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PG&E Letter DCL-10-121

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 NRC Letter dated August 25, 2010, Request for Additional Information (Set 19) 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 August 25, 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 12, resulting from the responses, is included in Enclosure 2 showing the changed pages with line-in/line-out annotations.

PG&E makes no regulatory commitments (as defined in NEI 99-04) in this letter.

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.

Sincerely,

James R. Becker

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NRR

pns/50338457
Enclosure
cc: Diablo Distribution
cc/enc: Elmo E. Collins, NRC Region IV Regional Administrator
Nathanial 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 Project Manager, Office of Nuclear Reactor Regulation

PG&E Response to NRC Letter dated August 25, 2010, Request for Additional Information (Set 19) for the Diablo Canyon License Renewal Application

<u>RAI 4.3-1</u>

<u>Background</u>: In LRA Section 4.3.1, "Cycle Count Action Limits and Corrective Actions" subsection, the applicant identifies that the corrective actions for the Metal Fatigue of Reactor Coolant Pressure Boundary Program if an action limit on the cycle counting of a design basis transient is reached. The applicant states, in part, that if one of the cycle count action limits is reached, corrective actions will include a review of the fatigue usage calculations will be performed to ensure that the analytical bases of the leak-before-break (LBB) fatigue crack propagation analysis is maintained.

The applicant also makes the following statement in LRA Section 4.3.1 to indicate that the action limit on cycle counting would be capable of initiating corrective actions in a timely fashion:

> *"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."*

SRP-LR Section 4.3.2.1.1.3 indicates the program description in GALL AMP X.M1, "Metal Fatigue of Reactor Coolant Pressure Boundary," states an applicant may reference the program in GALL AMP X.M1 to accept a CUF-based metal fatigue TLAA in accordance with the TLAA acceptance criterion in 10 CFR 54.21(c)(1)(iii). The program description in the GALL AMP states that the AMP is an acceptable option for managing metal fatigue for the reactor coolant pressure boundary (RCPB) components, considering environmental effects. The "scope of program" element in GALL AMP X.M1 states that the scope of the program includes preventive measures to mitigate fatigue cracking of metal components of the reactor coolant pressure boundary caused by anticipated cyclic strains in the material.

<u>Issue 1</u>: Fatigue usage calculations are ASME Section III mandated design calculations. LBB fatigue flaw growth analyses are performed pursuant to the requirement in 10 CFR Part 50, Appendix A, General Design Criterion 4, "Dynamic Effects," and are submitted to the NRC for staff approval. It is not evident how a

component fatigue usage factor calculation can be applied to an LBB analysis and how the integrity of the LBB analysis is maintained by this count.

Request 1: Provide your basis for expanding the cycle counting activities of the DCPP Metal Fatigue of Reactor Coolant Pressure Boundary AMP to include the 10 CFR 54.21(c)(1)(iii) aging management of the LBB TLAA. Identify the design basis transients accounted for in the fatigue flaw growth analysis in the LBB. Clarify whether the counting activities will be based on a comparison of the total number of all transients monitored for the LBB or on the number of transient types in the LBB. Clarify whether the relationship between the cycle counting activities in the Metal Fatigue of Reactor Coolant Pressure Boundary Program and the LBB is currently accounted for in a plant procedure or in the UFSAR.

Issue 2: The staff notes that, according to the last sentence of the previously quoted material, the applicant will take corrective actions "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."

Request 2: Identify all transients in LRA Table 4.3-2 that are considered to be the significant contributors to fatigue usage and explain the criteria used to make this determination. Explain why PG&E's cycle count action limit is based on only significant contributors to fatigue usage and does not account for less significant transients. Please describe the confirmatory analysis supporting the conclusion that a lower contributing transient would not significantly impact the CUFs for the components.

PG&E Response to RAI 4.3-1

1. The Diablo Canyon Power Plant (DCPP) Metal Fatigue of Reactor Coolant Pressure Boundary Aging Management Program (AMP) was expanded to include leak-before-break (LBB) because the LBB fatigue crack growth analysis uses the same type of transients used in the initial design of the nuclear steam supply system, which were used to construct the current DCPP Metal Fatigue of Reactor Coolant Pressure Boundary AMP.

   The LBB analysis is described in License Renewal Application (LRA) Section 4.3.2.12. The transients associated with this analysis are shown below in Table 1 (as provided in DCPP's leak-before-break submittal to NRC dated March 16, 1992, DCL-92-059).

The counting activities will be based on a comparison of the number of transient types used in the LBB analysis. The relationship between the cycle counting activities in the Metal Fatigue of Reactor Coolant Pressure Boundary AMP and the LBB is not currently accounted for in a plant procedure or in the Final Safety Analysis Report (FSAR) Update, but it is an enhancement in X.M1 (LRA program B3.1), as stated in LRA Table A4-1.

As shown in DCPP LRA Table 4.3.2, the design transients currently in the DCPP FSAR are:

Table 1

Design Transient	40-yr Transients in LBB Analysis	50-yr Design Transients
Normal		
RCS heatup and cooldown at ≤100°F/hr	200	250
Unit loading and unloading at 5 percent of full power/min	18300	18300
Step increase and decrease of 10 percent of full power.	2000	2500
Large step load decrease	200	250
Steady state fluctuations	10^6	Infinite
Upset		
Loss of load (above 15 percent full power), without immediate turbine or reactor trip	80	100
Loss of all offsite power	40	50
Partial loss of flow	80	100
Reactor trip from full power	400	500
Test Conditions		
Turbine roll test	10	10
Primary side hydrostatic test	5	10
Primary side leak test	50	60
Cold hydrostatic test	10	Not Included

2. All transients in LRA Table 4.3-2 are considered to be the significant contributors to fatigue usage and are tracked by the DCPP Metal Fatigue of Reactor Coolant Pressure Boundary AMP (LRA program B3.1), except those transients identified with a "See Note e." Transients which were deemed nonsignificant are those whose stress intensities are low enough to preclude fatigue or those events which are precluded because of DCPP operating practices. These conclusions are supported by the current design or licensing basis analyses (as discussed in LRA Section 4.3.2) and with the use of engineering judgments.

   Two transients used in the LBB analysis have been deemed nonsignificant: (1) Unit loading and unloading at 5 percent of full power/min, and (2) steady state fluctuations. These transients are not counted because consistent with current plant procedures:

   a. This transient is associated with load following. 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.

   b. The number of steady state fluctuation occurrences listed in the FSAR table is "infinite;" therefore, there is no need to count this transient.

<u>RAI 4.3-2</u>

<u>Background</u>: In LRA section 4.3.1, "Cumulative Usage Corrective Actions" subsection, the applicant states, in part, that if the action limit on the CUF monitoring is reached, corrective actions will include:

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.*

<u>Issue 1</u>: Corrective Action (1) is included in the enhancement of LRA Metal Fatigue of Reactor Coolant Pressure Boundary Program in LRA Appendix A Commitment No. 21. The staff noted that the corrective action is only applicable to reactor coolant pressure boundary components. However, in its review of LRA Section 4.3.2, the staff confirmed that the TLAA does include the CUF results for some ASME Code Class 2 components that were analyzed to ASME Section III CUF requirements for Code Class 1 components. As a result, the staff noted that the action in CUF monitoring corrective action 1 may be applicable to the ASME Code Class 2 components analyzed within the scope of the AMP.

<u>Request 1</u>: Verify corrective action (1) on LRA page 4.3-5, applies to reactor coolant pressure boundary components, component supports, and ASME Code Class 2 components analyzed to ASME Section III CUF requirements for Code Class 1 components.

<u>Issue 2</u>: Corrective Action 2 of LRA page 4.3-5 states "Enhance fatigue managing to confirm continued conformance to the code limit"

<u>Request 2</u>: Clarify what actions would be taken to enhance the fatigue monitoring for this corrective action.

<u>PG&E Response to RAI 4.3-2</u>

1. License Renewal Application (LRA), Appendix A, Commitment Number 21, does not identify ASME Code Class 2 components, as none are included within the scope of the Metal Fatigue of Reactor Coolant Pressure Boundary Aging Management Program. Specifically, the only ASME Code Class 2 component with an identified time limited aging analysis (TLAA) based on a calculated cumulative use factor (CUF) is the steam generator feedwater nozzles that are discussed in LRA Section 4.3.2.5. These components were replaced in 2008 and 2009 and were analyzed for an additional 50 years of operation.

Thus, the associated ASME Code Class 1 fatigue analysis is valid through the period of extended operation and the TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(i), not 10 CFR 54.21(c)(1)(iii).

2. Corrective Action 2 of LRA, page 4.3-5, is not meant to commit to a specific action, but identifies that the methods or assumptions could change (or "be enhanced") to demonstrate that the component is below the ASME Code allowable. For example, the CUF at the location in question: (1) Could be baselined using actual plant historical data in a NB-3200 analysis; (2) The monitoring method could be revised to incorporate revised transients, which removes conservatisms; or (3) Diablo Canyon Power Plant (DCPP) could implement stress-based monitoring, which utilizes six stress tensors or has been appropriately benchmarked. Any corrective actions taken by DCPP to confirm continued conformance with the ASME Code limit will be submitted to the NRC for approval as required.

<u>RAI 4.3-3</u>

<u>Background</u>: LRA Section 4.3.1.1 indicates the applicant will use FatiguePro® to perform the cycle counting for the applicant's design basis transients and updates of the CUF values for ASME Section III Code Class 1 components and for those Class 2 components that were conservatively analyzed to ASME Section III CUF requirements for Class 1 components.

<u>Issue</u>: The staff has confirmed that the use of FatiguePro® software is currently accounted for in the applicant's design basis cycle count procedure. The use of FatiguePro® applies a one-dimensional Green's function method to compute the stress value inputs for the component CUF values that the software program tracks. The staff addressed potential non-conservatisms in the ability of FatiguePro® to perform CUF calculations in NRC RIS 2008-30, "Fatigue Analysis of Nuclear Power Plant Components," dated December 16, 2008. In RIS 2008-30, the staff recommended that license renewal applicants perform an analysis to confirm the use of FatiguePro® would yield conservative CUF values relative to those that would be generated using the ASME Section III Subarticle NB-3200 methods. The staff notes that the use of FatiguePro® is not currently reflected in LRA Commitment No. 21, and the LRA does not provide a basis to determine if the afore mentioned FatiguePro® methodology will yield conservative CUF values relative to the use of the methodology described in ASME Section III, Subarticle NB-3200.

<u>Request</u>: Provide your technical basis to show FatiguePro® cycle tracking and CUF update methodology generates results more conservative than those generated using the CUF methodology of ASME Section III, Subarticle NB-3200. Explain how the Metal Fatigue of Reactor Coolant Pressure Boundary Program addresses the confirmatory analysis, recommended in RIS 2008-30.

<u>PG&E Response to RAI 4.3-3</u>

As described in License Renewal Application (LRA), Section 4.3.1.1 (page 4.3-2), the FatiguePro cycle tracking method (termed cycle based fatigue monitoring method) simply counts the transients to demonstrate the plant is below the analyzed value, thereby demonstrating the code allowable cumulative use factor (CUF) is satisfied.

FatiguePro CUF updates that are credited in the Diablo Canyon Power Plant Metal Fatigue of Reactor Coolant Pressure Boundary Aging Management Program use cycle based fatigue (CBF) methods, which apply usage to the current CUF based on the actual plant events experienced. The usage accumulated from each event is determined using NB-3200 methods. As stated in LRA, Section 4.3.4 (page 4.3-44, footnote 6), the CBF methods do not use the Green's function, therefore RIS 2008-30 does not apply.

<u>RAI 4.3-4</u>

<u>Background</u>: LRA Section 4.3.1.2 provides the applicant's present and projected status of monitored locations. On LRA page 4.3-6, the applicant states that a "review of the operating history of DCPP Units 1 and 2 was performed from initial startup to year-end 2008 in order to baseline the transient event count in the enhanced Fatigue Management Program." In LRA Section 4.3.1.2, Baselining Method subsection, (LRA page 4.3-7), the applicant states that a DCPP specific procedure defines tracking requirements and recording of plant cyclic transients. The applicant states that in 1996, FatiguePro software was installed at DCPP to monitor and record plant instrumentation in order to identify transients and that this provided actual plant transient data from the time of the software installation date through 2008, except for a gap in the data from mid-2002 through year-end 2004, which affected the baseline count for the charging and feedwater (FW) cycling transients. LRA Section 4.3.1.2, Baselining Method subsection also provides specific details on the cycle count baselining methods and assumptions for the "Auxiliary Spray during Cooldown" transient, RHR Operation (during Cooldown)" transient, charging cycling transient, and FW cycling transient.

<u>Issue 1</u>: LRA Section 4.3.1.2 gives no indication about the rigor used to develop the cycle count at DCPP. On page 4.3-7, the applicant only states that "data from several sources were considered" for the recount activities.

<u>Request 1</u>: Identify the sources of information used to develop the DCPP transient operating history.

<u>Issue 2</u>: On LRA page 4.3-7, the applicant states that, after considering the documented sources of cycle counting information, "an explicit cycle count could not be determined for some transients." However, the LRA does not identify which transients are not determined explicitly.

<u>Request 2</u>: Identify the transients that were not derived explicitly. Discuss the technical rationale used to derive the 60-year cycle projections for the identified transients.

<u>Issue 3</u>: The applicant's number-of-events basis for the "Auxiliary Spray during Cooldown" transient is given on LRA page 4.3-7. The staff has determined that LRA Table 4.3-2 does not list this transient as within the scope of the design basis transients for this TLAA.

<u>Request 3</u>: Provide the basis for excluding the "Auxiliary Spray during Cooldown" transient from LRA Table 4.3-2.

<u>Issue 4</u>: The applicant's number of events basis for the charging system is given at the bottom of LRA page 4.3-7. In LRA Table 4.3-2, the applicant identifies three transients for the charging system (Transients 15, 16, and 17 in the table). The applicant does not provide any correlation in the LRA between the number of events basis for charging system on LRA page 4.3-7 and the design basis transients in LRA Table 4.3-2 that are

impacted by this charging system basis. The applicant also does not specify which quantitative SF was applied to these events or justify its use in the projection basis.

Request 4: *Identify which of the transients in LRA Table 4.3-2 were assessed in accordance with charging system events basis that was provided at bottom of page 4.3-7. For each of the transients that were assessed in accordance with this projection basis, identify the SF that was applied to the assessment and justify its use.*

PG&E Response to RAI 4.3-4

1. The Diablo Canyon Power Plant (DCPP) transient operating history information was taken from:

   - The current plant transient tracking procedure in which transient data is provided by plant operators and is verified by engineering; and

   - Computer-assisted cycle counting records (actual plant operating data obtained from the plant process computer).

2. The absence of an event was confirmed for the following transients by interviews with DCPP plant personnel (engineering, operations, and licensing) and by review of reportable events.

   - Inadvertent reactor coolant system (RCS) depressurization

   - Excessive feedwater flow

   All other transients which do not include an explicit cycle count are discussed below:

   Events Related to Other Counted Events
   As stated in License Renewal Application (LRA), Section 4.3.1.2 (page 4.3-7), the numbers of events to date for "Auxiliary Spray during Cooldown" and "RHR Operation (during Cooldown)" are based on an assumed number of events per RCS cooldown. Specifically, the "Auxiliary Spray during Cooldown" event generally occurs one or more times late in each cooldown, when normal spray becomes unavailable (because the reactor coolant pumps [RCPs] must be taken off-line at low RCS pressures). It is assumed to occur twice for each counted "Plant Cooldown" event. The "RHR Operation (during Cooldown)" event happens when the RHR system is first brought on-line late in a cooldown (to continue cooling the RCS after the RCPs are stopped). This event is assumed to occur once per "Plant Cooldown" event.

   Charging System Events
   As stated in LRA, Section 4.3.1.2 (page 4.3-7), the numbers of events for the charging system are based on the event frequency for which data is available. The charging system transients include:

Transient Number from LRA Table 4.3-2	Transient Description
15	Charging and Letdown Flow Shutoff and Return to Service (Loop 4 / 3)
16	Loss of Charging with Prompt Return to Service (Loop 4 / 3)
17	Loss of Charging with Delayed Return to Service (Loop 4 / 3)
18	Loss of Letdown with Prompt Return to Service (Loop 4 / 3)
19	Loss of Letdown with Delayed Return to Service (Loop 4 / 3)

Feedwater Cycling

As stated in LRA Section, 4.3.1.2 (page 4.3-8), the feedwater cycling events are assumed to correlate to pressurizer heatup cycles. The numbers of events to date was determined by taking the ratio of the number of documented pressurizer heatups through 2008 to the number of expected pressurizer heatups for 60 years of operation and multiplying it by the total number of allowed feedwater cycling events (2,500). For DCPP Unit 1, there were 49 pressurizer heatups through 2008 and 179 total pressurizer heatups projected for 60 years. For DCPP Unit 2, there were 33 pressurizer heatups through 2008 and 179 total pressurizer heatups projected for 60 years.

3. The "Auxiliary Spray during Cooldown" transient discussed on LRA, page 4.3-7, is included in the DCPP Metal Fatigue of Reactor Coolant Pressure Boundary Aging Management Program. See revised LRA, Table 4.3-2, in Enclosure 2.

4. The charging system transients are identified in PG&E's response to Request for Additional Information 4.3-4, Part 2, above.

   A safety factor of 2.15 is applied in all charging system transient cases to account for the likely higher rate of events during periods for which no actual instrument data is available (prior to FatiguePro installation and from mid-2002 to year-end 2004). It was considered that reactor trips would constitute an extreme example of this effect. Considering the reactor trips recorded by plant procedures (from 1984 to 2008), the trips during the unmonitored periods occurred 2.15 times more often than during the monitored periods.

RAI 4.3-5

Background: On page LRA 4.3-8, the applicant states that the projection rate (PR) for the unaccounted periods were performed using both a long-term rate based on the entire transient history for the plant (i.e., number of occurrences since initial plant startup) and a short term rate for the incremental cycles that have occurred over the last 10 years. On this page, the applicant states that the two rates were combined using a weighted average in accordance with the following equation:

PR = [(LTW)(long-term rate) + (STW)*(short-term rate)] / [(LTW) + (STW)],*

with LTW being the long-term weighting factor and STW being the short-term weighting factor.
The applicant states that the values of LTW and STW were determined on an event- or component-specific basis to reflect the most likely future behavior of that event or component.

Issue 1: It is not evident how the LTW and STW values could be derived on a component-specific basis when presumably the design basis CUF calculations for Class 1 components (and possibly some Class 2 components analyzed to ASME Section III Class 1 CUF criteria) would involve more than one analyzed transient, and under this basis individual LTW and STW values would have to be assigned to each transient contributing to the CUF calculation for a given component.

Request 1: Explain the technical rationale for selection of LTW and STW and how this accommodates events on a component basis.

Issue 2: The PR basis provided on LRA page 4.3-8 only involves a general description about the PR value derivation; the LTW and STW values were derived on a transient-specific (event-specific) or component-specific basis. Thus, the PR basis discussion on LRA page 4.3-8 does not provide the staff with any quantitative basis correlation with the LTW and STW factors used to derive the PRs for the design basis transients in LRA Section 4.3.

Request 2: Identify which transients, in LRA Table 4.3-2, this applies to. Explain how the LTW and STW values were used for the transient projection basis.

PG&E Response to RAI 4.3-5

1. The long-term weighting (LTW) and short-term weighting (STW) values are not derived on a component-specific basis. The LTW and STW values are only derived on an event-specific basis. See revised License Renewal Application, page 4.3-8, in Enclosure 2.

2. The specific LTW and STW values used for each transient were estimated by taking into account the history of each transient, number of cycles, distribution, and

qualities of the transient itself. Values were then selected which would likely work. These values were then compared with the cycle history plot. If the plot showed a projection that fit the past history, the work was done. If not, the weights were adjusted until satisfied with the results.

Assuming no other information, it was assumed that the short-term past was 3 times more likely to predict future performance than the long-term history (i.e., STW = 3, LTW = 1). This was modified based on empirical factors as follows:

- If an event had few total cycles, then the distribution of those events is more likely to reflect random variation than deterministic trends; this would indicate a reduction in the STW relative to the LTW. In cases with very few occurrences, the STW may be reduced to zero - giving a simple linear projection based on the full history.

- If the distribution of past events showed a clear pattern of either increasing or decreasing rate of occurrence, then the STW was increased relative to the LTW.

- STW values were increased for transients relating to planned evolutions (e.g., *Aux. Spray during C/D, RHR Operation* and *Refueling*). Transients that reflect unplanned or accident conditions *(e.g., Loss of Power and Loss of Load)* had their STW values reduced.

The cycle projections were determined in SIA calculation FP-PGE-305, *Cycle and Fatigue Baseline up through YE 2008*. Plots of the cycle histories and projections are provided on pages 29-53. A tabulation of the specific STW and LTW values for each transient, taken from that calculation, is reproduced on the following page.

DCPP, Unit 1 Transient	1/01/84	# cyc as of: 1/01/99	1/01/09	25.00 LTR	10.00 STR	LTW	STW	Rate
Aux. Spray during C/D	20	60	78	2.320	1.800	1	3	1.9299
Charging SI into Cold Leg	0	13	15	0.600	0.200	1	2	0.3333
Control Rod Drop	0	0	0	0.000	0.000	1	3	0.0000
High Head SI into CL	0	0	1	0.040	0.100	1	3	0.0850
Inadv. Aux. Spray Actuation	0	2	2	0.080	0.000	1	0	0.0800
Large Step Load Decrease	0	4	5	0.200	0.100	1	1	0.1500
Loss of All Offsite Power	0	1	1	0.040	0.000	1	1	0.0200
Loss of Load (TT w/o RT)	0	0	5	0.200	0.500	1	1	0.3500
Lp4 Chrg & Ltdn Shutoff	0	2	3	0.120	0.100	1	1	0.1100
Lp4 Chrg Trip Delayed Rtn.	1	3	3	0.080	0.000	1	0	0.0800
Lp4 Chrg Trip Prompt Rtn.	8	48	64	2.240	1.600	1	3	1.7599
Lp4 Ltdn Trip Delayed Rtn.	1	6	6	0.200	0.000	1	0	0.2000
Lp4 Ltdn Trip Prompt Rtn.	7	43	60	2.120	1.700	1	3	1.8049
Partial Loss of Flow	0	1	1	0.040	0.000	1	0	0.0400
Plant (RCS) Cooldown	10	30	42	1.280	1.200	1	3	1.2200
Plant (RCS) Heatup	10	31	43	1.320	1.200	1	3	1.2300
Pressurizer Cooldown	10	33	49	1.560	1.600	1	3	1.5900
Pressurizer Heatup	10	34	49	1.560	1.500	1	3	1.5150
RHR Operation (Train A)	10	30	42	1.280	1.200	1	3	1.2200
RHR Operation (Train B)	10	30	42	1.280	1.200	1	3	1.2200
Reactor Trip - C/D no SI	0	10	10	0.400	0.000	1	1	0.2000
Reactor Trip - no C/D	0	45	48	1.920	0.300	1	2	0.8399
Refueling	0	8	14	0.560	0.600	1	3	0.5900
Step Load Decrease 10 percent	0	2	2	0.080	0.000	1	0	0.0800
Step Load Increase 10 percent	0	22	25	1.000	0.300	2	1	0.7666
Switchover Norm/Alt Charg.	0	0	1	0.040	0.100	1	3	0.0850
Tavg Coastdn to Red. Temp.	0	4	6	0.240	0.200	1	3	0.2100

DCPP, Unit 2 **Transient**	1/01/85	# cyc as of: 1/01/99	1/01/09	24.00 LTR	10.00 STR	LTW	STW	Rate
Aux. Spray during C/D	14	42	54	1.667	1.200	1	3	1.3167
Charging SI into Cold Leg	0	13	13	0.542	0.000	1	2	0.1806
Control Rod Drop	0	1	1	0.042	0.000	2	1	0.0278
High Head SI into CL	0	0	0	0.000	0.000	1	3	0.0000
Inadv. Aux. Spray Actuation	4	5	5	0.042	0.000	1	0	0.0417
Large Step Load Decrease	0	3	4	0.167	0.100	1	3	0.1167
Loss of All Offsite Power	0	1	1	0.042	0.000	3	1	0.0312
Loss of Load (TT w/o RT)	0	0	3	0.125	0.300	3	1	0.1687
Lp4 Chrg & Ltdn Shutoff	0	2	4	0.167	0.200	1	3	0.1917
Lp4 Chrg Trip Delayed Rtn.	0	3	3	0.125	0.000	1	1	0.0625
Lp4 Chrg Trip Prompt Rtn.	1	38	54	2.208	1.600	1	3	1.7521
Lp4 Ltdn Trip Delayed Rtn.	0	4	6	0.250	0.200	1	3	0.2125
Lp4 Ltdn Trip Prompt Rtn.	1	33	45	1.833	1.200	1	3	1.3583
Partial Loss of Flow	0	2	3	0.125	0.100	1	1	0.1125
Plant (RCS) Cooldown	7	21	30	0.958	0.900	1	3	0.9146
Plant (RCS) Heatup	7	22	31	1.000	0.900	1	3	0.9250
Pressurizer Cooldown	7	21	32	1.042	1.100	1	3	1.0854
Pressurizer Heatup	7	22	33	1.083	1.100	1	3	1.0958
RHR Operation (Train A)	7	21	29	0.917	0.800	1	3	0.8292
RHR Operation (Train B)	7	21	29	0.917	0.800	1	3	0.8292
Reactor Trip - C/D no SI	0	11	11	0.458	0.000	1	2	0.1528
Reactor Trip - no C/D	0	35	37	1.542	0.200	1	2	0.6472
Refueling	0	8	14	0.583	0.600	1	3	0.5958
Step Load Decrease 10 percent	0	3	5	0.208	0.200	1	3	0.2021
Step Load Increase 10 percnet	0	20	25	1.042	0.500	1	3	0.6354
Switchover Norm/Alt Charg.	0	0	0	0.000	0.000	1	3	0.0000
Tavg Coastdn to Red. Temp.	0	3	4	0.167	0.100	1	3	0.1167

RAI 4.3-6

Background: LRA Table 4.3-2 provides the applicant's list of design basis transients that pertain to the metal fatigue assessments for ASME Code Class 1, 2, or 3 components or components designed to the ANSI B31.1 design specification. UFSAR Table 5.2-4 provides a list of design basis transients for the DCPP units.

Issue 1: The applicant has determined that UFSAR Table 4.3-2 provides an accurate correlation for all normal operation condition, upset condition, and test condition transients and their design limits in UFSAR Table 5.2-4, with the exception of normal operating condition transient #8, "T_{avg} Coastdown from Nominal to Reduced Temperature," which currently is not within the scope of LRA Table 4.3-2.

Request 1: Provide your basis for why UFSAR Table 5.2-4, normal operating condition transient #8, "T_{avg} Coastdown from Nominal to Reduced Temperature," is not currently within the scope of LRA Table 4.3-2 and why the applicable 60-year cycle projection data have not been included for this transient in LRA Table 4.3-2.

Issue 2: LRA Table 4.3-2 identifies that the normal operating condition transient Nos. 5, 13, 14, 15, 16, 17, 18, and 19, and upset condition transient Nos. 24, 26, 27, 28, 29, 30, and 31 are applicable to the scope of the metal fatigue analyses but are not currently within the scope of UFSAR Table 5.2-4.

Request 2: Clarify how these transients relate to the scope of the design basis that is currently described in the DCPP UFSAR (if at all) or applicable design basis procedures or calculations.

Issue 3: LRA Table 4.3-2 includes transient data entries for the "Design Basis Cycles, FSAR Table 5.2-4" and "Limiting Analyzed Value" columns in the table. The "Limiting Analyzed Value" column is subject to the following Footnote (c) clarification:

> *"The limiting analyzed value is the lowest number of transients that are considered in DCPP fatigue analyses. The enhanced Fatigue Management Program compares actual to this limiting analyzed value so that all plant analyses remain valid."*

The staff has observed that for those transients in LRA Table 4.3-2 that derive from the list of transients in UFSAR Table 5.2-4, the value listed in the "Limiting Analyzed Value" column is sometimes the same as that listed in the "Design Basis Cycles, FSAR Table 5.2-4" column and sometimes it is lower than the value listed in the "Design Basis Cycles, FSAR Table 5.2-4" column.

Request 3: Clarify which columns (the value in the "Design Basis Cycles, FSAR Table 5.2-4" column or the value in the "Limiting Analyzed Value" column) should be relied upon for the design basis transient occurrence limits.

<u>Issue 4</u>: LRA Table 4.3-2 includes test condition transient #37, "Tube Leak Tests." The applicant identifies 800 as the value for the "Design Basis Cycles, FSAR Table 5.2-4" column and "Limiting Analyzed Value" column entries for this transient. The staff has determined however, that UFSAR Table 5.2-4 lists this as test condition transient #3.b, and that for this transient, the design basis is broken down into four cases for the transient as follows:

- *Case 1 with a design limit of 400 cycles*
- *Case 2 with a design limit of 200 cycles*
- *Case 3 with a design limit of 120 cycles*
- *Case 4 with a design limit of 80 cycles*

<u>Request 4</u>: Justify why the "Design Basis Cycles, FSAR Table 5.2-4" column and "Limiting Analyzed Value" column entries in LRA Table 4.3-2 for "Tube Leak Test" transient are not same as those given in UFSAR Table 5.2-4 for this transient. Specifically define and discuss each of the Case bases for this transient as defined in UFSAR Table 5.2-4, and explain how DCPP arrived at design basis limit values for each of the Case bases (i.e., for Cases 1 – 4).

<u>PG&E Response to RAI 4.3-6</u>

1. As stated in Diablo Canyon Power Plant (DCPP) Final Safety Analysis Report (FSAR) Update, Revision 18, Table 5.2-4, footnote c, for the replacement steam generators (SGs), "Tavg/power coastdown design transient conditions are enveloped by analyses and evaluations contained in a design change implemented to support operation over a Tavg range of 565°F to 577.6°F for Cycle 15." As of submittal of this License Renewal Application (LRA), all old SGs had been replaced by replacement SGs. The revised FSAR Update (Revision 19) was submitted to NRC in 2010 and reflects the removal of this transient from FSAR Table 5.2-4. Hence, this transient is no longer part of the design basis, does not need to be projected to
60 years, and will not be tracked.

2. Although most of the transients mentioned in Request for Additional Information 4.3-7 are not cited in the FSAR Update, they are used in design basis analyses; and therefore, will conservatively be monitored by the DCPP Metal Fatigue of Reactor Coolant Pressure Boundary Aging Management Program (AMP). The basis for inclusion of these transients is shown below:

LRA Table 4.3-2 Item Number	Transient Name	Inclusion in Design Basis
	Normal	
5	Pressurizer Heatup or Cooldown Cycle in Excess of Tech. Specs. and Within the Bounds of WNEP-8828, Rev. 1	Identified in the Pressurizer design reports.
13	Residual Heat Removal Initiation During Cooldown	Used in the reanalysis of the several NUREG/CR-6260 locations.
14	Refueling	Identified in the analysis Pressurizer surge line for thermal stratification (NRC Bulletin 88-11). Used in the reanalysis of the several NUREG/CR-6260 locations.
15	Charging and Letdown Flow Shutoff and Return to Service	Used in the re-analysis of the several NUREG/CR-6260 locations.
16	Loss of Charging with Prompt Return to Service	
17	Loss of Charging with Delayed Return to Service	
18	Loss of Letdown with Prompt Return to Service	
19	Loss of Letdown with Delayed Return to Service	
	Upset	
24	Inadvertent Reactor Coolant System (RCS) Depressurization (Resulting in Reactor Trip)	Identified in the analysis Pressurizer surge line for thermal stratification (NRC Bulletin 88-11). Used in the reanalysis of the several NUREG/CR-6260 locations.
26	Control Rod Drop	Identified in the Pressurizer design reports. Identified in the analysis of the Pressurizer surge line for thermal stratification (NRC Bulletin 88-11). Used in the reanalysis of the several NUREG/CR-6260 locations.
27	Inadvertent Emergency Core Cooling System Actuation	
28	Excessive Feedwater Flow	
29	Safety Injection into RCS Cold Leg / High Head Safety Injection	Used in the reanalysis of the several NUREG/CR-6260 locations.
30	Inadvertent Accumulator Blowdown	
31	Design Earthquake (OBE)	Identified in FSAR Table 5.2-4 as upset transient #6, Design Earthquake.

3. The values in FSAR Table 5.2-4 are the design basis values, meaning all future design work should meet these values. However, this does not mean that all historical fatigue analyses meet these values. During the development of LRA Section 4.3, some analyses were identified which were performed using a transient number other than those values in FSAR Table 5.2-4. If the number of transients in the analysis were more limiting than values in FSAR Table 5.2-4, then these values were then incorporated into the DCPP Metal Fatigue of Reactor Coolant Pressure Boundary AMP and were identified in the "Limiting Analyzed Value" column.

   The "Limiting Analyzed Value" column should be used when determining what value the DCPP Metal Fatigue of Reactor Coolant Pressure Boundary AMP will count to.

4. The 800 tube leak test cycles, listed in LRA, Table 4.3-2, is the summation of Cases 1 through 4 that are listed in FSAR Table 5.2-4, and was meant to be a simplification for the purposes of the LRA.

   FSAR Section 5.2.1.5.5 provides the following details on the "Tube Leak Test" transient.

Case	Test Pressure, psig	FSAR Table 5.2-4 No. of Occurences
1	200	400
2	400	200
3	600	120
4	840	80

   The current plant cycle counting procedure monitors each of the four cases individually.

RAI 4.3-7

Background: LRA Section 4.3 dispositions the CUF-based TLAAs for many ASME Code Class 1 components by multiplying the CUF values for the components by a factor of 1.2 if the design basis CUF was based on a 50-year design life or by 1.5 if the design basis CUF was based on a 40-year design life. For these TLAAs, DCPP states that the CUF values remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

Issue: The multiplication of the design basis CUF by a factor of 1.2 or 1.5 represents a projection of the CUF value for the period of extended operation in that it is changing the CUF value for the component. Thus, components dispositioned in accordance with this methodology should be dispositioned in accordance with the criteria in 10 CFR 54.21(c)(1)(ii) in that the CUF values have been projected for the period of extended operation and have been found to be acceptable when compared to a CUF value acceptance criterion of 1.0.

Request: Provide your basis why Class 1 components that are subject to this metal fatigue projection basis have not been dispositioned in accordance with the criterion in 10 CFR 54.21(c)(1)(ii).

PG&E Response to RAI 4.3-7

PG&E agrees that the multiplication of the design basis cumulative use factor (CUF) by a factor of 1.2 or 1.5 represents a projection of the CUF value for the period of extended operation in that it is changing the CUF value for the component. Therefore, the License Renewal Application (LRA) has been revised. See revised LRA Sections 4.1, 4.3, and Appendix A3 in Enclosure 2.

Sections to revise:

- Table 4.1-1 (LRA page 4.1-6 thru 4.1-7) for the below 4.3 sections
- Section 4.3.2.1 (LRA page 4.3-15)
- Section 4.3.2.2 (LRA page 4.3-16 & 17)
- Section 4.3.2.3 (LRA page 4.3-20)
- Section 4.3.2.4 (LRA page 4.3-24)
- Section 4.3.4 (LRA page 4.3-45)
- LRA Appendix A3 (pages A-29, A-30, A-31, A-34) for the above 4.3 sections

RAI 4.3-8

Background: In the LRA Table 4.3-1, the applicant credits the "Global" monitoring (i.e. cycle count monitoring) of AMP B3.1 as the 10 CFR 54.21(c)(1)(iii) aging management monitoring basis for dispositioning the CUF analyses for the RPV Core Support Pads, Pressurizer Spray Nozzle, and Pressurizer Heater Penetration.

Issue: LRA Table 4.3-1 or LRA Table 4.3-6 indicated that the RPV Core Support Pads, Pressurizer Spray Nozzle, and Pressurizer Heater Penetration in Unit 1 have a maximum limiting design basis CUFs of ~0.89, ~0.95, and ~0.94 respectively and limiting 60-year projected CUFs of ~1.07, ~1.14, and ~0.9391.

Request: Justify your basis using the "Global" monitoring method of AMP B3.1 to monitor these components during the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii), and why it would not be more appropriate to monitor for these components using the CBF monitoring method.

PG&E Response to RAI 4.3-8

A fundamental basis for the Diablo Canyon Power Plant (DCPP) Metal Fatigue of Reactor Coolant Pressure Boundary Aging Management Program is that as long as the number of transients used in the analysis remain below the analyzed value, then it has be demonstrated that the components are less than the code allowable value and structural integrity is demonstrated.

All transients included in the design basis for the Unit 1 reactor pressure vessel (RPV) core support pads, the pressurizer spray nozzle, and the pressurizer heater penetration are either: (1) counted when the actual transient cycles are experienced by the plant, or (2) determined that the transient used in the design basis does not need to be counted. A list of the transients not being counted, and the basis for not counting them, is included below:

LRA Table 4.3-2 Item Number	**Component Applicability**	**Transient Name**	**Basis for not Counting the Transient**
6 & 7	RPV Core Support Pad Pressurizer Spray Nozzle Pressurizer Heater Penetration	Unit loading and unloading at 5 percent of full power/min	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.

LRA Table 4.3-2 Item Number	Component Applicability	Transient Name	Basis for not Counting the Transient
Not Included	RPV Core Support Pad Pressurizer Heater Penetration	Reduced Temperature Return to Power	This transient is associated with load following operation. DCPP does not operate as a load following plant. Thus, it is not necessary to track this transient's occurrence.
11	RPV Core Support Pad Pressurizer Heater Penetration	Steady State Fluctuations	The number of steady state fluctuation occurrences listed in the Final Safety Analysis Report table is "infinite;" therefore, there is no need to count this transient.
Not Included	Pressurizer Spray Nozzle Pressurizer Heater Penetration	Boron Concentration Equalization	This transient is associated with load following operation. DCPP does not operate as a load following plant. Thus, it is not necessary to track this transient's occurrence.
Not Included	Pressurizer Heater Penetration	Loop Out-of-Service	The loop out-of-service event is not a credible transient for DCPP because the DCPP operating licenses do not allow operation with a loop out of service. Thus, this transient does not need to be tracked.
Not Included	Pressurizer Heater Penetration	Inadvertent Startup of an Inactive loop	This transient is associated with a loop out of service. The loop out-of-service event is not a credible transient for DCPP because the DCPP operating licenses do not allow operation with a loop out of service. Thus, this associated transient does not need to be tracked.

RAI 4.3-9

Background: LRA Section 4.3.2.3 (top of pg. 4.3-20) states that the "Unit Loading and Unloading" transient does not need to be counted under the enhanced fatigue management program.

Issue: The staff have determined that the the "Unit Loading and Unloading" transient is within the scope of FSAR Section 5.2.1.5.1 and UFSAR Table 5.2-4, and that Technical Specification (TS) 5.5.5 makes reference to 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. Thus, the staff is of the perception that the counting of this transient would be the activity that corresponds to the control to track the transient under the TS requirement.

Request: Provide your basis why the Metal Fatigue of Reactor Coolant Pressure Boundary Program would not need to count this transient during the period of extended operation when it does appear to be within the scope of the TS tracking requirement.

PG&E Response to RAI 4.3-9

Consistent with current plant procedures, which implement Technical Specification 5.5.5, the current operating strategy for the Diablo Canyon Power Plant (DCPP) units is continuous base-load power generation. Therefore, the actual number of unit loading and 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.

DCPP cannot change the current operating strategy from continuous base-load power generation to load following since the current design basis does not support load following.

<u>RAI 4.3-10</u>

<u>Background</u>: LRA Section 4.3.3 provides the fatigue analyses of the reactor pressure vessel internals. The applicant stated that the qualification of reactor vessel internals was first performed by Westinghouse on a generic basis for 40 years of operation. The applicant stated that some DCPP internal components were subsequently analyzed on a DCPP-specific basis. The applicant indicated that the lower support plate, lower support columns, and core barrel nozzles had the highest cumulative usage factor values (CUF values) for the reactor vessel internal (RVI components and that the CUFs for the remaining RVI components were bounded by the CUF results for these limiting components. The applicant further stated that the enhanced DCPP Fatigue Management Program will monitor the 50-year design basis number of transients used in the T_{avg} operating range analysis to ensure that it remains valid during the period of extended operation.

<u>Issue 1</u>: The staff is unable to determine from the LRA discussion which RVI components were required to be analyzed for fatigue as part of the ASME Section III design.

<u>Request 1</u>: Identify all DCPP RVI components that were required to receive CUF calculations under applicable ASME Section III design requirements. For these components, identify the transients that were involved in the calculation of the CUF values and identify what the CUF values are for the components, along with an indication on whether the value for a given RVI component represents an existing design basis value or 60-year projected values. Clarify how the value was calculated if the CUF value for the given RVI components represents a 60-year project value for the TLAA.

<u>Issue 2</u>: The LRA indicated that the fatigue of the RVI components will be managed by the DCPP Fatigue Management Program by monitoring the number of transients. The LRA does not provide any justification why it would be acceptable for the applicant to use cycle monitoring of the transients for the lower support plates, lower support columns, and core barrel nozzles as a bounding basis for monitoring the other RVI components that received CUF calculations.

<u>Request 2</u>: Provide your basis for why it is acceptable to use cycle-based monitoring of the transients associated with the lower support plates, lower support columns, and core barrel nozzles as a bounding basis for those non-monitored RVI components with CUF values.

PG&E Response to RAI 4.3-10

1. The reactor vessel internals components presented in the table below are those that were required to receive cumulative use factor (CUF) calculations applicable to ASME Section III design requirements. The table also lists the results of the Diablo Canyon Power Plant (DCPP)-specific fatigue analyses for the existing plant design basis (i.e., 50 years).

Component	**50-Year Usage Factor for Unit 1** (existing design basis)	**50-Year Usage Factor for Unit 2** (existing design basis)
Lower Support Plate – Atypical Region	0.52	0.706
Lower Support Columns	0.945	0.486
Core Barrel Nozzle – Section A-A	0.413	0.413
Lower Support	0.388	0.388
Lower Core Plate	0.52	0.52
Upper Core Plate	0.8	0.88
Baffle Bolts	≤1.0	≤1.0

The table below presents the transients that were used to calculate the CUF values above.

Design Transients	
Normal	**Design Number of Cycles**
Plant Heatup	250
Plant Cooldown	250
Unit Loading at 5 Percent/Min	18,300
Unit Unloading at 5 Percent/Min	18,300
Step Load Increase of 10 Percent of Full Load	2500
Step Load Decrease of 10 Percent of Full Load	2500
Large Step Decrease with Steam Dump	250

Design Transients	
Upset	
Loss of Load w/o Immediate Turbine or Reactor Trip	100
Loss of Power	50
Partial Loss of Flow	100
Reactor Trip from Full Power	500
Design Earthquake	20

2. A fundamental basis for the DCPP Metal Fatigue of Reactor Coolant Pressure Boundary Aging Management 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.

   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. A list of the transients used in the reactor vessel internals analyses that are not being counted and the basis for not counting them is included below:

LRA Table 4.3-2 Item Number	Transient Name	Basis for not Counting the Transient
6 & 7	Unit loading and unloading at 5 percent of full power/min	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.

<u>RAI 4.3-11</u>

<u>Background</u>: The GALL Report states that the AMP addresses the effects of coolant environment by applying an environmental life correction factors to existing ASME code fatigue analyses based on factors in NUREG/CR-6583 and NUREG/CR-5704, or appropriate alternative methods.

<u>Issue</u>: The applicant has stated that the environmental factors are determined by NUREG/CR-6583 and NURGE/CR-5704, or appropriate alternative methods.

<u>Request</u>: Clarify what appropriate alternative method would be used to calculate the environmental factors for fatigue calculations.

<u>PG&E Response to RAI 4.3-11</u>

This statement regarding "appropriate alternative methods" is not meant to commit to a specific method, but merely identifies that alternative methods exist (as stated in NUREG-1801) to calculate environmentally assisted fatigue factors (F_{en}). As stated in License Renewal Application, Section 4.3.4, to address the environmental effects on fatigue, Diablo Canyon Power Plant (DCPP) used material-specific guidance presented in NUREG/CR-6583 and NUREG/CR-5704.

The determination of an "appropriate alternative method" can only be made by the NRC. Therefore, if DCPP uses an "appropriate alternative method" in the future, it would require the approval of the NRC.

RAI 4.3-12

Background: 10 CFR Part 54.21 states that each application must identify and list those structures and components subject to an aging management review.

Issue 1: LRA Section 4.3 indicates that the following components were required to be analyzed in accordance with an applicable CUF analysis; however, the AMR Tables in LRA Section 3.1 do not appear to include applicable AMR items that address cumulative fatigue damage for the components:

- *RV core support lugs or pads (as indicated in LRA Table 4.3-1)*
- *RV inlet and outlet nozzle support pads (as indicated in LRA Table 4.3-1*
- *Reactor coolant pump (RCP) casings (as indicated in LRA Section 4.3.2.3)*
- *Valve support bracket for the Unit 2 pressurizer (as indicated in LRA Table 4.3-6)*
- *SG primary manway, secondary, and feedring components (as indicated in LRA Table 4.3-7)*
- *RV internal lower support plate, lower support columns, core barrel nozzles, and baffle-former plates (as indicated in LRA Section 4.3.3)*

Request 1: Provide your basis why the AMR tables in LRA Section 3.1 do not appear to include any AMR items addressing the management of cumulative fatigue damage for these components.

Issue 2: The staff have noted that the LRA includes AMRs on cumulative fatigue damage only for ASME Section III Class 2 or 3 or ANSI B31.1 piping in the following balance of plant emergency safety feature (ESF), auxiliary system (AUX), and steam and power conversion subsystems:

- *Safety Injection System (LRA Table 3.2.2-1)*
- *RHR System (LRA Table 3.2.2-3)*
- *Chemical and Volume Control System (LRA Table 3.3.2-8)*
- *Turbine Steam Supply System (LRA Table 3.4.2-1)*
- *Feedwater System (LRA Table 3.4.2-3)*
- *Auxiliary Feedwater System (LRA Table 3.4.2-5)*

Request 2: Provide your basis why the AMR tables in LRA Sections 3.2, 3.3, and 3.4 do not appear to include any AMR items addressing cumulative fatigue damage for the ANSI B31.1 or B31.7 piping components in the systems:

- *LRA Table 3.2.2-2, Containment Spray System AMRs*
- *LRA Table 3.2.2-4, Containment HVAC System AMRs*
- *All Table 2 AMR Tables for AUX subsystems in LRA Section 3.2 other than that for Table 3.3.2-8, Chemical and Volume Control System*
- *LRA Table 3.4.2-2, Auxiliary Steam System*
- *LRA Table 3.4.2-4, Condensate System*

PG&E Response to RAI 4.3-12

1. See revised License Renewal Application (LRA), Tables 3.1.2-1 and 3.1.2-4, in Enclosure 2 to add aging management review (AMR) items for the following components subject to cumulative fatigue damage:

   - Reactor vessel (RV) nozzle support pads
   - Reactor vessel and internals (RVI) core barrel assembly
   - RV core support lugs
   - Valve support bracket (Unit 2 only)
   - Steam generator (SG) secondary manway and handhole covers
   - SG feedwater ring
   - SG primary manway covers

   The following list specifies those components that did not require LRA revisions and the logic for this determination:

   - Reactor coolant pump casing – Currently included as component type "pump" in LRA, Table 3.1.2-2, page 3.1-94.
   - RV internal lower support plate – Currently included as part of "RVI Lower Core Support Structure (All RVI Stainless Steel Components)" in LRA, Table 3.1.2-2, page 3.1-78.
   - RV internal lower support column – Currently included as part of "RVI Lower Core Support Structure (All RVI Stainless Steel Components)" in LRA Table 3.1.2-2, page 3.1-78.
   - Baffle former plates - RVI baffle & former assembly – No time limited aging analysis (TLAAs) exist for this component, therefore no TLAA line is required.

2. With the exception of those listed below, the piping systems listed in Request for Additional Information 4.3-12: (1) are designed to ASME Class 2, 3, or ANSI B31.1 piping requirements, (2) are within the scope of license renewal, and (3) 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. Their inclusion in the AMR tables would only reference 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 other generic aging lessons learned line items.

   - Miscellaneous heating, ventilation, and air conditioning (HVAC) (Table 3.3.2-9) – Not designed to ASME Class 2, 3, or ANSI B31.1 requirements
   - Control Room HVAC (Table 3.3.2-10) – Not designed to ASME Class 2, 3, or ANSI B31.1 requirements
   - Auxiliary Building HVAC (Table 3.3.2-11) – Not designed to ASME Class 2, 3, or ANSI B31.1 requirements

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LRA Section	RAI
Table 4.3-2	4.3-4
Section 4.3.1.2	4.3-5
Table 3.1.2-1	4.3-5
Table 3.1.2-3	4.3-5
Table 3.1.2-4	4.3-5
Table 4.1-1	4.3-7
Section 4.3.2.1	4.3-7
Section 4.3.2.2	4.3-7
Section 4.3.2.3	4.3-7
Section 4.3.2.4	4.3-7
Section 4.3.4	4.3-7
Section A3.2.1.1	4.3-7
Section A3.2.1.2	4.3-7
Section A3.2.1.3	4.3-7
Section A3.2.1.4	4.3-7
Section A3.2.3	4.3-7
Section 4.3.1.2	4.3-5

Table 4.3-2 DCPP Units 1 and 2 Transient Cycle Count and 60-year Projections[i, ii]

Transient Description	Design Basis Cycles, FSAR Table 5.2-4	Limiting Analyzed Value[iii]	Unit 1		Unit 2	
			Events (1984-2008)	Projected Events for 60-Years	Events (1985-2008)	Projected Events for 60-Years
14. Refueling[f]	NS	80	14	35	14	36
15. Charging and Letdown Flow Shutoff and Return to Service (Loop 4 / 3)[f]	NS	75	3 / 0	7 / 8[iv]	4 / 0	11 / 8[g]
16. Loss of Charging with Prompt Return to Service (Loop 4 / 3)[f]	NS	25	64 / 0	126 / 3[g]	54 / 0	118 / 3[g]
17. Loss of Charging with Delayed Return to Service (Loop 4 / 3)[f]	NS	25	3 / 0	6 / 3[g]	3 / 0	6 / 3[g]
18. Loss of Letdown with Prompt Return to Service (Loop 4 / 3)[f]	NS	250	60 / 0	124 / 25[g]	45 / 0	94 / 25[g]
19. Loss of Letdown with Delayed Return to Service (Loop 4 / 3)[f]	NS	25	8 / 0	13 / 3[g]	10 / 0	22 / 3[g]
20. Auxiliary Spray during Plant Cooldown	NS	NS	78	146	54	102
Upset Conditions						
~~20.~~21. Loss of Load (above 15% Full Power), Turbine Trip without Reactor Trip	100	19	5	18	3	10
~~21.~~22. Loss of All Offsite Power	50	3	1	2	1	3
~~22.~~23. Partial Loss of Flow (1 RCP)	100	3	1	3	3	8
~~23.~~24. Reactor Trip from Full Power	500	88	58	100	48	83
~~24.~~25. Inadvertent RCS De-Pressurization (Resulting in Reactor Trip)	NS	20	0	5[h]	0	5[h]

Table 4.3-2 DCPP Units 1 and 2 Transient Cycle Count and 60-year Projections(i, ii)

Transient Description	Design Basis Cycles, FSAR Table 5.2-4	Limiting Analyzed Value(iii)	Unit 1		Unit 2	
			Events (1984-2008)	Projected Events for 60-Years	Events (1985-2008)	Projected Events for 60-Years
~~25.~~26. Inadvertent Auxiliary Spray (differential temperature > 320°F)	12	7	2	5	5	7
~~26.~~27. Control Rod Drop(f)	NS	80	0	5	1	2
~~27.~~28. Inadvertent ECCS Actuation(f)	NS	60	0	5	0	5
~~28.~~29. Excessive Feedwater Flow(f)	NS	30	0	1	0	1
~~29.~~30. Safety Injection into RCS Cold Leg / High Head Safety Injection	NS	97	1	4(i)	0	4(v)
~~30.~~31. Inadvertent Accumulator Blowdown(f)	NS	5	0	1	0	1
~~31.~~32. Design Earthquake (OBE)	20	20	0	1	0	1
Test Conditions						
~~32.~~33. Turbine Roll Test	10	10	5	8(vi)	6	9(j)
~~33.~~34. Primary Side Hydrostatic Test	10	5	1	2(j)	1	2(j)
~~34.~~35. Secondary Side Hydrostatic Test (each generator)	10	10	0	1	0	1
~~35.~~36. Primary Side Leak Test	60	5	0	5	0	5
~~36.~~37. Secondary Side Leak Test	10	10	0	1	0	1
~~37.~~38. Tube Leak Tests	800	800	0	See Note k	0	See Note k

LRA Section 4.3.1.2 Present and Projected Status of Monitored Locations

Projection Method

Projected cycle counts were calculated using a dual linear projection of the historical results, except as noted in Table 4.3-2. For each event, two rates were determined; a long-term rate based on the entire history (i.e., the number of cycles since plant startup), and a short term rate (i.e., the incremental cycles over the last 10 years / 10 years). These two rates were combined using a weighted average:

Projection rate = [(LTW)*(long-term rate) + (STW)*(short-term rate)] / [(LTW) + (STW)].

The values of LTW (long-term weight) and STW (short-term weight) were determined on an event~~- or component-~~specific basis to reflect the most likely future behavior of that event or component. For most transients, the projection weighted the last 10 years more heavily based on the assumption that recent (short-term) history defines a trend which will likely continue into the future. For events that occurred infrequently, the projection increased the long-term weight since the recent history may have reflected isolated incidences rather than real trends.

These projections are intended to be a best estimate of the actual cycles expected. They do not represent a revision of the design basis for the DCPP Units. The purpose is to demonstrate that the 50-year design numbers of transients are reasonable for 60 years. Future cycle count projections will be based on the actual accumulation history over the analysis period, adjusted on a component-specific basis by scaling factors to account for expected future operating conditions.

Table 3.1.2-1 Reactor Vessel, Internals, and Reactor Coolant System – Summary of Aging Management Evaluation – Reactor Vessel and Internals

Component Type	**Intended Function**	**Material**	**Environment**	**Aging Effect Requiring Management**	**Aging Management Program**	**NUREG-1801 Vol. 2 Item**	**Table 1 Item**	**Notes**
RV Nozzle Support Pads	SS	Carbon Steel	Plant Indoor Air	Cumulative Fatigue Damage	Time Limited Aging Analysis evaluated for the period of extended operation	IV.A2-20	3.1.1.01	C
RVI Core Barrel Assembly	DF, SH, SS	Stainless Steel	Reactor Coolant	Cumulative Fatigue Damage	Time Limited Aging Analysis evaluated for the period of extended operation	IV.B2-31	3.1.1.05	A
RV Core Support Lugs	SS	Nickel Alloys	Reactor Coolant	Cumulative Fatigue Damage	Time Limited Aging Analysis evaluated for the period of extended operation	IV.B2-31	3.1.1.05	C

Table 3.1.2-3 Reactor Vessel, Internals, and Reactor Coolant System – Summary of Aging Management Evaluation – Pressurizer

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
Valve Support Bracket (Unit 2 only)	SS	Carbon Steel	Plant Indoor Air	Cumulative Fatigue Damage	Time Limited Aging Analysis evaluated for the period of extended operation	III.B1.1-12	3.5.1.42	A

Table 3.1.2-4 Reactor Vessel, Internals, and Reactor Coolant System – Summary of Aging Management Evaluation – Steam Generators

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
SG Secondary Manway and Handhole Covers	PB	Nickel Alloys	Secondary Water (ext)	Cumulative Fatigue Damage	Time Limited Aging Analysis evaluated for the period of extended operation	IV.D1-21	3.1.1.06	C
SG Feedwater Ring	DF	Nickel Alloy	Secondary Water (int)	Cumulative Fatigue Damage	Time Limited Aging Analysis evaluated for the period of extended operation	IV.D1-21	3.1.1.06	C
SG Feedwater Ring	DF	Carbon Steel	Secondary Water (ext)	Cumulative Fatigue Damage	Time Limited Aging Analysis evaluated for the period of extended operation	IV.D1-11	3.1.1.07	D
SG Primary Manway Covers	PB	Stainless Steel	Reactor Coolant (ext)	Cumulative Fatigue Damage	Time Limited Aging Analysis evaluated for the period of extended operation	IV.D1-8	3.1.1.10	C

Table 4.1-1 List of TLAAs

TLAA Category	Description	Disposition Category(vii)	Section
1.	**Reactor Vessel Neutron Embrittlement Analysis**	NA	**4.2**
	Neutron Fluence Values	ii, iii	4.2.1
	Pressurized Thermal Shock	ii, iii	4.2.2
	Charpy Upper-Shelf Energy	ii	4.2.3
	Pressure - Temperature Limits	iii	4.2.4
	Low Temperature Overpressure Protection	iii	4.2.4
2.	**Metal Fatigue Analysis**	NA	**4.3**
	DCPP Fatigue Management Program	NA	4.3.1
	ASME Section III Class A Fatigue Analysis of Vessels, Piping, and Components	NA	4.3.2
	Reactor Pressure Vessel, Nozzles, and Studs	ii, iii	4.3.2.1
	Reactor Vessel Closure Head and Associated Components	i, ii	4.3.2.2
	Reactor Coolant Pump Pressure Boundary Components	i, ii, iii	4.3.2.3
	Pressurizer and Pressurizer Nozzles	i, ii, iii	4.3.2.4
	Steam Generator ASME Section III Class 1, Class 2 Secondary Side, and Feedwater Nozzle Fatigue Analyses and Fatigue Qualification Tests	i	4.3.2.5
	Absence of TLAA for Reactor Coolant System Boundary Valves	NA	4.3.2.6
	Reactor Coolant Pressure Boundary Piping	NA	4.3.2.7
	Absence of Supplemental Fatigue Analysis TLAAs in Response to Bulletin 88-08 for Intermittent Thermal Cycles due to Thermal-Cycle-Driven Interface Valve Leaks and Similar Cyclic Phenomena	NA	4.3.2.8
	Bulletin 88-11 Revised Fatigue Analysis of the Pressurizer Surge Line for Thermal Cycling and Stratification	iii	4.3.2.9

Table 4.1-1 List of TLAAs

TLAA Category	Description	Disposition Category[(vii)]	Section
	Absence of a TLAA for Thermal Embrittlement of Cast Austenitic Stainless Steel (CASS) Reactor Coolant Pumps	NA	4.3.2.10
	Absence of a Cumulative Fatigue Usage Factor TLAA to Determine High Energy Line Break (HELB) Locations	NA	4.3.2.11
	TLAAs in Fatigue Crack Growth Assessments and Fracture Mechanics Stability Analyses for Leak-Before-Break (LBB) Elimination of Dynamic Effects of Primary Loop Piping Failures	iii	4.3.2.12
	Fatigue Analyses of the Reactor Pressure Vessel Internals	iii	4.3.3
	Effects of the Reactor Coolant System Environment on Fatigue Life of Piping and Components (Generic Safety Issue 190)	ii, iii	4.3.4
	Assumed Thermal Cycle Count for Allowable Secondary Stress Range Reduction Factor in ANSI B31.1 Piping	i	4.3.5
	Fatigue Design and Analysis of Class IE Electrical Raceway Support Angle Fittings for Seismic Events	i	4.3.6

LRA Section 4.3.2.1, page 4.3-15

Disposition: ~~Validation~~Revision, 10 CFR 54.21(c)(1)(ii) and Aging Management, 10 CFR 54.21(c)(1)(iii)

~~Validation~~Revision

As shown in Table 4.3-3, the usage factors for all RPV components, with the exception of the RPV studs and core support pads, calculated in this analysis remain significantly below 1.0 (i.e., do not exceed 0.6, when projected to 60 years). All RPV components, with the exception of the RPV studs and core support pads, will be valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

LRA Section 4.3.2.2, pages 4.3-16 and 4.3-17

Disposition: ~~Validation~~Revision, 10 CFR 54.21(c)(1)(ii)

Validation - RRVCH

The Unit 1 and 2 replacement reactor vessel heads including the RRVCHs, CRDMs, CETNAs, and thermocouple nozzles will be analyzed for a 50-year design life, which will extend beyond the period of extended operation. Therefore the fatigue analyses the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

~~Validation~~ Revision - Thermocouple Column with Low Design Basis Usage Factors

The current fatigue analyses of the thermocouple column demonstrate that the maximum 40-year usage factor is 0.29. If multiplied by 1.5 (60/40) to account for the 60-year period of extended operation, these results do not exceed 0.6, providing a large margin to the code acceptance criterion of 1.0. The analyses of these components are therefore valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

LRA Section 4.3.2.3, page 4.3-20

**Disposition: Validation, 10 CFR 54.21(c)(1)(i); Revision, 10 CFR 54.21(c)(1)(ii); and Aging Management,
10 CFR 54.21(c)(1)(iii)**

Validation - Hydraulic Nuts and Studs

The Unit 1 RCP 1-2 hydraulic nuts and studs were installed in 2005 with a 50-year design life, which will extend beyond the period of extended operation. Therefore, the fatigue analyses for the hydraulic nuts and studs will remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

~~Validation~~ Revision - Thermal Barrier Flange and Main Flange Thermowell

The design basis fatigue usage in the thermal barrier flange is a negligible 0.0002. The thermal barrier flange design CUF was multiplied by 1.5 (60/40) resulting in a CUF of 0.0003 for 60 years of operation.

The design basis analysis qualified the main flange thermowell for greater than 10^6 cycles, indicating an alternating stress intensity that is less than the endurance limit. The increase in design life from 40 years to 60 years does not affect this basis for the safety determination.

Therefore, the fatigue analyses of the thermal barrier flange and main flange thermowell will remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

LRA Section 4.3.2.4, page 4.3-24

Disposition: Validation, 10 CFR 54.21(c)(1)(i); Revision, 10 CFR 54.21(c)(1)(ii); and Aging Management, 10 CFR 54.21(c)(1)(iii)

Validation - Unit 2 Relief Valve Support Bracket, Including Permitted Relief Valve Operating Cycles

The analysis of the Unit 2 relief valve support bracket determined the partial usage factor due to loads required by the design specification is much less than 0.1. Maintaining the usage factor below 1.0 is controlled by the permitted number of valve operating cycles. However, the limit is above 9,000 operations, far in excess of any expected in any foreseeable design life. The fatigue analysis of the Unit 2 relief valve support bracket is therefore valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

~~Validation~~ Revision - Pressurizer Subcomponents with Projected 60-Year Usage Factors Less Than 0.6

As shown in Table 4.3-6, the projected 60-year fatigue usage factors of some subcomponents remain significantly below 1.0 (i.e. do not exceed 0.6, when projected to 60 years). The analyses of these subcomponents are therefore valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

LRA Section 4.3.4, page 4.3-45

Disposition: ~~Validation~~Revision, 10 CFR 54.21(c)(1)(ii); and Aging Management, 10 CFR 54.21(c)(1)(iii)

~~Validation~~Revision, 10 CFR 54.21(c)(1)(ii)

As shown in Table 4.3-8, the evaluation of fatigue effects in three of the NUREG/CR-6260 locations, reactor vessel shell to lower head junction, reactor vessel inlet nozzles, and RHR line tee, has demonstrated that the EAF CUF values will remain sufficiently below 1.0, i.e., less than 0.5. If multiplied by 60/50 to account for the period of extended operation, these results do not exceed 0.6, providing a large margin to the code acceptance criterion of 1.0. The evaluation of fatigue effects in these locations has been validated and projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

A3.2.1.1 Reactor Pressure Vessel, Nozzles, and Studs

The DCPP Unit 1 reactor pressure vessel is designed to ASME Code, Section III, 1965 Edition through the Winter 1966 Addenda. The DCPP Unit 2 reactor pressure vessel is designed to ASME Section III 1968 Edition.

Pressure-retaining and support components of the reactor pressure vessel are subject to an *ASME Boiler and Pressure Vessel Code* Section III fatigue analysis. This original fatigue analysis has been updated to incorporate redefinitions of loads and design basis events, operating changes, replacement steam generators, and minor modifications using the 50-year design basis number of transients.

The usage factors for all reactor pressure vessel components, with the exception of the RPV studs and core support pads, remain below 1.0 when projected to 60 years. All RPV components, with the exception of the RPV studs and core support pads, will be valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

The Metal Fatigue of Reactor Coolant Pressure Boundary program described in Section A2.1 will ensure that the fatigue analyses for RPV studs and core support pads remain valid, or that appropriate reevaluation or other corrective measures maintain the design and licensing basis. Action limits will permit completion of corrective actions before the design basis number of events is exceeded, and before the cumulative usage factor exceeds the code limit of 1.0. Therefore, effects of fatigue in the reactor pressure vessel pressure boundary and its supports will be managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

A3.2.1.2 Reactor Vessel Closure Heads and Associated Components

The reactor pressure boundary components associated with the reactor vessel closure head are the control rod drive mechanisms (CRDM) pressure housings, core exit thermocouple nozzle assemblies (CETNAs), thermocouple nozzles, and thermocouple columns. The Units 1 and 2 CRDMs pressure housings, the CETNAs, and the thermocouple nozzles will be replaced with the replacement reactor vessel closure heads (RRVCHs). The RRVCHs, CRDM pressure housings, CETNA, and thermocouple nozzles will be designed to ASME Code, Section III. The Unit 1 and 2 RRVCHs, CRDMs, CETNAs, and thermocouple nozzles will be analyzed for a 50- year design life, and therefore will remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

The only reactor pressure boundary components associated with the reactor vessel closure head that will not be replaced are the thermocouple columns. These components were designed to the ASME Code, Section III. The current fatigue analyses of the thermocouple column demonstrate a large margin to the code acceptance criterion of 1.0. The analyses of these components are therefore valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A3.2.1.3 Reactor Coolant Pump Pressure Boundary Components

There are four Model 93A Reactor Coolant Pumps (RCPs) for each reactor (one pump per coolant loop). The RCP design reports demonstrate that the pressure components satisfy all the Class A requirements of the ASME Code, Section III, 1968 Edition through the Winter 1970 Addenda.

Thermal Barrier Flange and Main Flange Thermowell
Fatigue in the thermal barrier flange and main flange thermowell was shown to be negligible based on the low CUF and the high number of allowable cycles. Increasing the 40-year design life results from the generic stress reports by a factor of 1.5 to account for a 60-year design life would not change this determination. Therefore the fatigue analysis of the thermal barrier flange and main flange thermowell will remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

Hydraulic Nuts and Studs
Hydraulic nuts and studs were installed on Unit 1 RCP 1-2 in 2005. These components were analyzed with a 50-year design life, which will extend beyond the period of extended operation. Therefore the fatigue analyses for the hydraulic nuts and studs will remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

The Metal Fatigue of Reactor Coolant Pressure Boundary program described in Section A2.1 ensures that either the assumed numbers of design cycles or transient events used by the RCP design documents for the Locating Slot, Main Flange Bolts, and Seal Housing Penetrations and Bolts are not exceeded, or that appropriate reevaluation or other corrective action is taken if a design basis number of events is approached. Action limits will permit completion of corrective actions before the design basis number of events is exceeded, and before the cumulative usage factor exceeds the code limit of 1.0. Effects of fatigue in the reactor coolant pump pressure boundaries will be managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

A3.2.1.4 Pressurizer and Pressurizer Nozzles

The pressurizers and their integral support skirts are designed to ASME Section III, 1965 Edition, with Addenda through Summer of 1966, as ASME Section III Class A components.

Unit 2 Relief Valve Support Bracket
The analysis of the Unit 2 relief valve support bracket determined the partial usage factors due to loads required by the design specification and due to relief valve operation. Assuming an increase in usage factor due to design specification loads, proportional to the increase in licensed operating period, limits the fatigue usage available for relief valve operation. However the limiting number of valve operating cycles is far in excess of any expected in any foreseeable design life. The fatigue analysis of the Unit 2 relief valve support bracket is therefore valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

Pressurizer Subcomponents with Projected 60-Year Usage Factors Less Than 0.6
The projected 60-year fatigue usage factors of some pressurizer subcomponents remain significantly below 1.0 (i.e., do not exceed 0.6 when projected to 60 years). The analyses of these subcomponents are therefore valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A3.2.3 Effects of the Reactor Coolant System Environment on Fatigue Life of Piping and Components (Generic Safety Issue 190)

DCPP addressed Generic Safety Issue - 190 review requirements by assessing the environmental effect on fatigue at the NUREG/CR-6260 sample locations for older-vintage Westinghouse plants. NUREG/CR-6260 identifies seven sample locations for older-vintage Westinghouse plants:

- Reactor vessel shell and lower head
- Reactor vessel inlet nozzles
- Reactor vessel outlet nozzles
- Pressurizer surge line (hot leg nozzle safe end)
- Charging system nozzle
- Safety injection system nozzle
- Residual heat removal system piping.

The evaluation of fatigue effects in three of the NUREG/CR-6260 locations, reactor vessel shell to lower head junction, reactor vessel inlet nozzles, and RHR line tee, has demonstrated that the CUF values will remain sufficiently below 1.0 using the maximum applicable F_{en} values to validate them for the period of extended operation. If multiplied by 60/50 to account for the 60-year period of operation, these results do not exceed 0.6, providing a large margin to the code acceptance criterion of 1.0. The evaluation of fatigue effects in these locations has been validated and projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).