UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 20, 2010

Mr. John Conway
Senior Vice President
Generation and Chief Nuclear Officer
Pacific Gas and Electric Company
77 Beale Street, MC B32
San Francisco, CA 94105

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RELATED TO THE REVIEW OF THE DIABLO CANYON NUCLEAR POWER PLANT, UNITS 1 AND 2, LICENSE RENEWAL APPLICATION (TAC NOS. ME2896 AND ME2897)–TIME LIMITED AGING ANALYSES

Dear Mr. Conway:

By letter dated November 23, 2009, Pacific Gas & Electric Company submitted an application pursuant to Title 10 of the *Code of Federal Regulations* Part 54, to renew the operating licenses for Diablo Canyon Nuclear Power Plant, Units 1 and 2, for review by the U.S. Nuclear Regulatory Commission (NRC or the staff). The staff is reviewing the information contained in the license renewal application and has identified, in the enclosure, areas where additional information is needed to complete the review.

The request for additional information was discussed with Mr. Terry Grebel, and a mutually agreeable date for the response is within 30 days from the date of this letter. If you have any questions, please contact me at 301-415-1045 or by e-mail at nathaniel.ferrer@nrc.gov.

Sincerely,

Nathaniel Ferrer, Project Manager
Projects Branch 2
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket Nos. 50-275 and 50-323

Enclosure:
As stated

cc w/encl: Distribution via Listserv

Diablo Canyon Nuclear Power Plant, Units 1 and 2
License Renewal Application
Request for Additional Information Set 37
Time-Limited Aging Analysis

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.

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 Fatgitue 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.

RAI 4.3-5 (follow-up)

Background:

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."

Issue:

The staff seeks confirmation that the cycle numbers that were given for these transients in the response to RAI 4.3-5, 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.

Request:

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.

RAI 4.3-10 (follow-up)

Background:

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; RAI 4.3-8, and RAI 4.3-9. 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 5.2-4 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 5.2-4 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.

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 B31.1 or B31.7 requirements. The applicant also stated that the remaining piping systems listed in the RAI are designed to ASME Class 2, 3, or ANSI B31.1 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 RAI 4.3-12, request 2, clearly identifies cumulative fatigue damage as an applicable aging effect for either ASME Code Section III Class 2 or 3 or ANSI B31.1 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 III requirements for Class 2 or 3 components or to ANSI B31.1 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.

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.[1] 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

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[1] LRA AMP B2.1.5, Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of Pressurized Water Reactors Program, indicates that the schedule for replacing Unit 1 upper RV head is set for the October 2010 refueling outage; However, Commitment No. 28 in LRA FSAR Supplement Table A4-1 only commits to that the head will be replaced at some time prior to entering into the period of extended operation.

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).

RAI 4.3-14

Background: 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.

Issue:

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.

Request:

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.

RAI 4.3-15

The applicant includes its environmentally-assisted metal fatigue analyses for specific reactor coolant pressure boundary (RCPB) 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 4.3-6, for Class 1 pressurizer components, the applicant reported that some of the RV and pressurizer components had either 40-year design basis CUFs or 60-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 60-year projected CUF of 1.136 for its spray nozzles
- Pressurizer heat penetration nozzles – unit 1 is the limiting unit 50-year design basis CUF value of 2.964 and a updated 60-year projected CUF of 0.940
- RV bottom mounted instrumentation nozzles, which are nickel alloy RCPB component locations – with a 50-year design basis CUF value of 0.378 and a 60-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 RCPB 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 RCPB 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 RCPB 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 (RCPB) 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 RCPB 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.

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.

December 20, 2010

Mr. John Conway
Senior Vice President
Generation and Chief Nuclear Officer
Pacific Gas and Electric Company
77 Beale Street, MC B32
San Francisco, CA 94105

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RELATED TO THE REVIEW OF THE DIABLO CANYON NUCLEAR POWER PLANT, UNITS 1 AND 2, LICENSE RENEWAL APPLICATION (TAC NOS. ME2896 AND ME2897) – TIME LIMITED AGING ANALYSES

Dear Mr. Conway:

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The request for additional information was discussed with Mr. Terry Grebel, and a mutually agreeable date for the response is within 30 days from the date of this letter. If you have any questions, please contact me at 301-415-1045 or e-mail nathaniel.ferrer@nrc.gov.

Sincerely,
/RA/
Nathaniel Ferrer, Safety Project Manager
Projects Branch 2
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket Nos. 50-275 and 50-323

Enclosure:
As stated

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Letter to John Conway from Nathaniel Ferrer dated December 20, 2010

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