, 1989 at 54 FR 32715.

2.0 EVALUATION

The NRC staff has evaluated the proposed changes and finds them acceptable, based on its review of the analyses and evaluations presented by the licensee. A discussion of each of the specific technical specification changes made by these amendments is given below.

These amendments make changes in three areas. Each area is covered below as a separate item.

Item A Figure 3.4-1, "Reactor Coolant System Heatup Limitations," and Figure 3.4-3, "Reactor Coolant System Cooldown Limitations," are revised to update the controlling chemical composition and P/T curves.

In reviewing these changes, the staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the Diablo Canyon 1 and 2 reactor vessels. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff has determined that the material with the highest ART at 8 EFPY for both units is intermediate shell plate B5454-2 in Unit 2. This plate is 0.14% copper (Cu) and 0.59% nickel (Ni), and has an initial RT_{ndt} of 67°F.

The licensee has removed one surveillance capsule each from Diablo Canyon Units 1 and 2. The results from capsule S from Unit 1 were published in the Westinghouse Topical Report WCAP-11567. The results from capsule U from Unit 2 were published in Westinghouse Topical Report WCAP-11851. All surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline material, plate B5454-2, the staff calculated the ART to be 163°F at 1/4T (T = reactor vessel beltline thickness) and 142°F for 3/4T at 8 EFPY. The ART was determined using Section 1 of RG 1.99, Rev. 2.

The licensee used the method in RG 1.99, Rev. 2, to calculate an ART of 164°F at 8 EFPY at 1/4T for the same limiting plate material. The staff judges that a difference of 1°F between the licensee's ART of 164F°F and the staff's ART of 163°F is acceptable. The licensee used a technique based on Section XI of the ASME Boiler and Pressure Vessel Code to calculate the Diablo Canyon 1 and 2 P/T limits. The staff has reviewed this technique and finds it acceptable, although it produces a P/T curve that is lower than the one the staff calculated using the methods of SRP 5.3.2. Thus, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the requirements of Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions that are highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Based on the flange reference temperature of 53°F, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

Section IV.B of Appendix G requires that the predicted Charpy USE at end of life be above 50 ft-lb. Of the materials that have unirradiated Charpy USE data available, plate B4106-3 in Unit 1 had the lowest value, 77.5 ft-lb. Using the method in RG 1.99, Rev. 2, the predicted Charpy USE of the plate material at the end of life will be 59.7 ft-lb. This is greater than 50 ft-lb and, therefore, is acceptable. However, there are no unirradiated Charpy USE data for the intermediate to lower shell girth weld in Unit 1, and the intermediate to lower shell girth weld and lower shell longitudinal weld seams in Unit 2. The staff will obtain the USE of these welds in the future.

In summary, the NRC staff has concluded that the revised P/T limits for the Diablo Canyon Units 1 and 2 reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 8 EFPY because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The licensee's submittal also satisfies Generic Letter 88-11 because the licensee used the method in RG 1.99, Rev. 2 to calculate the ART. Consequently, the proposed P/T limits are hereby incorporated into the Diablo Canyon 1 and 2 Technical Specifications. Also, the staff finds the updated controlling chemical composition acceptable because it reflects measurements made on the limiting reactor vessel plate. Therefore, the TS changes identified under Item A, above, are acceptable.

Item B   TS 4.4.9.1.2 and TS Table 4.4-5, "Reactor Vessel Material Surveillance Program Withdrawal Schedule," are deleted.

The staff has reviewed the proposed deletion of TS 4.4.9.1.2 and the associated reactor vessel surveillance capsule withdrawal schedule (TS Table 4.4-5), and finds these changes acceptable on the basis that they are administrative rather than substantive. That is, 10 CFR 50, Appendix H, requires that a withdrawal schedule must be established and submitted to the NRC for approval prior to its implementation. The licensee has committed to include the currently approved withdrawal schedule in the next FSAR Update Revision. Repeating this requirement in the Technical Specifications is unnecessary and redundant. Therefore, the TS changes identified under Item B, above, are acceptable.

Item C TS Bases 3/4.4.9 are revised to update the information contained therein, and to delete certain figures and tables that will be included in the FSAR Update.

Deletion of these tables and figures from the TS Bases is an administrative change that reflects the other TS changes made by these amendments. Therefore, the TS changes under Item C, above, is acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve changes to a requirement with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and a change in surveillance requirements. At Diablo Canyon, the restricted area coincides with the site boundary. 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 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.

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 the health and safety of the public.

Principal Contributors: John Tsao
Harry Rood

Dated: May 11, 1990