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)–AGING MANAGEMENT PROGRAM AND 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

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Diablo Canyon Nuclear Power Plant, Units 1 and 2
License Renewal Application
Request for Additional Information Set 36
Aging Management Programs/Time-Limited Aging Analysis

RAI 4.1-6

Background:
License renewal application (LRA) Section 4.3.2.6, "Absence of a TLAA for Reactor Coolant System Boundary Valves," provides the applicant's basis for its conclusion that the current licensing basis (CLB) for the safety-related valves in the reactor coolant pressure boundary (RCPB) valves does not include any analyses that need to be identified as Time-Limited Aging Analyses (TLAA) for the LRA under the TLAA identification criteria in 10 CFR 54.3. Final Safety Analysis Report (FSAR) Table 5.2-9 provides the list of applicable RCPB valves. FSAR Table 5.2-2 identifies that the applicable design codes and standards for the reactor coolant pressure boundary valves are: (1) "USAS B16.5," (2) MSS-SP-66; (3) "ASME III 68," or (4) "ASME III 74."

Additionally, the review of the CLB indicates that some of the RCPB values may have been designed to one or more of the following additional code and standards not currently reflected in FSAR Table 5.2-2: (1) ANSI B31.7 [several editions listed]; (2) ASME Boiler and Pressure Vessel Code, Section III, 1966 Edition; (3) ASME Boiler and Pressure Vessel Code, Section III, 1971 Edition, inclusive of 1973 Addenda; (4) for Target Rock Head Vent Valves, ASME III, Class II, 1977 Edition; (5) draft ASME Pump and Valve Code for Nuclear Power Plants, 1968 Edition; and (6) ASME Code Section III, 1986 Edition.

FSAR Table 5.2-2 identifies that the design code for the reactor coolant system safety valves is the 1965 Edition of ASME Code Section, III, Article 9, and that the design code for the reactor coolant system relief valves is USAS B16.5 (edition not specified).

Issue and Requests:
The information and basis in LRA Section 4.3.2.6 does not give the staff a sufficient basis for verifying that there is no need for any TLAAs to be identified in the LRA for the RPCB valves based on the following observations:

Issue 1:
FSAR Table 5.2-9 identifies the valves that are applicable to the RCPB design. The table does not identify which specific design code was used for the design, design analysis, procurement, and fabrication of each RCPB valve that was listed in the table. In addition, FSAR Table 5.2-2 only identifies the codes and standards that are applicable to the RCPB valves based on a commodity grouping, not on an individual RCPB valve basis. In addition, DCPP has not provided the staff with access to the specific design specifications that were used for the design stress analyses of the RCPB valves that are listed in FSAR Table 5.2-9. Thus, the staff is unable to verify (based on the current information) whether the design code for a given RCPB valve required a time-dependent fatigue analysis based on its design code and its nominal valve size.

ENCLOSURE

Request 1:
Clarify whether FSAR Table 5.2-9 provides a comprehensive list of all Class 1 or Class A valves in the RCPB. For each Class 1 or Class A valve in the RCPB, identify which design code or standard was designated in the owner's design specification for the valve's design stress analysis. For each valve: (1) identify whether the code used for the valve's design analysis included a cycle dependent cumulative usage factor (CUF) analysis, I_t analysis (similar to CUF except the analysis only considers cyclical stresses imposed to heat/cooldown cycles), or a maximum allowable stress reduction analysis, and if so, (2) summarize the criteria in the code that would call for a given valve to be included within the scope of the code's fatigue analysis criteria.

Issue 2:
LRA Section 4.3.2.6 states that ASME III, Article 9, did not require a time dependent analysis. FSAR Table 5.2-2 appears to appropriately indicate that the reactor coolant safety valves were procured to 1965 Edition of ASME Code Section III, Article 9. Staff review of this code has determined that the code is only applicable to the design of vessel components, and that ASME III, Article 9, is limited only to the application of the low-pressure overpressure protection (LTOP) system setpoints associated with these valves. Article 9 in this code clearly identifies that the remaining design rules and aspects for the valves are to be done in accordance with other applicable standards or codes. Thus the applicant has not provided a clear basis on which code or standard was used to perform the design stress analysis for these safety valves or whether the designated Code or Standard required either a CUF or I_t type explicit fatigue analysis or an implicit fatigue analysis (i.e., maximum allowable stress range reduction analysis, as might be required by ANSI B31.1 or B31.7).

Request 2:
Clarify which design code was used for the design stress analysis of the 6-inch nominal size reactor coolant system safety valves. Identify whether the specific code required the valve to be within the scope of a cycle dependent CUF analysis, I_t analysis (similar to CUF except the analysis only considers cyclical stresses imposed to heat/cooldown cycles), or a maximum allowable stress reduction analysis.

Issue 3:
FSAR Table 5.2-2 indicates that some of the Class 1 or Class A valves in the RCPB were procured to ASME Code Section III, 1968 Edition. However, there appears to be an inconsistency in the design basis information in that table because the design requirements in the 1968 Edition of the ASME Code Section III, Subarticle NB, appear to be limited only to vessel components and do not appear to be applicable to Class 1 or Class A valves in the RCPB. Thus, it is not evident how some of the valves in the RCPB could have been procured to ASME Code Section III 1968 Edition or which design code was used for the stress analysis of the valves and whether the code or standard for the stress analysis required a cycle dependent CUF analysis, I_t analysis (similar to CUF except the analysis only considers cyclical stresses imposed to heat/cooldown cycles), or a maximum allowable stress reduction analysis.

Request 3:
Clarify whether, consistent with the information in FSAR Table 5.2-2, any Class 1 or Class A valves in the RCPB have been designed to the design requirements (including design stress requirements and cyclical fatigue analysis requirements) in the 1968 Edition of the ASME Code

Section III. If so, justify the basis for using a vessel-related Code for the design, fabrication, analysis, and procurement of a given Class 1 or Class A valve in the RCPB, and clarify, with an explanation and justification, whether or not the cyclical metal fatigue analysis in Section N-415 of the Code would have been required to have applied as part of the stress analysis for the valves procured to this ASME Code Section III edition.

Issue 4:
The staff has determined that the some of the small bore Class 1 or Class A valves in the RCPB have been designed, fabricated, analyzed, and procured to a 1968 Draft ASME Code for Pumps and Valves for Nuclear Power Code and that Sections 452 and 454 of this Code include applicable time-dependent cyclic or fatigue assessment criteria for pumps and valves designed and procured to this code. Specifically, Section 454 of the Code has a I_t parameter metal fatigue analysis (cycling loading analysis) that is similar to the type of CUF analysis that is required for ASME Code Class 1 or Class A components in ASME Section III Article NB-3200 requirements or N-415 requirements for older versions of ASME Section III. The staff has verified that Section 142 of this Code identifies that the fatigue analysis requirements in Section 452 and 454 would need to be performed only if the inlet nozzle size for the Class 1 pump or valve was greater than 4 inches diameter nominal pipe size. However, Section 410 of the Code qualifies this somewhat by stating the Code's Chapter 4 procedures and analyses (including those in Sections 452 and 454) would need to be performed for small bore pumps or valves (i.e., for those pump or valves with inlet nozzles less than or equal to 4 inches in nominal pipe size) if the owner's design specification for a given small bore pump or valve specified this need, as determined by the owner. Thus, there could be circumstances where a small-bore pump or valve could be within the scope the Code's fatigue assessment criteria (Section 452) and cyclical loading assessment criteria (Code Section 454).

Request 4:
Clarify the review and steps that DCPP took to confirm whether or not the owner's design specification for a small bore Class 1 or Class A valve designed to the 1968 Draft ASME Code for Pumps and Valves for Nuclear Power Code had designated the valve for analysis pursuant to the Code's I_t fatigue analysis criteria. Identify all small bore Class 1 or Class A valves that were designed to the 1968 Draft ASME Code for Pumps and Valves for Nuclear Power Code and were permitted to be exempted from the I_t analysis based on the exemption criteria in Section 410 of this Code based and their nominal valve inlet size. In addition, identify all small bore Class 1 or Class A valves (if any) that were designed to this draft Code for which the owner had gone beyond the small bore fatigue exemption criteria in Section 410 of the Code and had specifically designated the time-dependent I_t analysis to be performed in the owner's design specification for a given small bore Class 1 or Class A valve.

Issue 5:
The staff has determined that FSAR Table 5.2-2 indicates that USAS B16.5 is designated as an appropriate design code for specific small bore and large Class 1 or Class A valves in the RCPB. However, the staff has noted that the scope of USAS B16.5 only is limiting to the following valve design and quality activities: (1) pressure-temperature ratings; (2) size and methods for designated openings; (3) markings; (4) minimum requirements for valve material selection; (5) valve dimensions; (6) valve tolerances; and (7) valve hydrostatic test criteria. The staff has noted that the scope of USAS B16.5 does not appear to include design stress analysis criteria for Class 1 or Class A valves in the RCPB. Thus, for a given Class 1 or Class A valve

procured to the USAS B16.5 pressure-temperature rating criteria, it is not evident which design codes (if any) were used to perform the design stress analyses for the specific valve, and if applicable, whether the code used for the design stress analysis required either a cycle-dependent CUF analysis, I_t analysis (similar to CUF except the analysis only considers cyclical stresses imposed to heat/cooldown cycles), or a maximum allowable stress reduction analysis.

Request 5:
For each Class 1 or Class A valve that was procured to USAS B16.5 pressure-temperature rating criteria, identify the code or standard (if any) that was used to perform the design stress analysis for the procured valve, and if applicable, clarify whether the design code or standard used for the stress analysis of the valve required the valve to be analyzed in accordance with either an applicable cycle-dependent CUF analysis, I_t analysis (similar to CUF except the analysis only considers cyclical stresses imposed to heat/cooldown cycles), or a maximum allowable stress reduction analysis.

Issue 6:
The staff has determined that some of the Class 1 or Class A valves in the RCPB piping subsystems were designed to either ANSI B31.1 or B31.7 design. LRA Section 4.3.5 identifies that the implicit fatigue analyses (i.e., maximum allowable stress reduction analyses) for piping, piping components, and piping elements in these subsystems are analyses that meet the definition of a TLAA in 10 CFR 54.3. The staff has determined that the scope of components in piping systems designed to ANSI B31.1 code criteria includes applicable valves in the systems. Thus, it is not evident to the staff why Class 1 or Class A valves in portions of the RCPB designed to ANSI B31.1 or B31.7 criteria would not be within the scope of the ANSI B31.1 or B31.7 stress analysis criteria or the implicit fatigue analysis criteria in these codes.

Request 6:
Identify all Class 1 or Class A valves in the RCPB that were designed to ANSI B31.1 stress analysis criteria and all Class 1 or Class A valves in the RCPB that were designed of ANSI B31.7 stress analysis criteria. For those Class 1 or Class A valves procured to these design codes, clarify, with a justified explanation, on whether the implicit fatigue analysis in these Codes are applicable to any Class 1 or Class A valves that are procured to these design code criteria, and if so, justify whether or not the implicit fatigue analyses performed on the subsystems containing the valves need to be identified as a TLAA for the DCPP LRA.

Issue 7:
The staff has determined that the applicant has indicated that, based on its current review of the CLB, there were some small bore Class 1 or Class A valves (less than or equal to 4 inches nominal size) in the RCPB where the applicant could not determine which the design code or standard was used for the design, analysis, fabrication, and procurement of the valves, but where the applicant indicated there would not be any associated fatigue-related TLAAs based on their size. Presumably, these valves are valves in the RCPB and Safety and Seismic Class 1 valves. Thus, pursuant to 10 CFR Part 50, Appendix B, Criterion III, Design Control, the NRC would have required that these valves be within the scope of appropriate design standards (or Codes). Thus, it is not evident to the staff why these valves would not have been required to be within the scope of applicable design codes or standards, including those governing the stress analyses for such valves. Thus, without further clarification, the staff cannot determine whether these valves were procured to appropriate design codes or standards, and if so, whether the

given code or standard for a valve in this category would have required the valve to be analyzed in accordance with either a cycle-dependent CUF analysis, I_t analysis (similar to CUF except the analysis only considers cyclical stresses imposed to heat/cooldown cycles), or a maximum allowable stress reduction analysis.

Request 7:
Identify the design codes or standards that were used for the design of these valves in order to comply with the provision in 10 CFR Part 50, Appendix B, Criterion III, Design Control, that states that the design measures shall include: "provisions to assure that appropriate design standards are specified and included in design documents..." For each valve in this category, identify whether the design code or standard used (if any) for the design stress analysis of the valve required either a cycle-dependent CUF analysis, I_t analysis (similar to CUF except the analysis only considers cyclical stresses imposed to heat/cooldown cycles), or a maximum allowable stress reduction analysis. Justify your basis for concluding that the CLB does not include any fatigue related analyses in the CLB that meet the definition of a TLAA in 10 CFR 54.3.

RAI 4.1-7

The applicant includes its TLAA for the reactor vessel internal (RVI) core support structure components in LRA Section 4.3.3. The applicant stated that the CUF analysis for the baffle-former bolts originally calculated a CUF value less than the design limit of 1.0. The applicant also stated that the adequacy of the baffle-former bolt is an industry issue and the design analyses and evaluations may not currently be sufficient to support their initial safety determination. The applicant stated that the baffle-former bolt analyses will be addressed by participation in industry level initiatives.

The applicant is currently an active member of the Electric Power Research Institute's Materials Reliability Program (EPRI MRP) and for license renewal, the applicant has committed to participating in the EPRI MRP activities to ensure the structural integrity of Westinghouse designed RVI components, including Westinghouse designed baffle bolts and former bolts. Specifically, the staff confirmed that the applicant's FSAR Supplement Table Commitment No. 22, Part B, states the following:

> B. For Reactor Vessel Internals:
>
> (1) Participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, PG&E will submit an inspection plan for reactor internals to the NRC for review and approval.

The current industry-wide program for Westinghouse designed facilities is defined in MRP-227 and that the industry-wide initiatives include appropriate measures to perform ultrasonic testing (UT) inspections of Westinghouse baffle and former bolts for evidence of either stress induced or fatigue induced cracking. The staff noted that this is consistent with Standard Review Plan-License Renewal (SRP-LR) Sections 3.1.2.2.15 and 3.1.2.2.17.

Issue:
The staff noted that the applicant is taking the position that the CUF calculation for the baffle bolts no longer serves a safety basis and the CUF analysis for the bolts does not need to be identified as a TLAA because the analysis did not meet 10 CFR 54.3, Criterion 4. The staff noted that the CUF calculation for the baffle and former bolts was required to meet the 1968 Edition of the ASME Code Section III, Article NG CUF calculation requirements for core support components in the reactor vessel. The staff further noted that the the fact that the EPRI MRP is currently investigating industry initiatives to inspect for cracking in these components, does not invalidate the applicant's CLB or design basis that required a CUF calculation for these components.

Request:
Explain why the performance of the required design CUF calculation for the baffle bolts does not satisfy 10 CFR 54.3, Criterion 4, the analysis was used in a safety basis decision. Justify why the CUF analysis for the baffle bolts does not need to be identified as TLAA for the LRA.

RAI B2.1.21-1 (follow-up)

Background:
Generic Aging Lessons Learned (GALL) aging management program (AMP) XI.M.37 program element "acceptance criteria" states, in part:

> The acceptance criteria will be technically justified to provide an adequate margin of safety to ensure that the integrity of the reactor coolant system pressure boundary is maintained. The acceptance criteria will include allowances for factors such as instrument uncertainty, uncertainties in wear scar geometry, and other potential inaccuracies, as applicable, to the inspection methodology chosen for use in the program. Acceptance criteria different from those previously documented in NRC acceptance letters for the applicant's response to Bulletin 88-09 and amendments thereto should be justified.

In LRA Section B2.1.21, "Flux Thimble Tube Inspection" Program, the applicant states that its program is an existing program that is consistent with the recommended program element criteria in GALL AMP XI.M.37, "Flux Thimble Tube Inspection." By letter dated July 14, 2010, the staff issued request for additional information (RAI) B2.1.21-1, and requested that the applicant clarify its basis for the AMP's through-wall wear acceptance criterion and clarify how sources of instrument measurement and wear scar uncertainties and inaccuracies are accounted for in the AMP, as recommended in both GALL AMP XI.M37 and NRC Bulletin 88-09. In its response dated August 12, 2010, the applicant stated that the AMP's current through-wall wear acceptance criterion basis was established in the February 1991 revised inspection procedure for the AMP, which set the acceptance criterion at 68% of the nominal thimble tube wall thickness. However, the applicant also explained that the updated procedure eliminated the application of the applicant's prior 10% uncertainty adjustment on the nondestructive examination (NDE) estimate, as was made based on the applicant's steps to confirm the accuracy of the program's NDE testing methods in during Unit 1 refueling outage (RO) 1R4 and the applicant's review of the generic Westinghouse acceptance criteria bases in Westinghouse Proprietary Class 2 Report WCAP-12866,[1] which was issued in January 1991.

The applicant's response letter of August 12, 2010, indicates that the applicant eliminated application or accounting for any source of measurement uncertainty and wear rate estimation uncertainty in the program elements for the AMP. However, Westinghouse Class 2 Proprietary Report review of WCAP-12866 does include an appropriate allowance for the wall thickness acceptance criterion that is recommended in the generic report, and this appears to satisfy the need to account for appropriate uncertainties generic flux thimble tube program report, as recommended in NRC Bulletin 88-09 and in GALL AMP XI.M37.

Issue:
The staff has determined that the current DCPP Flux Thimble Tube Program does not include any uncertainty allowances in the program, even though the applicant has set acceptance criterion for the AMP to a value that is more conservative than that recommended for these types of programs in the Westinghouse report. This does not appear to conform to recommendation in either NRC Bulletin 88-09 or in the "monitoring and trending" program element of GALL AMP XI.M37 which state that these types of programs should include appropriate allowances for instrument measurement and wear scar uncertainties. In addition, DCPP's elimination of appropriate instrument measurement and wear scar uncertainties from the scope of the AMP may be non-conservative when taken in light of the flux thimble tube wear data for Unit 2 thimble tube L13, as obtained from eddy current inspections of the tube during Unit 2 ROs 2R11, 2 R12, and 2R13, and in light of the fact that this tube leaked within 4 months of returning to power operations out of RO 2R13. Specifically, the wear data obtained from the inspections of Unit 2 tube L13 indicate that the wear in the tube might have been occurring at an increasingly non-linear rate. Thus, the staff finds that the applicant's decision to eliminate appropriate instrument measurement uncertainties and wear scar uncertainties for the scope of the AMP is out of conformance of the recommendations of the applicable NRC bulletin and GALL AMP, and may not be conservative relative to relevant thimble tube operating experience for the facility.

Request:
In light of the respective operating experience for Unit 2 thimble tube 2L13, justify the basis for not including an appropriate margin term to account for NDE measurement and wear scar uncertainties in either the wear projection basis for the AMP or accounting for them in the acceptance criterion for the AMP, and provide a basis for not identifying this as an appropriate exception to the "acceptance criteria" program element in GALL AMP XI.M37, "Flux Thimble Tube Inspection."

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[1] The staff notes that the WCAP-12866 is a Class 2 Proprietary Westinghouse report which the WOG has not requested to be formally reviewed by the staff nor has the staff formally endorsed this report for use. Staff's discussion and observations on this report, as noted here, are necessarily limited and exclude specifics.

RAI B2.1.21-2 (follow-up)

Background:
The GALL AMP XI.M37 program element, "monitoring and trending," states, "[t]he wall thickness measurements will be trended and wear rates will be calculated. Examination frequency will be based upon wear predictions that have been technically justified as providing conservative estimates of flux thimble tube wear." The GALL AMP recommends that the "interval between inspections" should be established "such that no flux thimble tube is predicted to incur wear that exceeds the established acceptance criteria before the next inspection."

The "operating experience" (OE) program element for the Flux Thimble Tube Program discussed the impacts of a leak that occurred in thimble tube L13 of DCPP Unit 2 in 2006. This leak occurred at normal operating pressure with no prior warning or expectation, and occurred within four months of returning to power operations out of Unit 2 RO 2R13 and repositioning corrective actions that were implemented on that tube during the RO. In its August 12, 2010, response to RAI B2.1.21-3, the applicant added a "License Renewal Commitment" to preclude repositioning a tube more than once (without capping or replacing).

Issue:
The OE and related observations on plant-specific wear rate projections do not conform to, or meet the intent of, the GALL AMP "monitoring and trending" program element. As noted in RAI B2.1.21-2, issued by letter dated July 14, 2010, the "incremental wear" and "cumulative wear" projection methods as implemented in the applicant's AMP do not provide conservative wear projection because they do not account for possible accelerating wear nor do they account for uncertainty in the method of wear projection. Neither the OE discussion (for L13 event in 2006) nor the applicant's response to RAI B2.1.21-3 identified the apparent cause (aging mechanism) of the degradation in Unit 2 thimble tube L13, or explained why the leak occurred so soon after returning to power operations, even after indicating repositioning (corrective action) of the tube during RO 2R13.

The wear history of several flux thimble tubes, including Unit 2 thimble tube L13, indicates that the wear in the tubes may be occurring at an increasingly accelerated wear rate, and in other instances, repositioning of the tubes appears to have moderated the wear rate increase. The applicant has not addressed whether cracking could have been a main contributing factor in the rapid-time failure of Unit 2 thimble tube L13. Thus, multiple repositioning of tube L13 may not be the only feasible explanation for the rapid failure in the tube, and the staff is concerned that either rapidly progressing wear, rapidly propagating cracking, or rapidly propagating wear coupled to cracking may have been the main contributing factor for the leak in Unit 2 thimble tube L13 during Unit 2 operating cycle 14.

Request:

1. Identify the quality activities that DCPP takes to identify and confirm the apparent cause of age-related degradation that is detected in a DCPP flux thimble tube, and identify all age-related degradation effects and mechanisms (including any cracking and its mechanisms, if applicable) that have been detected in the DCPP flux thimble tubes to date.

2. Describe how the trending of thimble tube wear rates accounts for the possibility of a non-linear or accelerating wear rate.

3. Identify all aging effects and mechanisms that contributed to the degradation in Unit 2 flux thimble tube L13 over time (i.e., as detected during ROs 2R11, 2R12, and 2R13) and discuss the failure analysis activities that were performed at the site or were contracted out to confirm the apparent cause of the degradation that had occurred in the tube and the rapid progression of the degradation mechanism that lead to the relative rapid leak in 2006 (i.e., the leak occurred within four months of returning to power).

4. Provide your basis for concluding that the "monitoring and trending" activities, "acceptance criteria" and "corrective action" criteria for the Flux Thimble Tube Program will be capable of detecting degradation in a flux thimble prior to the occurrence of a through-wall failure.

5. If aging effects other than wear were determined to have occurred in tube L13 or any other thimble tube, describe how these other aging effects will be managed by the Flux Thimble Tube Program.

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

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Sincerely,
/RA/
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

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

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) – AGING MANAGEMENT PROGRAM AND TIME LIMITED AGING ANALYSES

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