Pacific Gas and Electric Company

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August 15, 2003

PG&E Letter DCL-03-101

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
Response to NRC Request for Additional Information Regarding "License Amendment Request 02-04, Revision of Technical Specification 5.6.6 – Reactor Coolant System Pressure and Temperature Limits Report"

Dear Commissioners and Staff:

In Pacific Gas & Electric (PG&E) letter DCL-02-079, "License Amendment Request (LAR) 02-04, Revision to Technical Specification (TS) 5.6.6 – Reactor Coolant System Pressure and Temperature Limits Report (PTLR)," dated July 31, 2002, PG&E requested NRC approval of the methodology to be used to make changes to the Diablo Canyon Power Plant PTLR without prior NRC approval.

In a telephone call on July 3, 2003, PG&E provided responses to NRC requests for additional information concerning LAR 02-04. Additional questions were transmitted by the NRC staff on July 17, 2003.

A summary of the responses PG&E provided in the July 3, 2003 phone call, and the PG&E responses to the questions transmitted by the NRC on July 17, 2003, are contained in Enclosure 1. Revised marked-up and retyped TS pages are provided in Enclosures 2 and 3, respectively.

This additional information does not affect the results of the technical evaluation and no significant hazards consideration determination previously transmitted in PG&E letter DCL-02-079.

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If you have any questions or require additional information, please contact Stan Ketelsen at (805) 545-4720 or Tom Grozan at (805) 545-4231.

Sincerely

David H. Oatley
Vice President and General Manager -Diablo Canyon

jer/3664
Enclosures

cc: Edgar Bailey, DHS
Thomas P. Gwynn
David L. Proulx
Diablo Distribution

cc/enc: Girija S. Shukla

PG&E Letter DCL-03-101

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	Docket No. 50-275
PACIFIC GAS AND ELECTRIC COMPANY	)	Facility Operating License
	)	No. DPR-80
Diablo Canyon Power Plant	)	Docket No. 50-323
Units 1 and 2	)	Facility Operating License
	)	No. DPR-82

AFFIDAVIT

David H. Oatley, of lawful age, first being duly sworn upon oath states that he is Vice President and General Manager - Diablo Canyon of Pacific Gas and Electric Company; that he has executed this response to the NRC request for additional information on License Amendment Request 02-04 on behalf of said company with full power and authority to do so; that he is familiar with the content thereof; and that the facts stated therein are true and correct to the best of his knowledge, information, and belief.

D. H. Oatley

David H. Oatley
Vice President and General Manager - Diablo Canyon

Subscribed and sworn to before me this 15th day of August, 2003.

Chuck Mackey

Notary Public
County of San Luis Obispo
State of California

CHUCK MACKEY
Commission # 1397547
Notary Public - California
San Luis Obispo County
My Comm. Expires Feb 1, 2007

Responses to Requests for Additional Information Concerning License Amendment Request 02-04 "Revision of Technical Specification 5.6.6 – Reactor Coolant System Pressure and Temperature Limits Report"

Requests for additional information Nos. 1 through 5 were discussed in a telephone call between the NRC staff and PG&E on July 3, 2003.

NRC Request No. 1

Referring to section 4.3 of the submittal f_s is calculated using the $f_{c/bm}$ values where f_s and $f_{c/bm}$ are the inside wetted surface and the clad base interface values respectively. Regulatory Guide (RG) 1.99, Revision 2, uses the term "inner wetted surface of the vessel at the location of the postulated defect" which is commonly understood to mean the clad-base metal interface. The metal flaw is on the base metal surface. The clad is not credited as part of the vessel structure. It is not clear from the submittal whether the 1/4t and 3/4t values accounted for the thickness of the cladding. In this context please explain how the fluence, temperatures and stresses were calculated. If indeed the cladding is part of the vessel thickness demonstrate that the values are conservative or make the necessary adjustments.

PG&E Response:

The metal flaw is evaluated to extend from the clad/base metal interface. The clad is not credited as part of the vessel structure for calculating the 1/4t and 3/4t temperature and stress. The cladding fluence attenuation is accounted for in estimating the fluence at each flaw.

NRC Request No. 2

Referring to enclosure 4 (the PTLR report) page 20, the projected RT_{PTS} = 259 degrees F at 32 effective full power years (EFPYs) of operation. (1) The references (WCAP-13771 and WCAP-14364) are dated from 1993 and 1995 respectively. The PTLR references do not include either one of these WCAPs. Has the projected value of the fluence been adjusted using the PTLR references? (2) The fluence values are stated in terms of EFPYs. Have there been any power uprates; are any power uprates planned, and is the projected load factor to remain at 80 percent for the life of the plant?

PG&E Response

The projected fluence value is reflective of the PTLR references. PG&E recognizes that fluence projections are subject to change due to uprates, loading patterns and load factors. PG&E intends to update the pressurized thermal shock evaluation in the PTLR when the capsule V data are incorporated to include any changes in

power rating, loading patterns or load factor. It is expected that the projected load factor will increase for the remaining life of the plant. Capsule V incorporation will be done as stated on page 10 of Enclosure 1 to PG&E Letter DCL-02-079, dated July 31, 2002 (LAR 02-04).

NRC Request No. 3

An annotated copy of requirements from Generic Letter 96-03 provided by the NRC for a telephone conference call on July 3, 2003, states that the DCPP PTLR should be updated to incorporate any impacts from the Unit 1 Capsule V and Unit 2 Capsule V data results (WCAP-15948 and WCAP-15423).

PG&E Response

Page 10 of Enclosure 1 to PG&E Letter DCL-02-079, dated July 31, 2002 (LAR 02-04), states:

> *PG&E letter DCL-01-004, "Reactor Vessel Material Surveillance Program Capsule V Technical Report," dated January 12, 2001, transmitted Westinghouse technical report, WCAP-15423, Revision 0, "Analysis of Capsule V from Pacific Gas and Electric Company Diablo Canyon Unit 2 Reactor Vessel Radiation Surveillance Program," dated September 2000. PG&E stated in DCL-01-004 that: (1) the results of the Capsule V specimen testing show that the limiting vessel beltline plate and weld material are behaving in accordance with previous predictions; (2) the results do not indicate any changes needed to the LTOP setpoints or P/T curves currently approved to 16 EFPY; and (3) that the PTLR will be updated with the Capsule V data upon approval of the PTLR methodology.*

WCAP-15948, which was issued in January 2003, reported the same conclusions for Unit 1 Capsule V data.

A followup recommendation provided by the NRC on July 23, 2003, requests that the next revision of the PTLR (to be submitted to the NRC for information only) reflect the latest data from the Unit 1 Capsule V and Unit 2 Capsule V data results (WCAP-15948 and WCAP-15423). It is PG&E's intention to do so.

NRC Request No. 4

An annotated copy of requirements from Generic Letter 96-03 provided by the NRC for a telephone conference call on July 3, 2003, states the staff would like clarification as to which addition of ASME Section XI, Appendix A, is being used for the Mm and Mb factor determinations.

PG&E Response

The 1983 Edition, S84 Addenda was used.

NRC Request No. 5

An annotated copy of requirements from Generic Letter 96-03 provided by the NRC for a telephone conference call on July 3, 2003, states the staff requests that the licensee provide the K_{it} values and delta-temperature values for the 1/4t and 3/4t locations of the vessel for each P-T point provided in the Tables for the PTLR.

PG&E Response

These values were provided in calculation NCM-97010 previously submitted to the NRC by PG&E Letter DCL-99-017, dated February 15, 1999.

Requests for additional information Nos. 6 through 9 were sent to PG&E by the NRC staff on July 17, 2003.

NRC Request No. 6

In license amendment request for the PTLR, dated July, 31, 2002, Pacific Gas and Electric Company's (PG&E's) proposed changes to Technical Specification (TS) Section 5.5.6 lists WCAP-14040-NP-A, Revision 2, as the approved methodology for generating the pressure-temperature (P-T) limits and low pressure overpressure protection system (LTOP) setpoint limits for Diablo Canyon Power Plant (DCPP) Units 1 and 2. The list of methodologies on TS page does not appear to be inclusive of all the approved methodologies that will be used as part of the PTLR process for generating the P-T limits and LTOP limits for the DCPP, Units 1 and 2. The staff requests that proposed TS 5.5.6 (TS page 5.0-28) be revised to include the following methodologies within the scope of the methodologies being credited for generation of the DCPP P-T limits and LTOP setpoint limits:

A. WCAP-14040-NP-A, Methodology Used to Develop Cold Overpressure, Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves.

B. Chapter 6.0 of WCAP-15958, Revision 0, Analysis of Capsule V from Pacific Gas and Electric Company Diablo Canyon Unit 1 Reactor Vessel Radiation Surveillance Program (January 2003).

PG&E Response

PG&E concurs with the addition of Chapter 6.0 of WCAP-15958 to the scope of methodologies being credited for generation of the DCPP P-T limits and LTOP setpoint limits. The Capsule V report concludes that the current adjusted reference temperature (ART) that forms the basis of the current PTLR bounds the ART based on the fluences determined within that report.

To maintain consistency with NRC approved Industry/TSTF Standard Technical Specification Change Traveler No. 419, PG&E has referenced the Topical Report (WCAP-15958) in TS 5.6.6 by number and title. As required by TSTF-419, the complete citation for each Topical Report cited in the PTLR will include the report number, title, revision, date, and any supplements.

Revised marked-up and retyped TS pages are provided in Enclosures 2 and 3, respectively.

NRC Request No. 7

The PTLR's (Enclosure 4 to the letter) evaluation of reactor vessel (RV) beltline materials that are represented in the DCPP RV material surveillance programs for Units 1 and 2 are based on the evaluation of surveillance data from Capsules S and Y for DCPP Unit 1 and Capsules U, X, and Y for DCPP Unit 2. Provide the following confirmatory responses with respect to the surveillance capsule data currently available from the RV materials surveillance programs for DCPP Units 1 and 2.

A. Confirm that the following WCAPs provide the most current surveillance capsule reports and updated surveillance data for DCPP Units 1 and 2:

- Unit 1: WCAP-15958, Revision 0, Analysis of Capsule V from Pacific Gas and Electric Company Diablo Canyon Unit 1 Reactor Vessel Radiation Surveillance Program (January 2003).

- Unit 2: WCAP-15423, Revision 0, Analysis of Capsule V from Pacific Gas and Electric Company Diablo Canyon Unit 2 Reactor Vessel Radiation Surveillance Program (September 2000).

B. Confirm that the updated surveillance data in WCAP-15958, Revision 0, for DCPP Unit 1 Capsules S, Y, and V and in WCAP-15423, Revision 0, for DCPP Unit 2 Capsules U, X, Y, and V will not affect the adjusted reference temperatures used to generate the P-T limits in DCPP Report No. PTLR-1, Revision 2 (i.e., the RT_{NDT} values for the limiting 1/4t and 3/4t beltline materials in the DCPP Units 1 and 2 reactor vessels) because the chemistry factors for the limiting materials are

conservatively being generated using the chemistry factor tables in Regulatory Guide 1.99, Revision 2 (May 1988).

PG&E Response

A. WCAP-15958, Revision 0, and WCAP-15423, Revision 0, provide the most current surveillance capsule reports and updated surveillance data for DCPP Units 1 and 2.

B. The updated surveillance data in WCAP-15958, Revision 0, for DCPP Unit 1 Capsules S, Y, and V and in WCAP-15423, Revision 0, for DCPP Unit 2 Capsules U, X, Y, and V do not affect the adjusted reference temperatures used to generate the P-T limits in DCPP Report No. PTLR-1, Revision 2.

NRC Request No. 8

Confirm that the 16 EFPY P-T limit curves incorporated in PG&E Report No. PTLR-1, Revision 2, are identical to those P-T limit curves that were approved in license amendment No. 133 to Facility Operating License No. DPR-80 for DCPP Unit 1 and license amendment No. 131 to Facility Operating License No. DPR-82 for DCPP Unit 2 (both dated May 3, 1999).

PG&E Response

The 16 EFPY P-T limit curves incorporated in PG&E Report No. PTLR-1, Revision 2, are identical to those that were approved in License Amendments 133 and 131 for Units 1 and 2, respectively, with the exception of bolt up temperature. The bolt up temperature value is discussed PG&E Letter DCL-02-079, dated July 31, 2002, Enclosure 1, page 6, item 3:

> *The bolt up temperature, based on ASME Appendix G and 10 CFR 50 Appendix G, Table 1, is required to be the initial nil-ductility temperature (RT_{NDT}) of the flange plus any irradiation effects. The highest initial RT_{NDT} of the vessel and closure head flange materials is 53°F. The flange area is sufficiently distant from the fuel region, that the fluence has negligible affect on the RT_{NDT} of the materials in this area. Currently, the bolt up temperature of 70°F is based on the value given in the original Combustion Engineering (CE) instruction manual for the RV. The proposed curves set the temperature at 60°F based on the Westinghouse WCAP-14040-NP-A position of Section 2.7 and correspondence from CE that upgraded the original instruction manual in conformance with ASME Code requirements.*

NRC Request No. 9

Position 1.1 and the chemistry factor tables of Regulatory Guide (RG) 1.99, Revision 2, are currently being used to determine the limiting 1/4t and 3/4t RT_{NDT} values used in the P-T limits for DCPP. Additional RV surveillance capsules will be removed from the DCPP RVs in accordance with the RV material surveillance programs for DCPP Units 1 and 2. Describe the controls in the PTLR process that would administratively mandate PG&E to change the basis for calculating the chemistry factors and RT_{NDT} values for the limiting 1/4t and 3/4t materials to Position 2.1 of RG 1.99, Revision 2 (i.e., use of surveillance data for calculating the chemistry factors and RT_{NDT} values).

PG&E Response

Section 5 of PTLR-1 (included as Enclosure 4 to PG&E Letter DCL-02-079, dated July 31, 2002) describes the five criteria from RG 1.99 utilized to determine the credibility of the surveillance capsule data. The discussion under Criterion 3 (page 16 of PTLR-1) states "Should the credibility criteria be met upon future surveillance capsule withdrawal and evaluation, then Reg. Guide 1.22, Rev. 2, Position C.2 may be utilized." (Note RG 1.99 Position 2.1 is included in Regulatory Position C.2, "Surveillance Data Available.") In the next revision of the PTLR, PG&E will replace the word "may" in the above quoted sentence with "shall." Thus, should capsule credibility be restored in the future, PG&E will use RG 1.99 Position 2.1.

Proposed Technical Specification Changes

Remove	Insert
Page 1.1-5	Page 1.1-5
Page 5.0-28	Page 5.0-28

1.1 Definitions (continued)

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, and the power operated relief valve (PORV) lift settings and arming temperature associated with the Low Temperature Overpressurization Protection (LTOP) System, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. Plant operation within these operating limits is addressed in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

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QUADRANT POWER TILT RATIO (QPTR)

QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.

RATED THERMAL POWER (RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3411 MWt for each unit.

REACTOR TRIP SYSTEM (RTS) RESPONSE TIME

The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

SHUTDOWN MARGIN (SDM)

SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and

b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the hot zero power temperatures.

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(continued)

5.6 Reporting Requirements

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

and LTOP

1. NRC Letter from NRC to Gregory M. Rueger dated May 28, 1999

2. The analytical methods used to determine the RCS pressure and temperature limits were developed in accordance with:

10 CFR 50, Appendix G and H
Regulatory Guide 1.99, Revision 2
NUREG-0800, Standard Review Plan Section 5.3.2
Branch Technical Position MTEB 5-2
ASME B&PV Code Section III, Appendix G
ASME B&PV Code, Section XI, Appendix A
WCAP-14040-NP-A, Section 2.2

Replace shaded text with insert 1.

3. LTOP limits (Power Operated Relief Valves (PORV) pressure relief setpoint and LTOP enable temperature) were developed in accordance with:

NUREG-0800, Standard Review Plan Section 5.2.2
Branch Technical Position RSB 5-2
10 CFR 50, Appendix G and H
Regulatory Guide 1.99, Revision 2
Branch Technical Position MTEB 5-2
WCAP-14040-NP-A, Section 2.2

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

5.6.7 Not Used

5.6.8 PAM Report

When a report is required by Condition B or G of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

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5.6.9 Not Used

(continued)

Insert 1 for TS 5.6.6

1. WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves."

2. Chapter 6.0 of WCAP-15958, "Analysis of Capsule V from Pacific Gas and Electric Company Diablo Canyon Unit 1 Reactor Vessel Radiation Surveillance Program."

Revised Technical Specification Pages

1.1 Definitions (continued)

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, and the power operated relief valve (PORV) lift settings and arming temperature associated with the Low Temperature Overpressurization Protection (LTOP) System, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6.
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3411 MWt for each unit.
REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the hot zero power temperatures.

(continued)

5.6 Reporting Requirements

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

b. The analytical methods used to determine the RCS pressure and temperature and LTOP limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP 14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves."

2. Chapter 6.0 of WCAP-15958, "Analysis of Capsule V from Pacific Gas and Electric Company Diablo Canyon Unit 1 Reactor Vessel Radiation Surveillance Program."

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

5.6.7 Not Used

5.6.8 PAM Report

When a report is required by Condition B or G of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.9 Not Used

(continued)