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

August 30, 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 REVIEW 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 21
Aging Management Review and Time Limited Aging Analyses

RAI 3.1.2.3.2-2

In license renewal application (LRA) Table 3.1.2-2, the applicant states that for calcium silicate insulation exposed to borated water leakage (external) there is no aging effect and no aging management program (AMP) proposed. The line item cites generic Note F. The Generic Aging Lessons Learned (GALL) Report does not have any line items associated with calcium silicate insulation.

The staff notes that calcium silicate insulation is easily damaged by water and because of this is typically provided with protective jacketing. The staff also notes that based on a search of the LRA, there are no line items for insulation jacketing.

Request:

1. Clarify whether the calcium silicate insulation is protected by jacketing. If it is, state how the seams in the jacketing are controlled to mitigate leakage.

2. State what program manages the aging effect of deterioration of the insulation by borated water leakage.

RAI 3.3.2.3.5-1

Background:
In LRA Tables 3.3.2-5 and 3.4.2.1, the applicant states that carbon steel indicators, sight glasses, piping, pumps, tanks, and valves internally exposed to sodium hydroxide (NaOH) are being managed for loss of material by the Water Chemistry and One-Time Inspection Programs. The line items cite generic Note G and a plant-specific note, which states "[t]he use of carbon steel up to 200°F (93°C) and 50 wt percent NaOH is common in industrial applications with no special consideration for aging. The NaOH concentration is controlled by the Water Chemistry Program." None of these carbon steel components exposed to NaOH are being managed for cracking due to stress-corrosion cracking (SCC).

According to the 2006 edition of The American Society for Metals (ASM) Handbook, Volume 13C, corrosion of carbon steels is expected when exposed to NaOH, but corrosion rates are generally acceptable for up to a 50% NaOH solution at temperatures up to approximately 150°F. The ASM Handbook also states that carbon steels under tensile stress can experience SCC. The ASM Handbook further states that SCC generally does not occur in carbon steels exposed to a 50% NaOH solution at temperatures below 150°F, but has occurred as low as 118°F.

GALL AMP XI.M2, "Water Chemistry," recommends that primary and secondary water chemistry be monitored and controlled in accordance with industry guidelines, such as Electric Power Research Institute (EPRI) TR-105714, *PWR Primary Water Chemistry Guidelines*, Rev. 3 and TR-102134, *PWR Secondary Water Chemistry Guideline*, Rev. 3 or later revisions. The applicant's Water Chemistry Program states that its primary water chemistry program is consistent with the guidelines of EPRI TR-105714, Revision 6; and its secondary water chemistry program is consistent with the guidelines of EPRI TR-102134, Revision 7. EPRI primary and secondary water chemistry guidelines include guidance for limiting the concentration of NaOH in reactor water because it is a contaminant that can cause SCC, but they do not include control parameters for concentrated NaOH used in auxiliary systems. GALL AMP XI.M2, "Water Chemistry," is a mitigation program that relies on the control of contaminants in primary and secondary water to prevent aging and does not include any activities to detect the effects of aging. GALL AMP XI.32, "One-Time Inspection," is an inspection program used to confirm the absence of an aging effect or confirm that the effect is occurring slowly enough such that it will not affect the intended function of the component during the period of extended operation.

Issue:
Neither the EPRI water chemistry guidelines nor the applicant's description of its Water Chemistry Program include the parameters that would be monitored and controlled in order to minimize corrosion and SCC due to exposure to NaOH. It is not clear to the staff what parameters are being monitored or the acceptance criteria that have been established in order to manage aging for these components exposed to NaOH.

It is not clear to the staff why SCC is not an applicable aging affect for these carbon steel components exposed to NaOH; given that SCC can occur in carbon steel at temperatures as low as 118°F. Additionally, threaded and flanged connections that experience leakage can allow the NaOH solution to concentrate, leading to SCC. It is not clear to the staff how these components are being adequately managed for aging by the One-Time Inspection Program given that SCC could be expected to occur in carbon steel components exposed to NaOH.

Request:

1. Describe the parameters (e.g., temperature, concentration) being monitored and acceptance criteria for the NaOH solution being monitored by the Water Chemistry Program.

2. Describe whether carbon steel components exposed to NaOH contain threaded or flanged connections or are exposed to temperatures greater than 118°F. If either of the above conditions exists, provide justification for why the carbon steel components exposed to NaOH do not need to be managed for SCC, or provide additional information on how SCC of the components will be managed during the period of extended operation.

RAI 3.3.2.3.11-1

GALL Report Section IX.F, "Significant Aging Mechanisms," states that elastomer degradation may include cracking, crazing, fatigue breakdown, abrasion, chemical attacks, and weathering. GALL Report IX.E, "Aging Effects," states that hardening and loss of strength can result from elastomer degradation of seals and other elastomeric components. LRA Section B2.1.20, External Surfaces Monitoring Program, states "[w]hen appropriate for the component configuration and material, physical manipulation of elastomers is used to augment visual inspections to confirm the absence of hardening or loss of strength." In LRA Tables 3.3.2-11 and 3.3.2-17 two line items associated with flexible connectors and caulking and sealant cite generic Note G and manage the aging effect of hardening and loss of strength with the External Surfaces Monitoring Program.

The LRA External Surfaces Monitoring Program does not state that elastomers will be inspected for cracking, and changes in surface condition.

Confirm that the External Surfaces Monitoring Program inspects elastomers for cracking and changes in surface conditions or justify why the program is acceptable to manage the aging effects of hardening and loss of strength.

RAI 4.1-1

Background:
In LRA Section 4.7.3, the applicant identifies that the generic fatigue flaw growth analysis for reactor vessel (RV) under-clad cracking in Westinghouse Topical Report No. WCAP-15338-A is not an analysis that meets the definition of a time-limited aging analysis (TLAA) in 10 CFR 54.3.

Issue 1:
In LRA Section 4.7.3, the applicant states that the analysis in WCAP-15338-A qualifies reactor pressure vessels (RPVs) for the 60-year operating period rather than the current licensed operating period (40 years), and based on this, the flaw growth analysis for under-clad cracks in low-alloy steel RV forgings is not a TLAA under 10 CFR 54.3(a)(3) criterion.

Request 1:
Clarify whether WCAP-15338-A is being credited in the current licensing basis (CLB) for analysis of under-clad flaws in the SA-508 forging materials used to make the RVs. If the WCAP analysis is not being relied upon in the CLB and the report needs to be credited for aging management during the period of extended operation, justify why the generic flaw growth analysis in WCAP-15338-A has not been identified as a TLAA.

Issue 2:
Non-proprietary Westinghouse Topical Report No. WCAP-15338-A provides a fracture toughness and flaw growth analysis for under-clad cracks that are postulated in the internal cladding of SA-508 Class 2 or 3 alloy steel components in Westinghouse-design reactor pressure vessels (RPVs). The flaw growth analysis in the WCAP is a generic TLAA for Westinghouse reactors that credit the report to manage under-clad cracking in their SA-508, Class 2 or 3 RPV forging components. The staff accepted the fracture toughness and flaw growth analyses in WCAP-15338 in a safety evaluation (SE) to the Westinghouse Owners Group (WOG) dated October 15, 2001. In this SE, the staff required license renewal applicants relying on the WCAP's generic methodology to respond to the following license renewal applicant action items (LRAAIs):

(1) "The license renewal applicant is to verify that its plant is bounded by the

WCAP-15338-A report (i.e., the number of design cycles and transients assumed in the WCAP-15338-A analysis bounds the number of cycles for 60 years of operation of its reactor pressure vessel)."

(2) "Section 54.21(d) of 10 CFR requires that an FSAR supplement for the facility contains a summary description of the programs and activities for managing the effects of aging and the evaluation of TLAA for the period of extended operation."

Request 2:
If in response to Request 1, the generic flaw growth in WCAP-15338-A needs to be identified as a TLAA for RV under-clad cracking, provide responses to the LRAAIs.

RAI 4.1-2

Background:
In LRA Table 4.1-1, TLAA Category 6, "Plant-specific Time-limited Aging Analyses," the applicant states that reactor coolant pump (RCP) flywheel fatigue crack growth analysis is not a TLAA. In LRA Section 4.7.4 the applicant further states that an evaluation of the probability of failure over the period of extended operation for all operating Westinghouse plants was performed in WCAP-14535-A, "Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination," November 1996. The applicant noted that the evaluation demonstrates that the flywheel design has a high structural reliability with a very high flaw tolerance and negligible flaw crack extension over a 60-year service life (assuming 6000 pump starts).

Issue:
Based on the evaluation in WCAP-14535-A the applicant concluded that since the flaw tolerance evaluation is based on the 60-year operating period rather than the current licensed operating period (40 years), it is not a TLAA under 10 CFR 54.3 Criterion 3.

Request:
Clarify whether a 60-year flaw growth analysis in WCAP-14535-A for the RCP flywheel discs is being relied upon in the CLB to support the inservice inspection interval for the RCP flywheels. If the WCAP analysis is not being relied upon in the CLB and the report needs to be credited for aging management during the period of extended operation, justify why the generic flaw growth analysis in WCAP-14535-A does not need to be identified as a TLAA for the LRA.

August 30, 2010

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NAME	IKing	NFerrer	DWrona	NFerrer
DATE	08/26/10	08/26/10	08/26/10	08/30/10

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Letter to John Conway from Nathaniel B. Ferrer dated August 30, 2010

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