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PG&E Letter DCL-11-023

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20852

Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
Response to Summary of Telephone Conference Call Held on February 28, 2011, Between the U.S. Nuclear Regulatory Commission and Pacific Gas and Electric Company Concerning Responses to Requests for Additional Information for the Diablo Canyon License Renewal Application

Dear Commissioners and Staff:

By Pacific Gas and Electric Company (PG&E) Letter DCL-09-079, "License Renewal Application," dated November 23, 2009, PG&E submitted an application to the U.S. Nuclear Regulatory Commission (NRC) for the renewal of Facility Operating Licenses DPR-80 and DPR-82, for Diablo Canyon Power Plant (DCPP) Units 1 and 2, respectively. The application included the Applicant's Environmental Report - Operating License Renewal Stage.

A telephone conference between the NRC and representatives of PG&E was held on February 28, 2011, to obtain clarification on the applicant's response to request for additional information (RAI) submitted to the NRC in PG&E Letters DCL-10-168, "Response to NRC Letter dated December 20, 2010, Request for Additional Information (Set 37) for the Diablo Canyon License Renewal Application," dated January 7, 2011 and DCL-10-167, "Response to NRC Letter dated December 20, 2010, Request for Additional Information (Set 36) for the Diablo Canyon License Renewal Application," dated January 12, 2011, regarding Time Limited Aging Analysis (TLAA) and aging management programs. The telephone conference call was useful in clarifying the intent of PG&E's responses. PG&E agreed to supplement the previous responses.

In addition, clarifying information was obtained regarding applicable transients for DCPP's Model 93A Reactor Coolant Pump (RCP) casings. These changes are shown in Enclosure 3.

PG&E's supplemental information to the RAI responses for which the staff requested clarification is provided in Enclosure 1. LRA Amendment 42 resulting from the responses is included in Enclosure 2 showing the changed pages with line-in/line-out annotations. PG&E amends commitments in revised LRA Table A4-1, License Renewal Commitments, shown in Enclosure 2.

If you have any questions regarding this response, please contact Mr. Terence L. Grebel, License Renewal Project Manager, at (805) 545-4160.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on March 25, 2011.

Sincerely,

James R. Becker
Site Vice President

tlg/50378804
Enclosures
cc: Diablo Distribution
cc/enc: Elmo E. Collins, NRC Region IV Regional Administrator
Nathanial B. Ferrer, NRC Project Manager, License Renewal
Kimberly J. Green, NRC Project Manager, License Renewal
Fred Lyon, NRC Project Manager, Office of Nuclear Reactor Regulation
Michael S. Peck, NRC Senior Resident Inspector
Alan B. Wang, NRC Project Manager, License Renewal

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RAI 4.1-7 Follow-up

The Staff requested clarification in Diablo Canyon Power Plant (DCPP) Commitment 22 regarding validation of the baffle and former bolt inspection interval under the Reactor Vessel Internals Aging Management Program.

PG&E Response to RAI 4.1-7 Follow-up

As stated in PG&E's response to RAI 4.1-7 in PG&E Letter DCL-10-167, dated January 12, 2011, fatigue of the baffle and former bolts will be managed in accordance with the Reactor Vessel Internals Aging Management Program. The schedule for inspection of the baffle and former bolts will be validated on a plant-specific basis to ensure that it will appropriately manage the design fatigue analysis. See amended license renewal application (LRA) Commitment 22 in Enclosure 2.

RAI 4.3-15 Follow-up

Request 1:
The Staff requested clarification in DCPP Commitment 58 on which guidance will be used to evaluate more limiting components for the effects of the reactor coolant environment.

Request 2:
The staff requested confirmation that DCPP maintains dissolved oxygen (DO) level below 0.05 ppm and has never had a prolonged period of DO greater than 0.05 ppm.

PG&E Response to RAI 4.3-15 Follow-up

Request 1:
In accordance with Commitment 58, if more limiting components are identified during the review of design basis ASME Class 1 component fatigue evaluations, the most limiting component will be evaluated for the effects of the reactor coolant environment on fatigue usage in using the Metal Fatigue of Reactor Coolant Pressure Boundary program. The effect of the reactor coolant environment on DCPP fatigue usage will be evaluated using material-specific guidance presented in NUREG/CR-6583 for carbon

and low alloy steels, NUREG/CR-5704 for stainless steels, and NUREG/CR-6909 for nickel alloys. See amended LRA Commitment 58 in Enclosure 2.

Request 2:
DCPP has never experienced a dissolved oxygen (DO) spike exceeding 0.05 parts per million (ppm) in the reactor coolant system (RCS) during operation. During operation, Unit 1 and Unit 2 remain less than 0.002 ppm DO. Elevated hydrogen levels prevent DO from exceeding 0.002 ppm. The RCS water is sampled three times per week for hydrogen and four times per week for DO during operation.

LRA Amendment 42

LRA Section	RAI
Table A4.1	4.1-7
Table A4.1	4.3-15

Table A4-1 License Renewal Commitments

Item #	Commitment	LRA Section	Implementation Schedule
22	B. For Reactor Vessel Internals: (1) Participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, PG&E will submit an inspection plan for reactor internals to the NRC for review and approval. *PG&E will validate the schedule for inspection of the baffle and former bolts on a plant-specific basis to ensure that it will appropriately manage the design fatigue analysis.*	4.3.3	Concurrent with industry initiatives and upon completion submit an inspection plan and not less than 24 months before entering the period of extended operation.
58	PG&E will perform a review of design basis ASME Class 1 component fatigue evaluations to determine whether the NUREG/CR-6260-based components that have been evaluated for the effects of the reactor coolant environment on fatigue usage are the limiting components for the DCPP plant configuration. If more limiting components are identified, the most limiting component will be evaluated for the effects of the reactor coolant environment on fatigue usage. *The effect of the reactor coolant environment on DCPP fatigue usage will be evaluated using material-specific guidance presented in NUREG/CR-6583 for carbon and low alloy steels, NUREG/CR-5704 for stainless steels, and NUREG/CR-6909 for nickel alloys.* This additional evaluation will be performed through the Metal Fatigue of Reactor Coolant Pressure Boundary Program in accordance with 10 CFR 54.21 (c)(1)(iii).	4.3.4	Prior to the period of extended operation

Summary of Table Revisions

PG&E Letter DCL-10-168, dated January 7, 2011, provided Request for Additional Information (RAI) responses pertaining to the DCPP License Renewal Application. As part of the response to RAI 4.3-1(follow-up), PG&E provided a table of transients. One of the transient sources was WCAP-13045 Flaw Growth Analysis.

Since the submittal of PG&E letter DCL-10-168, Westinghouse has provided a supplemental letter containing clarification between the transients in relation to generic Reactor Coolant Pump Casing (RCP) Model 93 used in WCAP-13405 and DCPP's Model 93A. The revised Table 1 shows these changed transients with line-in/line-out annotations.

Table 1

Transient	LBB Analysis	Auxiliary Feedwater Line 567 Analysis	WCAP-13045 Flaw Growth Analysis	60-Year Projections (Unit 1/Unit 2)
Normal Conditions				
RCS heatup and cooldown at ≤100°F/hr	200	250	200	85 / 65
Unit loading and unloading at 5% of full power/min	18,300	Not Included	Not Included	Not Projected
Step increase and decrease of 10% of full power	2,000	Not Included	~~2,000~~*Not Included*	56 / 61
Large step load decrease	200	Not Included	~~200~~*Not Included*	11 / 9
Steady state fluctuations	10^6	Not Included	~~3,150,000~~*Not Included*	Not Projected
Upset Conditions				
Loss of load (above 15% full power), without immediate turbine or reactor trip	80	Not Included	80	18 / 10
Loss of all offsite power	40	Not Included	40	2 / 3
Partial loss of flow	80	Not Included	~~Not Included~~*80*	3 / 8
Reactor trip from full power	400	Not Included	~~380~~*Not Included*	100 / 83
Inadvertent RCS depressurization	Not Included	Not Included	~~20~~*Not Included*	3 / 3
Control rod drop	Not Included	Not Included	~~80~~*Not Included*	5 / 2
Test Conditions				
Turbine roll test	10	Not Included	~~Not Included~~*20*	8 / 9
Primary side hydrostatic test	5	Not Included	Not Included	2 / 2
Primary side leak test	50	Not Included	~~Not Included~~*200*	5 / 5
Cold hydrostatic test	10	Not Included	10	Not Projected
Faulted Conditions				
7.5M Hosgri earthquake	Not Included	5	Not Included	1 / 1
Emergency Conditions				
Complete loss of flow	Not Included	Not Included	~~5~~*Not Included*	1 / 1