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PG&E Letter DCL-10-168

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
Response to NRC Letter dated December 20, 2010, Request for Additional Information (Set 37) for the Diablo Canyon License Renewal Application

Dear Commissioners and Staff:

By letter dated November 23, 2009, Pacific Gas and Electric Company (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 license renewal application (LRA), and Applicant's Environmental Report - Operating License Renewal Stage.

By letter dated December 20, 2010, the NRC staff requested additional information needed to continue their review of the DCPP LRA.

PG&E's response to the request for additional information is included in Enclosure 1. LRA Amendment 35 resulting from the responses is included in Enclosure 2 showing the changed pages with line-in/line-out annotations.

PG&E makes a new commitment in amended 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 January 07, 2011.

Sincerely,

James R. Becker
Site Vice President

TLG/50366246
Enclosures
cc: Diablo Distribution
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Nathanial B. Ferrer, NRC Project Manager, License Renewal
Kimberly J. Green, NRC Project Manager, License Renewal
Michael S. Peck, NRC Senior Resident Inspector
Alan B. Wang, NRC Licensing Project Manager

**PG&E Response to NRC Letter dated December 20, 2010
Request for Additional Information (Set 37) for the
Diablo Canyon License Renewal Application**

RAI 4.3-1 (follow-up)

Background:

License renewal application (LRA) Section 4.7.5 indicates that the cycle counting activities of the Metal Fatigue of Reactor Coolant Pressure Boundary Program is the basis for the applicant to disposition its Time-Limited Aging Analysis (TLAA) on the ASME Section XI supplemental fatigue flaw growth analysis for Unit 2 auxiliary feedwater line 567 in accordance with 10 CFR 54.21 (c)(1)(iii).

LRA Section 4.3.2.12 indicates that the cycle counting activities of the Metal Fatigue of Reactor Coolant Pressure Boundary Program is the basis for the applicant to disposition its TLAA on the leak-before break analysis (LBB) in accordance with 10 CFR 54.21 (c)(1)(iii). In its September 22, 2010, response to request for additional information (RAI) 4.3-1, request 1, the applicant indicated that cycle counting of design basis transients against the LBB is not currently accounted for in either the Final Safety Analysis Report (FSAR) or the plant's procedure, but that this type of activity has been accounted for as an enhancement in Commitment No. 21.

Issue:

The U.S. Nuclear Regulatory Commission (NRC or the staff) has noted that the proposal to use of the cycle counting activities of Metal Fatigue of Reactor Coolant Pressure Boundary Program for 10 CFR 54.21 (c)(1)(iii) disposition of the TLAAs on the ASME Section XI supplemental fatigue flaw growth analysis for auxiliary feedwater line 567, the Diablo Canyon Nuclear Power Plant (DCPP) LBB analysis, and the generic fatigue flaw analysis in WCAP-13045 (in support of ASME Code Case N-481 alternative examinations for reactor coolant pump [RCP] casings) is not accounted for in LRA Commitment No. 21 or in the Metal Fatigue of Reactor Coolant Pressure Boundary Program. The staff noted that the use of cycle counting for these analyses does not appear to be accounted for in TS 5.5.5, FSAR Section 5, the plants cycle counting procedure, or the plant's quality assurance procedures.

Request:

Part 1:
Justify your use of cycle counting activities from the Metal Fatigue of Reactor Coolant Pressure Boundary Program to disposition the TLAA for these non-cumulative usage factor (CUF) type of fatigue flaw growth or cycle dependent fracture mechanics analyses (including the LBB, the ASME Section XI fatigue flaw growth analysis for auxiliary feedwater line 567, and the generic fatigue flaw growth analysis in WCAP-

13045) in accordance with 10 CFR 54.221 (c)(1)(iii) when it is not accounted for in either the current licensing basis (CLB), the Metal Fatigue of Reactor Coolant Pressure Boundary Program, or in LRA Commitment No. 21.

Part 2:
Justify why the Metal Fatigue of Reactor Coolant Pressure Boundary Program does not include exceptions or enhancements that: (1) justify the use of cycle counting activities for these types of analyses, (2) defines the transients that would be monitored for when implementing the counting activities against these types of analyses, (3) establishes the action limit would need to be defined on the cycle counting activities when made and established in relation to the transients that are defined and analyzed for in these non-CUF fatigue analyses, and (4) defines the corrective actions that will be taken if this action limit on the given analysis is reached, including the need to perform the analysis and submit it for NRC review and approval if prior NRC approval was necessary for implementation of the original analysis in the CLB.

Part 3:
Justify why TS 5.5.5 or the FSAR, would not need to be amended to account for cycle counting against these types of non-CUF or non-usage factor fatigue analyses.

PG&E Response to D-RAI 4.3-1 (follow-up)

Part 1:
PG&E will revise the DCPP FSAR to include the transients and numbers of events related to the Leak Before Break (LBB) analysis, the ASME Section XI fatigue flaw growth analysis for auxiliary feedwater (AFW) line 567, and the generic fatigue flaw growth analysis in WCAP-13045. See amended LRA Table A4-1 in Enclosure 2.

Part 2:
(1) The scope of the Metal Fatigue of Reactor Coolant Pressure Boundary Program, as described in Element 1 of Section B3.1, has been amended to include the LBB analysis, the ASME Section XI fatigue flaw growth analysis for AFW line 567, and the generic fatigue flaw growth analysis in WCAP-13045.
(2) The transients monitored by the Metal Fatigue of Reactor Coolant Pressure Boundary Program, as described in Element 3 of Section B3.1, has been enhanced to include the analyzed transients provided in Table 1 below.
(3) The action limits in the Metal Fatigue of Reactor Coolant Pressure Boundary Program, as described in Element 6 of Section B3.1, has been amended to include limits for the analyzed transients.
(4) The corrective actions in the Metal Fatigue of Reactor Coolant Pressure Boundary Program, as described in Element 7 of Section B3.1, has been amended to address the LBB analysis, the ASME Section XI fatigue flaw growth analysis for AFW line 567, and the generic fatigue flaw growth analysis in WCAP-13045. The implementing plant procedure will include the corrective action of reanalyzing the LBB analysis, the ASME Section XI fatigue flaw growth analysis for AFW line 567,

and the generic fatigue flaw growth analysis in WCAP-13045 consistent with or reconciled to the original submitted analysis. The reanalysis will receive the same level of regulatory review as the original analysis.

See amended LRA Sections A2.1, B3.1, 4.3.1.1, and Table A4-1 in Enclosure 2.

Part 3:
PG&E will revise the DCPP FSAR to include the transients and numbers of events related to these analyses. See amended LRA Table A4-1 in Enclosure 2.

Table 1 below presents the current cycles used in the LBB analysis, the ASME Section XI fatigue flaw growth analysis for AFW line 567, and the generic fatigue flaw growth analysis in WCAP-13045. It also presents the 60-year projected cycles as shown in LRA Table 4.3-2.

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	56 / 61
Large step load decrease	200	Not Included	200	11 / 9
Steady state fluctuations	10^6	Not Included	3,150,000	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	3 / 8
Reactor trip from full power	400	Not Included	380	100 / 83
Inadvertent RCS depressurization	Not Included	Not Included	20	3 / 3
Control rod drop	Not Included	Not Included	80	5 / 2
Test Conditions				
Turbine roll test	10	Not Included	Not Included	8 / 9
Primary side hydrostatic test	5	Not Included	Not Included	2 / 2
Primary side leak test	50	Not Included	Not Included	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	1 / 1

RAI 4.3-4 (follow-up)

Background:

In the applicant's response to RAI 4.3-4, request 3, dated September 22, 2010, the applicant clarified that the "Auxiliary Spray during Cooldown" transient is within the scope of the Metal Fatigue of Reactor Coolant Pressure Boundary Program.
Issue:

The staff noted that the applicant's response only states that the "Auxiliary Spray during Cooldown" transient was within the scope of the Metal Fatigue of Reactor Coolant Pressure Boundary Program. The response does not justify why the transient was omitted from the scope of LRA Table 4.3-2. As a result, the staff is unable to determine whether or not the "Auxiliary Spray during Cooldown" transient would be projected to exceed the number of occurrences assumed for the transient prior to reaching the end of the period of extended operation. If this transient is within the scope of this AMP, then LRA Table 4.3-2 needs to include applicable projection bases for this transient.

Request:

If the "Auxiliary Spray during Cooldown" transient is an additional transient that is within the scope of the Metal Fatigue of Reactor Coolant Pressure Boundary Program, provide the LRA Table 4.3-2 "Design Basis Cycles," "Limiting Analyzed Value;" Unit 1 "Events (1984-2008)" and "Projected Events for 60-Years;" Unit 1 "Events (1984-2008)," and "Projected Events for 60-Years" values for the "Auxiliary Spray at Cooldown" transient.

PG&E Response to RAI 4.3-4 (follow-up)

As shown in Diablo Canyon Power Plant (DCPP) License Renewal Application (LRA) Amendment 12 (Enclosure 2 of PG&E Letter DCL-10-121, dated September 22, 2010), LRA Table 4.3-2 was amended to provide the following information:

Transient Description	Design Basis Cycles, FSAR Table 5.2-4	Limiting Analyzed Value [iii]	Unit 1		Unit 2	
			Events (1985-2008)	Projected Events for 60-Years	Events (1985-2008)	Projected Events for 60-Years
20. Auxiliary Spray during Plant Cooldown	NS	NS	78	146	54	102

As shown in LRA Table 4.3-2 footnote “a” (page 4.3-11), NS means “not stated,” “not specifically stated,” or “not applicable to this component.” The “Auxiliary Spray during Plant Cooldown” transient is not a transient that is included in the DCPP design or licensing basis (i.e., it is not in Final Safety Analysis Report Table 5.2-4 nor is it used in design analyses). PG&E determined it was prudent to monitor this transient based on industry experience for Westinghouse plants.

<u>RAI 4.3-5 (follow-up)</u>

<u>Background</u>:

In the applicant's response to RAI 4.3-5, request 2, dated September 22,2010, the applicant provide cycle data, longer term rate and weighting factor value data, and short term rate and weighting factor value data for five specific charging system transients in order to justify the applicant's weighted 60-year projection basis for these transient. However, in the applicant's response to RAI 4.3-4, request 2 (as provided in the same Pacific Gas & Electric (PG&E) letter), the applicant clarified how the number of reactor trips and a safety of factor (SF) of 2.15 were used to estimate and derive the number of times these transients had occurred in the past when the transients were not monitored. This request is applicable to the following charging system transients: (1) LRA Table 4.3-2 Transient 15, "charging and letdown, flow shutoff and return to service;" (2) LRA Table 4.3-2 Transient 16, "loss of charging with prompt return to service;" (3) LRA Table 4.3-2 Transient 17, "loss of charging with delayed return to service;" (4) LRA Table 4.3-2 Transient 18, "loss of letdown with prompt return to service;" and (5) LRA Table 4.3-2 Transient 19, "loss of letdown with delayed return to service."

<u>Issue</u>:

The staff seeks confirmation that the cycle numbers that were given for these transients in the response to RAI4.3-5, request 2, incorporate the 2.15 SF-based estimates for the transients that were discussed in the applicant's response to RAI 4.3-4, request 2.

<u>Request</u>:

Confirm that the cycle numbers given for the five charging system transients in response to RAI 4.3-5, request 2, include the 2.15 SF-based estimates for the transients when the transients were not monitored. For these transients, clarify what percentage of the cycle numbers given for the transients are based on the estimates for the periods the transients were unmonitored.

<u>PG&E Response to RAI 4.3-5 (follow-up)</u>

A safety factor of 2.15 was applied to the 5 charging system transients in determining the number of transients that occurred during the years when no monitoring was performed. The following table shows the number of cycles estimated versus the number counted for each unit. PG&E's response to RAI 4.3-5, Request 2 describes the estimation process for each transient.

Unit 1 Number of Cycles				Transient	Unit 2 Number of Cycles			
Actual Counted	Estimated	Total	Percentage *		Actual Counted	Estimated	Total	Percentage *
0	3	3	100%	Chrg & Ltdn	1	3	4	75%
0	3	3	100%	Chrg Delayed	0	3	3	100%
11	53	64	83%	Chrg Prompt	9	45	54	83%
0	6	6	100%	Ltdn Delayed	1	5	6	83%
13	47	60	78%	Ltdn Prompt	5	40	45	89%

* Provides what percentage of the cycle numbers given for the transients are based on the estimates for the periods the transients were unmonitored.

<u>RAI 4.3-10 (follow-up)</u>

<u>Background</u>:

The applicant includes its TLAAs for reactor coolant pressure boundary components in LRA Section 4.3.2 and for the reactor vessel internal (RVI) core support structure components in LRA Section 4.3.3.

By letter dated August 25,2010, the staff issued RAI 4.3-10, request 2, requesting that the applicant provide a basis for why it is acceptable to use cycle-based monitoring of the transients associated with the lower support plates, lower support columns, core barrel nozzles, and lower supports as a bounding basis for non-monitored RVI components with CUF values. In its response dated September 22, 2010, the applicant stated that a fundamental basis for the Metal Fatigue of Reactor Coolant Pressure Boundary Program is that as long as the number of transients used in the analysis remain below the analyzed value, then it has been demonstrated that the components are less than the code allowable value, and structural integrity is demonstrated. The applicant also stated that all transients included in the design basis for the lower support plates, lower support columns, and core barrel nozzles are either: (1) counted when the actual transient cycle is experienced by the plant, or (2) determined that the transient used in the design basis does not need to be counted, based on the following response:

> *This transient is associated with load following operation. The current operating strategy for the DCPP units is continuous base-load power generation. Therefore, the actual number of unit loading/unloading occurrences is expected to be a small fraction of the cycles assumed in the fatigue analyses. Due to the infrequent nature of this cyclic transient, and the large margin to the assumed number of occurrences, it is not necessary to track its occurrence.*

The applicant also made similar responses for the unit loading and unloading transients, and for the steady state fluctuations transient in its responses to other RAIs in letter of September 22, 2010, including the response to RAI 4.3-1, request 2; RAI4.3-8, and RAI4.3-9. However, DCPP Administrative Control Technical Specification (TS) 5.5.5, which requires administrative performance the following design basis transient monitoring activities:

> *5.5.5 Component Cyclic or Transient Limit*
>
> *This program provides controls to track the FSAR, Section 5.2 and 5.3, cyclic and transient occurrences to ensure that components are maintained within the design limits.*

FSAR Table 5.2-4 does not exempt the unit loading and unloading at 5 percent power per minute transients or the steady state fluctuations transient to be exempted from the cycle counting requirements in the same manner that FSAR Table 5.24 exempts the plant's faulted condition transients from the scope of the TS 5.5.5 monitoring *requirements, or in the manner the FSAR table was updated in FSAR Revision 19 to exempt the "T_{avg} Coastdown from Nominal to Reduced Temperature" transient from the counting requirements.*

<u>Issue</u>:

FSAR Table 5.24 requires that the unit load and unloading at 5 percent power per minute transients and the steady state fluctuation transient be monitored under the Metal Fatigue of Reactor Coolant Pressure Boundary Program's cycle monitoring requirements.

The staff noted that that the applicant's basis for stating that it does not need to do further tracking of the unit load and unload at 5 percent power per minute transients or the steady state fluctuations transient is not consistent with the CLB as described in TS 5.5.5 or the design basis transients in FSAR Table 5.2-4.

<u>Request</u>:

Clarify whether FSAR Table 5.2-4 currently exempts the unit loading and unloading at 5 percent power per minute transients from the design basis transients and cycle monitoring requirements of TS 5.5.5. Provide your basis why controls to monitor for unit loading and unloading at 5 percent power per minute transients do not need implemented for the period of extended operation consistent with FSAR Table 5.2-4.

<u>PG&E Response to RAI 4.3-10 (follow-up)</u>

The current implementing procedure for Technical Specification 5.5.5 documents the basis for excluding transients associated with unit loading and unloading. It currently states:

> "The number of occurrences listed in the FSAR table is 18,300. Over a 50-year design life, this equates to one cycle per day, every day. The current operating strategy for the DCPP units is continuous Base Load power generation. Therefore, the actual number of occurrences is expected to be a small fraction of the cycles assumed in the fatigue analyses (e.g., at 50 cycles per year, for 50 years would result in less than 15% of the assumed cycles). Due to the infrequent nature of this cyclic transient, and the huge margin to the assumed number of occurrences, data sheets will not be completed."

Through the period of extended operation, less than 17 percent of the 18,300 cycles will occur if 50 cycles per year for 60 years are assumed.

The Diablo Canyon Power Plant Final Safety Analysis Report will be revised to note the basis for exclusion of these transients from counting. See amended License Renewal Application Table A4-1 in Enclosure 2.

RAI 4.3-12 (follow-up)

Background:

In its September 22, 2010, response to RAI 4.3-12, request 2, the applicant provided an acceptable basis for not including aging management review (AMR) items on cumulative fatigue damage for HVAC systems because these systems were not designed to ASME Code Section III requirements for Class 2 or 3 components or to ANSI 831.1 or 831.7 requirements. The applicant also stated that the remaining piping systems listed in the RAI are designed to ASME Class 2, 3, or ANSI 831.1 piping requirements, are within the scope of license renewal, and are subject to cumulative fatigue damage through the application of a stress range reduction factor. PG&E has evaluated the above list of piping systems in LRA Section 4.3.5. However, the applicant also stated that the inclusion of the relevant AMR items on cumulative fatigue damage in their corresponding Table 2 AMR tables would only make reference to LRA Chapter 4.0 for the disposition through the inclusion of the phrase "Time Limited Aging Analysis evaluated for the period of extended operation" consistent with those that were including for other Generic Aging Lessons Learned (GALL) AMR items on cumulative fatigue damage.

Issue:

The staff noted that the applicant's response to RAI4.3-12, request 2, clearly identifies cumulative fatigue damage as an applicable aging effect for either ASME Code Section III Class 2 or 3 or ANSI 831.1 designed piping, piping components or piping elements in the following ESF, AUX and SPC subsystems: (1) containment spray system; (2) all Table 2 AMR Tables for non-HVAC AUX subsystems in LRA Section 3.3 other than those that were provided for in LRA Table 3.3.2-8, Chemical and Volume Control System; (3) auxiliary steam system; and (4) condensate system. However, the staff noted that the applicant did not amend the LRA to include the applicable AMR line items on cumulative fatigue damage for the applicable piping, piping components, and pipe elements in the applicant's auxiliary steam and condensate systems, to conform with the recommendations of NEI 95-10, Revision 6 for inclusion of the appropriate AMR line item for these systems.

Request:

Justify the basis for omitting the relevant AMR items on cumulative fatigue damage if cumulative fatigue damage is a relevant aging effect requiring management for the following applicable piping, piping components, or piping elements that was designed to either ASME Section 1/1 requirements for Class 2 or 3 components or to ANSI 831.1 design requirements in the following subsystems: (1) containment spray system; (2) associated with all Table 2 AMR Tables for non-HVAC AUX subsystems in LRA Section 3.3, other than those that were appropriately provided for in LRA Table 3.3.2-8, *Chemical and Volume Control System; (3) auxiliary steam system; and (4) condensate system.*

PG&E Response to RAI 4.3-12 (follow-up)

License Renewal Application (LRA) Section 4.3.5 describes the review that was completed of the plant piping systems to identify components that might be subject to cumulative fatigue damage based on temperatures screening criteria. Only those subsystems with temperatures in excess of 220°F for carbon steel and in excess of 270°F for stainless steel need to be included as aging management review items. The review determined the:

(1) The containment spray system does not exceed the temperature screening criteria.

(2) The only subsystems in LRA Section 3.2 3, Auxiliary Systems, that exceed the temperature screening criteria are:
- Nuclear Steam Supply Sampling System
- Diesel Generator System

(3) The Auxiliary Steam System does exceed the temperature screening criteria.

(4) The Condensate System does not exceed the temperature screening criteria.

The systems that exceed the temperature screening criteria have been added to the LRA. See amended LRA Tables 3.3.2-6, 3.3.2-14, and 3.4.2-2 in Enclosure 2.

RAI 4.3-13

Background:

LRA Section 4.3.2.2 provides the applicant's basis for dispositioning the CUF TLAAs for the upper reactor vessel (RV) closure heads and their control rod drive mechanism (CRDM) nozzle and control element thermocouple nozzle assembly (CETNA) components in accordance with the TLAA acceptance criterion in 10 CFR 54.21 (c)(1)(i).

A TLAA may be dispositioned pursuant to the TLAA acceptance criterion in 10 CFR 54.21 (c)(1)(i) only if it can be demonstrated that the existing analysis for the TLAA will be valid for the period of extended operation.

Issue:

Based on its review of LRA Section 4.3.2.2, the staff has determined that the applicant is using 10 CFR 54.21 (c)(1)(i) as the basis for dispositioning the CUF values for the 2009 replaced DCPP Unit 2 upper RV closure head components, and its CRDM and CETNA nozzle components without providing any supporting CUF values in the LRA to demonstrate continued validity of the CUF values for the period of extended operation. Thus, for these components, the LRA does not provide an adequate demonstration that the new CUF values of record for these components are all less than or equal to a CUF design limit value of 1.0.

The staff has determined that the applicant is also using 10 CFR 54.21 (c)(1)(i) as the basis for dispositioning the CUF values for the DCPP Unit 1 upper RV closure head components, and its CRDM and CETNA nozzle components. However, the staff has noted that, for these components, the applicant is applying the 10 CFR 54.21 (c)(1)(i) acceptance criterion on planned replacement of the Unit 1 RV closure head components, and thus on the CUF values that would presumably be calculated in the future in support of the head replacement activities. Thus, for these components, the applicant appears to be relying on 10 CFR 54.21 (c)(1)(i) based on CUF values that currently do not exist in the CLB for Unit 1, and there is not any way for the staff to verify that the new CUF values for this will all be less than or equal to a CUF design limit value of 1.0.

Thus, the staff cannot verify the validity of using 10 CFR 54.21(c)(1)(i) as the basis for accepting these CUF values because either: (1) the applicant did not include the CUF values for the components in the LRA, or (2) the applicant is relying on 10 CFR 54.21 (c)(1)(i) acceptance based on CUF values that do not currently exist in the CLB.

Request 1:

Provide the CUF values of record for the Unit 2 replacement upper RV closure head and its CETNA and CRDM penetration nozzle components. Alternatively, provide *justification for not providing the 2009 CUF values for these Unit 2 components and for dispositioning the TLAA for these components in accordance with 10 CFR 54.21(c){1)(i) without docketing the CUF values for the components in the LRA during the LRA review period.*

Request 2:

Provide the CLB CUF values for the Unit 1 upper RV closure heads and its CETNA and CRDM penetration nozzles that will be in place during the period of extended operation, such that the NRC can determine the appropriateness of the applicant's basis for dispositioning the CUF values for these Unit 1 components in accordance with 10 CFR 54.21(c)(1)(i).

PG&E Response to RAI 4.3-13

Request 1:

Table 1 below displays the cumulative usage factor (CUF) values of record for the 2009 replaced Diablo Canyon Power Plant (DCPP) Unit 2 upper reactor vessel closure head (RVCH) and its core exit thermocouple nozzle assembly (CETNA) and control rod drive mechanism (CRDM) penetration nozzle components. These CUF values demonstrate continued validity of the CUF values for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

Table 1

Unit 2 Component	CUF Value
RVCH	0.292
CRDM	0.297
CETNA	0.3792

Request 2:

Table 2 below displays the current licensing basis CUF values for the DCPP Unit 1 upper RVCH and its CETNA and CRDM penetration nozzles that will be in place during the period of extended operation. These CUF values demonstrate continued validity of the CUF values for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

Table 2

Unit 1 Component	CUF Value
RVCH	0.292
CRDM	0.297
CETNA	0.3792

RAI 4.3-14

<u>Background</u>: The applicant includes its TLAAs for the RVI core support structure components in LRA Section 4.3.3. LRA Section 4.3.3 includes the subsections for the RVI upper core plates and lower core plates and the applicant dispositions the CUF analyses for these RVI core support structure components in accordance with 10 CFR 54.21 (c)(1)(iii). Furthermore, the applicant's cycle counting activities, as part of its Metal Fatigue of Reactor Coolant Pressure Boundary Program, will verify the number of cycles for the transients, in the updated CUF analyses for these components, is bounded by cycle limits for these transients in the original design basis.

The following is provided in LRA Section 4.3.3 (LRA page 4.3-41):

> *The numbers of transients used in the analysis are bound by the numbers of transients in the current 50-year design basis.*

<u>Issue</u>:

The staff is not able to determine whether the reference to the words "are bound by" means that the number of assumed cycles for the transient analyzed in the updated CUF analyses for the upper core plates and lower core plates are greater than or equal to the existing limits on cycles for these transients in the design basis or less than or equal to the existing limits on cycles for these transients in the design basis. With respect to the updated CUF analyses for these components, the staff is not able to determine whether the cycle counting activities of the Metal Fatigue of Reactor Coolant Pressure Boundary Program should be associated with the number of cycles that were assumed for these transients in the updated CUF calculations for upper core plates and lower core plates or should be associated with the number of cycles that were assumed in the design basis for these transients.

<u>Request</u>:

Clarify whether the cycle counting activities of the Metal Fatigue of Reactor Coolant Pressure Boundary Program are associated with the number of cycles that were assumed for these transients in the updated CUF calculations for upper core plates and lower core plates or should be associated with the number of cycles that were assumed in the design basis for these transients as defined in FSAR Table 5.2-4.

<u>PG&E Response to RAI 4.3-14</u>

The number of cycles assumed in the updated cumulative usage factor (CUF) analyses for the upper core plates and lower core plates are greater than or equal to the number of cycles in the current 50-year design basis. As stated in License Renewal Application Section 4.3.3 (page 4.3-41), the Metal Fatigue of Reactor Coolant Pressure Boundary Program will monitor the 50-year design basis number of transients, as defined in Final Safety Analysis Report Table 5.2-4, to ensure that the updated CUF analyses for the upper core plates and lower core plates will remain valid for the period of extended operation.

RAI 4.3-15

The applicant includes its environmentally-assisted metal fatigue analyses for specific reactor coolant pressure boundary (RCPS) components in LRA Section 4.3.4. The applicant includes the following seven components in its environmentally-assisted fatigue analysis calculations in conformance with the NUREG/CR-6260 recommendations:

1. *RV shell to lower head juncture.*
2. *RV inlet nozzles*
3. *RV outlet nozzles*
4. *Pressurizer surge lines (i.e., pressurizer surge line nozzle to the hot leg)*
5. *Charging line nozzles*
6. *Safety Injection nozzles*
7. *Residual Heat Removal (RHR) line tee*

The locations selected by the applicant are consistent with the recommended locations for pressurized water reactor (PWR) designs in Table 5-98 of NUREG/CR-6260 for older vintage Westinghouse designed nuclear power plants, which is consistent with Standard Review Plan - License Renewal (SRP-LR) Sections 4.3.1.2 and 4.3.2.2.

In LRA Section 4.3.4, the applicant identifies that the F_{en} *adjustment factors in LRA Tables 4.3-8 and 4.3-9 are based, in part, on assumed dissolved oxygen content for the reactor coolant system (RCS) coolant of less than 0.05 ppb dissolved oxygen contents. In LRA Section 4.3.4, the applicant also identifies that the* F_{en} *adjustment factors that were used for the recalculations of the environmental CUF values for the charging system nozzles, safety injection nozzles, and surge line nozzles in LRA Table 4.3-9 were based on the strain rate methodology in Materials Reliability Program (MRP) Report No. MRP-47, and that the revised* F_{en} *adjustment factors for these components were derived from the report using the actual stresses from the load pairs for the limiting design transients that were applicable to these nozzle components.*

Issue 1:

In LRA Table 4.3-3, for RV components, and LRA Table 4.3-S. for Class 1 pressurizer components, the applicant reported that some of the RV and pressurizer components had either 40-year design basis CUFs or 50-year projected CUFs that were greater than those used for the corresponding pressurizer or RV locations selected in the applicant environmentally-assisted fatigue analysis evaluation:

- *Pressurizer spray nozzles -Unit 1 is the limiting unit with a 50-year design basis CUF value of 0.947 and a 50-year projected CUF of 1.135 for its spray nozzles*
- *Pressurizer heat penetration nozzles -unit 1 is the limiting unit 50-year design basis CUF value of 2.954 and a updated 50-year projected CUF of 0.940*

- *RV bottom mounted instrumentation nozzles, which are nickel alloy RCPS component locations - with a 50-year design basis CUF value of 0.378 and a 50-year projected CUF of 0.454*

However, the staff noted that the applicant did not include these component locations in the environmentally-assisted fatigue calculations.

The staff is concerned whether additional components (beyond those of NUREG/CR-6260) needed to be considered for environmental effects of reactor water on the CUF, consistent with the SRP and GALL guidance to consider environmental effects for the NUREG/CR-6260 locations "at a minimum" (see SRP-LR Sections 4.3.2.2 & 4.3.3.2 and Item 5 of GALL Section X.M1).

Request 1:

Clarify whether any additional RCPS components were considered for inclusion in the environmentally-assisted fatigue analyses beyond those assessed in LRA Tables 4.3-8 and 4.3-9.

If there were other components considered, justify why these additional RCPS components were not included within the scope of those components that were selected for environmentally-assisted fatigue analyses.

If other components were not considered, justify why additional RCPS components, beyond those in NUREG/CR-6260, were not considered for environmental effects of reactor water on the CUF, consistent with the recommendations in the GALL Report and SRP-LR, based on the magnitude of the design basis or 60-year projected CUF when compared to those locations selected for the environmentally-assisted fatigue analysis in LRA Tables 4.3-8 and 4.3-9.

Issue 2:

LRA Tables 4.3-8 and 4.3-9 indicate that the applicant's environmentally-assisted metal fatigue analysis locations include both low alloy steel components (the topic of NUREG/CR-6583) and stainless steel components (the topic of NUREG/CR-5704).

The applicant discusses the assumed dissolved oxygen (DO) content of less than 0.05 ppm DO for the derivation of F_{en} factors for stainless steel reactor coolant pressure boundary (RCPS) components; however, the staff is unclear regarding the assumed DO content for the derivation of F_{en} factors for the low alloy steel components.

Request 2:

Discuss and provide justification for the assumed DO concentration used in the derivation of F_{en} factors for the low alloy steel RCPS components that were evaluated *for environmentally-assisted fatigue effects. Justify why a F_{en} factor of 2.46 is considered to be conservative for these low alloy steel component locations.*

Issue 3:

In LRA Section 4.3.4, the applicant identified that the F_{en} factors for the stainless steel safety injection (SI) nozzles, charging nozzles, and hot leg surge nozzle safe ends were recalculated using the strain rate methodology in Materials Reliability Program (MRP) Report No. MRP-47. According to the results reported in LRA Table 4.3-9, application of this methodology resulted in the following changes to the F_{en}-adjusted CUF values for these components:

- *Reduced the F_{en}-adjusted CUF value for the SI nozzles from 48.54 to 0.76*
- *Reduced the F_{en}-adjusted CUF value for the charging nozzles from 1.18 to 0.44*
- *Reduced the F_{en}-adjusted CUF value for the hot leg surge nozzle safe ends from 6.49 to 3.22*

The MRP-47 report is not currently endorsed by the NRC for application to environmentally assisted metal fatigue calculations.

Request 3:

Explain the changes that were made to the assumptions for the updated F_{en}-adjusted CUF calculations for these components. Provide your basis why the application of the MRP-47 methodology is considered capable of yielding sufficiently conservative F_{en}-adjusted CUF values for these component locations and why the updated 60-year F_{en}-adjusted CUF values for these components are considered the representative values for the assessments.

PG&E Response to RAI 4.3-15

Request 1:

No additional reactor coolant pressure boundary components were considered for inclusion in the environmentally-assisted fatigue analyses beyond those assessed in License Renewal Application (LRA) Tables 4.3-8 and 4.3-9.

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 Diablo Canyon Power 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. 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). See amended LRA Table A4-1 in enclosure 2.

<u>Request 2</u>:

For a pressurized water reactor environment, the dissolved oxygen (DO) is less than 0.05 ppm, which corresponds to a O* value of 0. As such, the 3rd term in the exponent in the F_{en} equation is also 0, making the F_{en} equation dependent upon T only. The value of T is taken to be 25°C (77°F), which is the lowest temperature that the components will experience, and therefore results in the maximum F_{en} for the components. For these reasons, a F_{en} factor of 2.46 is considered to be conservative for these low alloy steel component locations.

<u>Request 3</u>:

The safety injection (SI) nozzles, charging nozzles, and hot leg surge nozzle safe ends updated F_{en}-adjusted cumulative usage factor (CUF) calculations was based on the number of transients projected to occur during the extended period (versus the design number of transients listed in FSAR Table 5.2-4). In addition, for the SI and hot leg surge nozzles, the updated F_{en}-adjusted CUF was revised using a F_{en} calculated with the integrated strain rate method described in MRP-47, Revision 1. In this method, the F_{en} factor is computed at multiple points over the increasing (tensile) portion of a paired strain range, and an overall F_{en} is integrated over the entire tensile portion of the strain range (i.e., from the algebraically lowest stress point of the maximum compressive stress event to the algebraically highest stress point of the maximum tensile stress event). MRP-47 uses the same F_{en} equations as those shown in NUREG/CR-5704 for austenitic stainless steels and NUREG/CR-6583 for carbon steels and low-alloy steels. The MRP-47 integrated strain rate approach discussed above is similar to the approach used in NUREG/CR-6909.

MRP-47, Revision 1 provides a technical basis prepared from NRC, NRC contractor, EPRI, and other industry participants to provide a more unified and consistent approach to determining F_{en} values throughout the industry. The basis for the cycle projections is presented in LRA Section 4.3.1. The acceptable use of MRP-47 to perform the F_{en} calculations is presented above. Therefore, the resulting F_{en}-adjusted CUF are acceptable representative values for the assessments.

RAI 4.3-16

Background:

LRA Section 4.3.6 provides the TLAA for the "Fatigue Design and Analysis of Class IE Electrical Raceway Support Angle Fittings for Seismic Events and dispositioned the TLAA in accordance with 10 CFR 54.21 (c)(1)(i)." The applicant stated that the current analysis is based on the number of occurrences that are currently assumed in the design basis for the following design earthquake categories: (1) five occurrences of the plant's design basis earthquake (DE), which is equivalent to the operational basis earthquake (OBE) defined in Appendix A of 10 CFR Part 100; (2) one occurrence of a double design basis earthquake (DDE) which is equivalent to the safe shutdown earthquake (SSE) defined in Appendix A of 10 CFR Part 100; and (3) one offsite 7.5 Richter scale magnitude "Hosgri" earthquake (HE), which is postulated as an seismic event for the offshore "Hosgri" fault.

FSAR Table 5.2-4 assumes the following design basis occurrences for these events: (1) 20 DE occurrences; (2) one DDE occurrence; and (3) one HE occurrence.

Issue:

The staff noted an inconsistency in the value that is reported as the design basis on the number of assumed occurrences of the DE event.

Request:

Explain why there are two different values that are being reported on the number of assumed occurrences for the DE seismic event (i.e., five in LRA Section 4.3.6 versus 20 in FSAR Table 5.2-4). Clarify and provide justification for which value represents the correct value.

PG&E Response to RAI 4.3-16

Reactor coolant pressure boundary (RCPB) components have a limiting value of 20 design earthquake (DE) cycles. This is reflected in Final Safety Analysis Report (FSAR) Table 5.2-4, which is intended to ensure the integrity of the RCPB. The raceway design analyses have a limiting value of 5 DE cycles. The raceways are not part of the RCPB and are, therefore, not covered by the scope of FSAR Table 5.2-4.

The acceptability of the design was documented in Diablo Canyon Power Plant Supplemental Safety Evaluation Reports 18 and 29 for Units 1 and 2, respectively. The number of DE events assumed is greater than the events actually experienced to date and the number projected to occur during 60 years of operation. A special action limit has been placed in the Metal Fatigue of Reactor Coolant Pressure Boundary Program for the raceways. This action limit will be reflected in the program's implementing procedure.

LRA Amendment 35

LRA Section	RAI
Appendix A2.1	4.3-1
Appendix B3.1	4.3-1
Section 4.3.1.1	4.3-1
Table 3.3.2-6	4.3-12
Table 3.3.2-14	4.3-12
Table 3.4.2-2	4.3-12
Table A4-1	4.3-1 4.3-10 4.3-15

A2.1 METAL FATIGUE OF REACTOR COOLANT PRESSURE BOUNDARY

The Metal Fatigue of Reactor Coolant Pressure Boundary program manages fatigue cracking caused by anticipated cyclic strains in metal components of the reactor coolant pressure boundary. The program will ensure that actual plant experience remains bounded by the transients assumed in the design calculations *and fatigue flaw growth analyses*, or that appropriate corrective measures maintain the design and licensing basis by other acceptable means. The Metal Fatigue of Reactor Coolant Pressure Boundary program will track the number of transient cycles and will track cumulative fatigue usage at monitored locations. If a cycle count or cumulative fatigue usage value increases to an action limit, corrective actions will be initiated to evaluate the design limits and determine appropriate specific corrective actions. Action limits permit completion of corrective actions before the design basis number of events is exceeded.

B3.1 METAL FATIGUE OF REACTOR COOLANT PRESSURE BOUNDARY

Enhancements

Prior to the period of extended operation, the following enhancements will be implemented in the following program elements:

Scope of Program – Element 1, Preventive Actions – Element 2, and Monitoring and Trending – Element 5

The scope of locations monitored by the DCPP Metal Fatigue of Reactor Coolant Pressure Boundary program will be enhanced to include additional locations which are not covered by the current Metal Fatigue of Reactor Coolant Pressure Boundary program. Additional locations will include the NUREG/CR-6260 locations for the effects of the reactor coolant environment on fatigue. Usage factors in the NUREG/CR-6260 sample locations will include the environmental factors, F(en), calculated by NUREG/CR-6583 and NUREG/CR-5704 or appropriate alternative methods.

Scope of Program – Element 1 and Parameters Monitored or Inspected – Element 3

The scope of transients monitored by the DCPP Metal Fatigue of Reactor Coolant Pressure Boundary program will be enhanced to include additional transients that contribute to fatigue usage *and those transients used in fatigue flaw growth analyses supporting the leak-before-break analysis, ASME Section XI tolerance evaluations, and relief from ASME Section XI inspections*, which are not covered by the current Metal Fatigue of Reactor Coolant Pressure Boundary program. Usage factors in the NUREG/CR-6260 sample locations will include the environmental factors, F(en), calculated by NUREG/CR-6583 and NUREG/CR-5704 or appropriate alternative methods.

Preventive Actions – Element 2 and Acceptance Criteria – Element 6

The procedures governing the DCPP Metal Fatigue of Reactor Coolant Pressure Boundary program will be enhanced to include additional cycle count and fatigue usage action limits, which will invoke appropriate corrective actions if a component approaches a cycle count action limit or a fatigue usage action limit. Action limits permit completion of corrective actions before the design limits are exceeded.

Cycle Count Action Limits:

An action limit initiates corrective action when the cycle count for any of the critical thermal or pressure transients is projected to reach the action limit defined in the program before the end of the next fuel cycle. In order to assure sufficient margin to accommodate occurrence of a low probability transient, corrective actions must be initiated before the remaining number of allowable cycles for any specified transient becomes less than one. *Action limits will also be established based on the number of transients used in fatigue flaw growth analyses.*

Cumulative Fatigue Usage (CUF) Action Limits:

An action limit requires corrective action when calculated cumulative usage factor (CUF) for any monitored location is projected to reach 1.0 within the next three fuel cycles.

Detection of Aging Effects – Element 4

The procedures governing the DCPP Metal Fatigue of Reactor Coolant Pressure Boundary program will be enhanced to specify the frequency of periodic reviews of the results of the monitored cycle count and cumulative usage factor data at least once per fuel cycle. This review will compare the results against the corrective action limits to determine any approach to action limits and any necessary revisions to the fatigue analyses will be included in the corrective actions.

Corrective Actions – Element 7

The procedures governing the DCPP Metal Fatigue of Reactor Coolant Pressure Boundary program will be enhanced to include appropriate corrective actions to be invoked if a component approaches a cycle count action limit or a fatigue usage action limit. The corrective action options for a component that has exceeded action limits include a revised fatigue analysis or repair or replacement of the component. *Corrective actions for fatigue crack growth analysis action limits include re-analyzing the fatigue crack growth analysis consistent with or reconciled to the originally submitted analysis. The reanalysis will receive the same level of regulatory review as the original analysis.*

4.3.1.1 Enhanced DCPP Fatigue Management Program

Corrective Action Limits and Corrective Actions

The enhanced DCPP Fatigue Management Program provides for evaluation of fatigue usage and cycle count tracking of critical thermal and pressure transients to verify that the ASME Code CUF limit of 1.0 and other *non-*CUF design limits (*e.g.*, *fatigue flaw growth analyses*) will not be exceeded. The program requires this evaluation at least once per fuel cycle.

The enhanced program specifies corrective actions to be implemented to ensure that appropriate reevaluation or other corrective action is initiated if an action limit is reached. Action limits permit completion of corrective actions before the design limits are exceeded.

Cycle Count Action Limits and Corrective Actions

Cycle count action limits have been established based on the design number of cycles. In order to assure sufficient margin to accommodate occurrence of a low probability transient, corrective actions must be taken before the remaining number of allowable cycles for any specified transient, including the low-probability, higher-usage-factor events, becomes less than one. Events which occur more frequently contribute less per event to the usage factor. To account for both cases, corrective actions are required when the cycle count for any of the significant contributors to the usage factor is projected to reach a specified percentage of the design number of cycles before the end of the next fuel cycle. *Action limits will also be established based on the number of transients used in fatigue flaw growth analyses.*

If one of these cycle count action limits is reached, acceptable corrective actions must include the first, and may include others of the following:

1. Review of fatigue usage calculations.
    - To identify the components and analyses affected by the transient in question.
    - To determine whether the transient in question contributes significantly to CUF.
    - To ensure that the analytical bases of the leak-before-break (LBB) fatigue crack propagation analysis is maintained.
    - To ensure that the analytical bases of a fatigue crack growth and stability analysis in support of relief from ASME Section XI flaw removal and inspection requirements for hot leg small-bore half nozzle repairs are maintained.

2. Evaluation of remaining margins on CUF based on the CBF calculations of the DCPP Fatigue Management Program software.

3. Redefinition of the specified number of cycles (e.g., by reducing specified numbers of cycles for other transients and using the margin to increase the allowed number of cycles for the transient that is approaching its specified number of cycles).

4. Redefinition of the transient to remove conservatism in the pressure and temperature ranges.

Corrective actions for fatigue crack growth analysis action limits include reanalyzing the fatigue crack growth analysis consistent with or reconciled to the originally submitted analysis. The reanalysis will receive the same level of regulatory review as the original analysis.

Cumulative Fatigue Usage Action Limits and Corrective Actions

The enhanced program periodically calculates CUFs at the cycle-based fatigue management locations. CUF action limits have been established based on these calculated CUFs. To provide adequate time for corrective actions and adequate margin to permit continued operation, corrective actions are required when calculated CUF for any monitored location is projected to reach 1.0 within the next 3 fuel cycles.

For DCPP locations identified in NUREG/CR-6260, *Effects of the Reactor Coolant System Environment on Fatigue Life of Piping and Components (Generic Safety Issue 190)*, the action limit is based on accrued fatigue usage calculated with the Fen factors required for including effects of the reactor coolant environment.

If the action limit is reached, acceptable corrective actions must include the first, and may include others of the following:

1. Determine whether the scope of the Fatigue Management Program must be enlarged to include additional affected reactor coolant pressure boundary locations. This determination will ensure that other locations do not approach design limits without an appropriate action.

2. Enhance fatigue managing to confirm continued conformance to the code limit.

3. Repair the component.

4. Replace the component.

5. Perform a more rigorous analysis of the component to demonstrate that the design code limit will not be exceeded.

6. Modify DCPP operating practices to reduce the fatigue usage accumulation rate.

Table 3.3.2-6 Auxiliary Systems – Summary of Aging Management Evaluation – Nuclear Steam Supply Sampling System

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
Piping	*PB*	*Stainless Steel*	*Treated Borated Water (Int)*	*Cumulative Fatigue Damage*	*Time Limited Aging Analysis evaluated for the period of extended operation*	*VII.E1-16*	*3.3.1.02*	*A*

Table 3.3.2-14 Auxiliary Systems – Summary of Aging Management Evaluation – Diesel Generator System

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
Piping	*PB*	*Carbon Steel*	*Diesel Exhaust (Int)*	*Cumulative Fatigue Damage*	*Time Limited Aging Analysis evaluated for the period of extended operation*	*VII.E1-18*	*3.3.1.02*	*A*

Table 3.4.2-2 Steam and Power Conversion System – Summary of Aging Management Evaluation – Auxiliary Steam System

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
Piping	*PB*	*Carbon Steel*	*Steam (Int)*	*Cumulative Fatigue Damage*	*Time Limited Aging Analysis evaluated for the period of extended operation*	*VIII.B1-10*	*3.4.1.01*	*A*

Table A4-1 License Renewal Commitments

Item #	Commitment	LRA Section	Implementation Schedule
21	Enhance the Metal Fatigue of Reactor Coolant Pressure Boundary program to: • Include additional locations which are not covered by the current Metal Fatigue of Reactor Coolant Pressure Boundary program. Additional locations will include the NUREG/CR-6260 locations for the effects of the reactor coolant environment on fatigue. Usage factors in the NUREG/CR-6260 sample locations will include the environmental factors, F(en), calculated by NUREG/CR-6583 and NUREG/CR-5704 or appropriate alternative methods, and • Include additional transients that contribute to fatigue usage *and those transients used in fatigue flaw growth analyses supporting the leak-before-break analysis, ASME Section XI tolerance evaluations, and relief from ASME Section XI inspections*, which are not covered by the current Metal Fatigue of Reactor Coolant Pressure Boundary program. Usage factors in the NUREG/CR-6260 sample locations will include the environmental factors, F(en), calculated by NUREG/CR-6583 and NUREG/CR-5704 or appropriate alternative methods, and • Include additional cycle count and fatigue usage action limits, which will invoke appropriate corrective actions if a component approaches a cycle count action limit or a fatigue usage action limit. Action limits permit completion of corrective actions before the design limits are exceeded. Cycle Count Action Limits: An action limit initiates corrective action when the cycle count for any of the critical thermal or pressure transients is projected to reach the action limit defined in the program before the end of the next fuel cycle. In order to assure sufficient margin to accommodate occurrence of a low probability transient, corrective actions must be initiated before the remaining number of allowable cycles for any specified transient becomes less than one. *Action limits will also be established based on the number of transients used in fatigue flaw growth analyses.* Cumulative Fatigue Usage (CUF) Action Limits:	B3.1	Prior to the period of extended operation

Table A4-1 License Renewal Commitments

Item #	Commitment	LRA Section	Implementation Schedule
	An action limit requires corrective action when calculated cumulative usage factor (CUF) for any monitored location is projected to reach 1.0 within the next 3 fuel cycles, and • The procedures governing the DCPP Metal Fatigue of Reactor Coolant Pressure Boundary program will be enhanced to specify the frequency of periodic reviews of the results of the monitored cycle count and cumulative usage factor data at least once per fuel cycle. This review will compare the results against the corrective action limits to determine any approach to action limits and any necessary revisions to the fatigue analyses will be included in the corrective actions, and • The procedures governing the DCPP Metal Fatigue of Reactor Coolant Pressure Boundary program will be enhanced to include appropriate corrective actions to be invoked if a component approaches a cycle count action limit or a fatigue usage action limit. The corrective action options for a component that has exceeded action limits include a revised fatigue analysis or repair or replacement of the component. *Corrective actions for fatigue crack growth analysis action limits include re-analyzing the fatigue crack growth analysis consistent with or reconciled to the originally submitted analysis. The reanalysis will receive the same level of regulatory review as the original analysis.*		
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. 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*

Table A4-1 License Renewal Commitments

Item #	Commitment	LRA Section	Implementation Schedule
59	*PG&E will revise the DCPP FSAR to include the basis for exclusion of unit loading and unloading transients from counting, and the transients and numbers of events related to the leak-before-break analysis, the ASME Section XI fatigue flaw growth analysis for auxiliary feedwater line 567, and the generic fatigue flaw growth analysis in WCAP-13045.*	*B3.1*	*Prior to the period of extended operation*