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NUCLEAR REGULATORY COMMISSION
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July 20, 2010

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) – REACTOR VESSEL INTERNALS

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, 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

cc w/encl: Distribution via Listserv

Diablo Canyon Nuclear Power Plant, Units 1 and 2
License Renewal Application
Request for Additional Information Set 11
Reactor Vessel Internals

RAI 3.1.2.2.3-1

In license renewal application (LRA) Table 3.1.2-1, for component type "RV (Reactor Vessel) Nozzles (Inlet/Outlet Nozzles)," material "Carbon Steel with Stainless Steel Cladding," environment "reactor coolant (int)," and aging effect "loss of fracture toughness," there are two line items. For the first line item, aligned with GALL Report item IV.A2.16, the aging management program (AMP) is "Time-Limited Aging Analysis (TLAA) evaluated for the period of extended operation." For the second line item, aligned with GALL Report item IV.A2.17, the AMP is "Reactor Vessel Surveillance (B2.1.15)." The first line item aligns with LRA Table 3.1.1, item 3.1.1.17, which states, in the AMP column:

> TLAA, evaluated in accordance with Appendix G of 10 CFR Part 50 and RG 1.99. The applicant may choose to demonstrate that the materials of the nozzles are not controlling for the TLAA evaluations.

The second line item aligns with LRA Table 3.1.1, item 3.1.1.18, which states, under the AMP column, "Reactor Vessel Surveillance Program (B2.1.15)." Further evaluation is recommended for both Table 3.1.1 items.

However, the neutron embrittlement TLAA evaluations, described in LRA Section 4.2, do not discuss how the nozzles were demonstrated not to be controlling, nor are any nozzle materials included among the extended beltline materials that are included in the pressurized thermal shock (PTS) TLAA evaluation (LRA Section 4.2.2) and the Upper Shelf Energy (USE) evaluation (LRA Section 4.2.3). Further, the RV Surveillance Program description does not address any nozzle materials.

1. Describe how the Diablo Canyon Power Plant (DCPP) RV nozzle materials were demonstrated to not be controlling with respect to the neutron embrittlement related TLAAs. This description should include a summary of any fluence evaluations and/or neutron embrittlement projections using the nozzle fluence and chemistry to project the PTS reference temperature (RT_{PTS}) and USE of the nozzle materials.

2. Indicate any changes this may have to the LRA.

3. Describe how the RV Surveillance Program manage loss of fracture toughness of the RV nozzles if specimens of the nozzle materials are not included in the surveillance program.

RAI 3.1.2.2.7-1

LRA Section 3.1.2.2.7 and LRA Table 3.1.2 indicate that material of the RV flange O-ring leak monitoring line is nickel alloy. However, in most pressurized-water reactors (PWRs) only the vessel penetration is nickel alloy while the adjoining piping is stainless steel.

Clarify whether the adjoining piping of the RV flange O-ring leak monitoring line is stainless steel. If so, clarify whether this piping is included within the scope of license renewal such as under another Table 2 line item such as Class 1 Piping <= 4in, GALL Report item IV.C2-1 in LRA Table 3.1.2-2.

RAI 3.1.2.2.7-2

In LRA Section 3.1.2.2.7.1, the applicant indicated that for managing aging due to stress corrosion cracking of stainless steel high pressure conduits (flux thimble guide-tubes-to seal table) exposed to reactor coolant, the applicant's Water Chemistry (B2.1.2) AMP will be augmented by their American Society of Mechanical Engineers (ASME) Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1) AMP. For stainless steel flux thimble tubes exposed to reactor coolant, cracking due to stress corrosion cracking (SCC) is managed by the DCPP Water Chemistry (B2.1.2) AMP. The staff notes that in LRA Table 3.1.2, the flux thimble tubes are included as a subcomponent of the RV Bottom Mounted Instrument Guide Tube, which aligns to GALL Report item IV.A2-1(RP-13) for the aging effect cracking.

The staff reviewed LRA Section 3.1.2.2.7.1 against the criteria in NUREG-1800, "The Standard Review Plan for Review of License Renewal Applications (SRP-LR)," Section 3.1.2.2.7.1, which states cracking due to SCC could occur in the PWR stainless steel reactor vessel flange leak detection lines and bottom-mounted instrument guide tubes. The GALL Report recommends further evaluation to ensure that these aging effects are adequately managed. The SRP-LR further states that the GALL Report recommends that a plant-specific AMP be evaluated because existing programs may not be capable of mitigating or detecting cracking due to SCC. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of the SPR-LR).

Branch Technical Position RLSB-1 states that a plant-specific AMP should include a "detection of aging effects" program element. The DCPP Water Chemistry Program provides mitigation of cracking through control of impurities, but does not provide for detection of aging effects. The ASME Section XI Inservice Inspection Program, Subsections IWB, IWC, and IWD AMP, provides for inspections of components. The standard examination requirements for flux thimble tubes under the ASME Section XI Inservice Inspection Program, Subsections IWB, IWC, and IWD AMP, is a VT-2 visual inspection per ASME Code Section XI, Table IWB-2500-1, Examination Category B-P, which would not generally be capable of detecting cracking unless a leak is already present, producing visible water and/or boric acid. The program description in LRA Section B.2.1.1 does not describe any augmented inspections for the flux thimble tubes which would be capable of early detection of cracking.

1. Identify any specific examinations included in the ASME Section XI Inservice Inspection Program, Subsections IWB, IWC, and IWD AMP, which would be capable of detecting cracking in the RV Bottom Mounted Instrument Guide Tubes (High Pressure Conduits, Seals) before a throughwall crack and leakage occurs.

2. If the ASME Section XI Inservice Inspection Program, Subsections IWB, IWC, and IWD AMP does not provide for detection of cracking prior to a leak, provide a plant-specific AMP or combination of existing AMPs that include a "detection of aging effect" program element for managing the aging effect of cracking due to SCC in the RV Bottom Mounted Instrument Guide Tubes (High Pressure Conduits, Seals); and

3. Describe what examination techniques will be used to detect (or confirm the absence of) the aging effect of cracking in the RV Bottom Mounted Instrument Guide Tubes (High Pressure Conduits, Seals), either as part of the ASME Section XI Inservice Inspection Program, Subsections IWB, IWC, and IWD, or an additional plant-specific program.

RAI 4.2.2-1

As part of its independent evaluation of the RT_{PTS} values, the staff checked the copper, nickel, and initial (unirradiated) reference temperature (RT_{NDT}) values against the corresponding values from the Reactor Vessel Integrity Database (RVID) for each beltline material. The applicant provided lower copper values for DCPP, Unit 1 intermediate shell plates B4106-1 (Heat No. C2884-1), B4106-2 (Heat No. C2854-2), and B4106-3 (Heat No. C2793-1), and lower nickel values for plates B4106-3 and B4107-3 (Heat No. C3131-1), than the corresponding copper and nickel values in RVID. The copper and nickel values provided in the LRA for plate B4106-3 match the best estimate values provided in the most recent surveillance capsule report for DCPP, Unit 1, and are therefore acceptable.

Provide data for the copper and nickel content for DCPP, Unit 1 Intermediate Shell Plates B4106-1, B4106-2, and B4107-3 as given in LRA Tables 4.2-4 and 4.2-6.

RAI 4.2.2-2

When using 10 CFR 50.61 to calculate RT_{PTS}, the margin term is defined as

Margin = $2\sqrt{(\sigma_u^2 + \sigma_\Delta^2)}$

10 CFR 50.61, defines σ_U as the standard deviation for the initial RT_{NDT}. If a measured value of initial RT_{NDT} for the material in question is available, σ_I is to be estimated from the precision of the test method. If not, and generic mean values for that class of material are used, σ_I is the standard deviation obtained from the set of data used to establish the mean.

The staff performed preliminary confirmatory calculations of RT_{PTS} using the σ_Δ term as defined in 10 CFR 50.61. To obtain the margin term for some of the materials listed in LRA Tables 4.2-4 and 4.2-5, it appears that a value of 17°F was used for σ_I, while a σ_I value of 0°F was used for other materials. In some cases, the values used for σ_I appear to be inconsistent with values in RVID.

For the materials listed in LRA Tables 4.2-4 and 4.2-5:

1. Identify whether the initial (unirradiated) $RT_{NDT(u)}$ value cited for the material is based on plant-specific data or on a generic value.

2. For those materials using a generic initial $RT_{NDT(u)}$ value, describe the method of determination and basis for the initial (unirradiated) $RT_{NDT(u)}$ values for the materials listed in LRA Tables 4.2-4 and 4.2-5, and provide a reference for each value.

3. For those materials in LRA Tables 4.2-4 and 4.2-5 using a generic value for initial RT_{NDT}, provide the σ_I value used and the basis for the σ_I value.

4. Confirm that the appropriate value of σ_I (0°F vs. 17°F) has been used for all materials listed in LRA Table 4.2-4 and 4.2-5.

RAI 4.2.2-3

For DCPP, Unit 2, the applicant stated that in accordance with Regulatory Guide 1.99 Revision 2, the Charpy V-Notch Upper Shelf Energy (C_V USE) data from Unit 2 Surveillance Capsule V were deemed credible for intermediate shell plate B5454-1 (Heat No. C5161-1). The applicant stated that the C_V USE values were projected to 54 effective full power years (EFPY) of operation using Regulatory Guide 1.99 Position 1.2. If credible surveillance data are available, Regulatory Guide 1.99 Revision 2 Position 2.2 recommends that the decrease in upper-shelf energy may be obtained by plotting the reduced plant surveillance data on Figure 2 of the regulatory guide and fitting the data with a line drawn parallel to the existing lines as the upper bound of all the data, and that this line should be used in preference to the existing graph. The staff notes that the most recent surveillance capsule report for DCPP, Unit 2 (Reference 1), indicated that the surveillance data for Plate B5454-1 were credible for both the ΔRT_{NDT} and USE, and that the surveillance data were used in the end of license extension (EOLE) projection of RT_{PTS} for this material. The EOLE C_V USE values for the Unit 2 beltline and extended beltline materials are provided in LRA Table 4.2-7.

Although surveillance data for intermediate shell plate B5454-1 are available, it was not clear whether the surveillance C_v USE data were used in the projection for that plate.

Clarify if surveillance program data used in the projection of USE for DCPP Unit 2 intermediate shell plate B5454-1 (Heat No. C5161-1). If surveillance data was not used, provide justification.

RAI B2.1.15-1

In its description of the Reactor Vessel Materials Surveillance Program in LRA Section B2.1.15, the applicant stated that for DCPP Unit 1, the last capsule is expected to be withdrawn during the current operating term after it has accumulated a fluence equivalent to 60 years of operation. The applicant further stated that the remaining five standby capsules have low lead factors, will remain inside the vessel throughout the vessel lifetime, and will be available for future testing.

The latest surveillance capsule withdrawal schedule was submitted by the applicant by letter dated March 12, 2008 (Ref. 4 of the LRA), and approved by a staff safety evaluation dated September 24, 2008 (Ref. 5 of the LRA). This schedule proposed that capsule B, with a lead factor of 3.46, will be withdrawn at 21.9 EFPY. The fluence will be equivalent to 75.8 EFPY which is between one and two times the vessel EOLE fluence (54 EFPY fluence). This meets the ASTM E 185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," criterion. The proposed schedule shows four capsules to be left in the vessel, all with lead factors around 1.3.

Clarify whether four or five surveillance capsules will remain installed in the vessel during the period of extended operation.

July 20, 2010

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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 July 20, 2010

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