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

August 26, 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 AND AGING MANAGEMENT PROGRAMS

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, Units1 and 2
License Renewal Application
Request for Additional Information Set 17
Time Limited Aging Analyses/Aging Management Programs

RAI 4.7.2-1

In license renewal application (LRA) Section 4.7.2, within the "Pressurizer" section, the applicant states that the fatigue crack growth analyses were projected to the end of the period of extended operation and are therefore valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

1. Discuss how the actual plant transient cycles are monitored to ensure that they are bounded by the number assumed in the fatigue crack growth analysis.

2. Discuss the transient cycles used in the crack growth analyses, including the number of cycles.

RAI 4.7.2-2

In LRA Section 4.7.2, within the "Pressurizer" section, the applicant states that "[n]o base-metal corrosion analyses exist for the pressurizers, since no half-nozzle or similar repairs have exposed the base metal to reactor coolant." The applicant also states that "[t]he Unit 1 pressurizer and its nozzles and safe ends contain no Alloy 600 or Alloy 82/182 weld material." The above statements are not clear regarding whether the half nozzle method was used in repairing heater sleeves in the pressurizer in both units.

1. For each unit, list all the pressurizer nozzles (e.g., pressurizer safety valve nozzle and heater sleeve nozzle). Identify the materials used to fabricate the nozzles. If a nozzle is welded to a safe end, identify the material of the safe end.

2. Discuss whether a fatigue crack growth calculation was performed for the remnant Alloy 82/182 welds. If so, discuss how the transient cycles used in the fatigue crack growth calculation are monitored to ensure they bound the actual plant cycles. If no fatigue crack growth calculation was performed, justify the structural integrity of the pressurizer shell.

3. Discuss any flaws that remained in service in the heater sleeves and in the attachment welds in both units. If so, discuss how these flaws are monitored and evaluated for the period of extended operation.

RAI 4.7.2-3

Discuss whether reactor vessel internals contain any nickel-based Alloy 600 components or nickel-based Alloy 82/182 welds. If so, discuss how these components are monitored for primary water stress corrosion cracking.

ENCLOSURE

RAI 4.7.2-4

In LRA Section 4.7.2, within the "Steam Generators" section, the applicant states that "[r]eplacement steam generators contain no Alloy 600 components or Alloy 82/182 welds."

1. Identify the material specification of the welds that join the replacement steam generator nozzles to the piping.

2. Identify the material specification of the safe ends that are welded to the steam generator nozzles.

RAI 4.7.5-1

In LRA Section 4.7.5, within the "Unit 2 RHR Piping Weld RB-119-11" section, the applicant states that "[t]he DCPP licensing basis assumes 250 heatups and 250 cooldowns for a 50-year plant life."

1. Discuss why only heatup and shutdown cycles are applied for flaw evaluation of weld RB-119-11 in the June 6, 2006 letter, but other transient cycles such as seismic, temperature, and pressure were not mentioned in the flaw evaluation for weld RB-119-11.

2. It is not clear in LRA Section 4.7.5 or in the flaw evaluation that the cycles used in the flaw evaluation for weld RB-119-11 bounds the accumulated transient cycles at the end of 60 years. LRA Section 4.7.5 states that "[t]he service life for Weld RB-119-11 is based on operating for 40 years from the date the flaw was identified, i.e., until 2046, during which the flaw would experience 500 startup-shutdown cycles. Thus, the evaluation encompassed a 60-year plant life and the analysis will be valid beyond the 2045 end date of the period of extended operation for Unit 2." The above statements do not provide a clear reasoning as to how the flaw evaluation for 40 years encompasses 60 years of plant life. Clarify how the flaw evaluation encompassed a 60-year plant life in terms of cycle counting (e.g., are the 500 startup and shutdown cycles bound the actual plant cycles at the end of 60 years?).

3. Discuss how the program will ensure that transient cycles used in the flaw evaluation for the Unit 2 residual heat removal (RHR) piping weld RB-119-11 do not exceed the actual operating cycles at the end of 60 years without the enhanced fatigue management program.

4. (a) Provide the material specification of weld RB-119-11 (e.g., E308L or Alloy 82/182). (b) Discuss whether the indication in weld RB-119-11 is surface-connected or embedded. (c) Discuss the degradation mechanism of the indication. (d) If the weld is fabricated with Alloy 82/182 metal or if the flaw is embedded in the pipe/weld wall thickness, discuss any mitigation measures applied to the flaw in Weld RB-119-11.

5. Discuss whether weld RB-119-11 will be examined in the future ASME 10-year inservice inspection (ISI) intervals. If not, provide justifications.

RAI 4.7.5-2

LRA Section 4.7.5 discusses the flaw evaluation of an indication detected in weld WIC-95 of the RHR injection line 985 to hot legs 1 and 2 as shown in Pacific Gas and Electric Company (PG&E) Letter DCL-97-086 dated May 7, 1997. LRA Section 4.7.5 states further that "[t]here have been no occurrences of a DE, DDE, or Hosgri seismic event at Diablo Canyon Nuclear Power Plant (DCPP) during the first 20 plus years of operation. Therefore, the seismic cycles in the Unit 1 RHR Weld WIC-95 fatigue crack growth evaluation for the 50-year design basis number of DE, DDE, and Hosgri events are sufficient to the end of the period of extended operation."

1. LRA Section 4.7.5 states that "[t]he number of seismic cycles used in the analysis [flaw evaluation] is consistent with the DCPP 50-year design basis described in FSAR Table 5.2-4..." Final Safety Analysis Report (FSAR) Table 5.2-4 specifies one cycle for the Hosgri earthquake, 20 cycles for the design earthquake (DE), and 1 cycle for the double design earthquake (DDE). In the flaw evaluation for weld WIC-95 in the applicant's letter dated May 7, 1997, none of these seismic cycles were discussed. The applicant's flaw evaluation discussed only "400 cycles of future loading for the governing pipe stress load case." Clarify whether the seismic cycles were included in the flaw evaluation of the indication at weld WIC-95.

2. FSAR Table 5.2-4 provides several transients that have more occurrences/cycles than 400 cycles used in the flaw evaluation for weld WIC-95. For example, Unit loading and unloading at 5% of full power has 18,300 occurrences (cycles), hot standby operation/feedwater cycling has 18,300 occurrences. (a) Identify the transients that are included in the 400 cycles. (b) Provide basis for those transients shown in Table 5.2-4 but were not included in the flaw evaluation for weld WIC-95.

3. FSAR Table 5.2-4 specifies 250 occurrences for reactor coolant system heatup and cooldown transients. The total cycles for heatup and shutdown transients would be 500. However, the flaw evaluation used only 400 cycles. The staff notes that 500 cycles were used in the flaw evaluation of the indication in weld RB-119-11. The cycles in FSAR Table 5.2-4 are for the design life of the plant which presumably is 50 years. It appears that the 400 cycles used in the flaw evaluation for weld WIC-95 are for 50 years, not 60 years, of plant operation. LRA Section 4.7.5 states that the seismic cycles in the weld WIC-95 fatigue crack growth evaluation for the 50-year design basis number of DE, DDE, and Hosgri events are sufficient to the end of the period of extended operation. Clarify whether (a) the seismic cycles in the flaw evaluation in the May 7, 1997, letter, are sufficient to cover the seismic cycles at the end of extended operation, (b) the 400 cycles cover all the transient cycles at the end of extended operation, and (c) why a total of 500 cycles for heatup and cooldown were not used.

4. (a) Provide the pipe diameter and wall thickness at weld WIC-95 of the Unit 1 RHR injection line 985 where an indication was detected in refueling outage 9. (b) In the flaw evaluation dated May 7, 1997, the applicant stated that it will re-examine the indication in weld WIC-95 in refueling outage 1R10. Discuss the inspection result of weld WIC-95 during refueling outage 1R10. Confirm that the indication was detected in 1997 and was re-examined in 1999. (c) Provide the material specification of weld WIC-95 (e.g., Alloy 82/182 weld or E308L). (d) Discuss whether the subject indication is surface-connected

or embedded. (e) Discuss the degradation mechanism of the indication. (f) Discuss the orientation of the indication (i.e., a circumferential or an axial indication). (g) Provide operating temperature and pressure of the subject pipe line at weld WIC-95.

5. Discuss whether weld WIC-95 will be examined in the future ASME 10-year ISI inspection intervals. If not, provide justifications.

6. It is not clear to the staff that the applicant has demonstrated that the cycles used in the flaw evaluation for weld WIC-95 bounds the cycles at the end of 60 years. Discuss how the program will ensure that transient cycles used in the flaw evaluation for the RHR piping weld WIC-95 do not exceed the actual operating cycles.

RAI 4.7.5-3

LRA Section 4.7.5 discussed the indication detected in Unit 2 Auxiliary feedwater piping line 567. The applicant submitted a flaw evaluation in PG&E letter DCL-99-136, dated October 22, 1999.

1. In the flaw evaluation for piping line 567, the applicant stated that it will re-examine the indication during the Unit 2 tenth refueling outage (2R10). Discuss the inspection results of the re-examination.

2. The applicant stated in the flaw evaluation that the indication is believed to be a fabrication defect (a lap in the pipe). Confirm that the indication is embedded in the pipe wall. As stated in the flaw evaluation, the flaw was characterized as 0.1 inches deep (approximately 46 percent through wall) and 12 feet in length. Describe in detail how the indication is modeled in the flaw growth calculation.

3. The flaw evaluation dated October 22, 1999, states that the 250 cycles of future seismic and thermal loading corresponding to the remaining plant life. In LRA Section 4.7.5, the applicant stated that the assumed transients are consistent with or bounded by the 50-year design basis described in FSAR Table 5.2-4. It is not clear to the staff that 250 cycles used in the flaw evaluation bound the cycles in Table 5.2-4 in FSAR. Identify the transients that are included in the 250 cycles. Discuss in detail how 250 cycles in the flaw evaluation bound the cycles in the licensing basis.

4. Discuss whether the indication in Unit 2 Auxiliary feedwater piping line 567 will be examined in the future ASME 10-year ISI intervals. If not, provide justification.

RAI B2.1.39-1

In LRA Section B2.1.39, the applicant states that the Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program will be implemented as part of the ASME Code, Section XI ISI program and will be completed within the 10-year inspection interval before the period of extended operation.

1. The NRC staff notes that ultrasonic testing (UT) has not yet been qualified to examine CASS material via the ASME Code, Section XI, Appendix VIII. Discuss how components fabricated with CASS material are inspected under the current licensing basis. Discuss whether the current inspection practices (methods, frequencies and acceptance criteria) will be applied in the future CASS aging management program (AMP).

2. In light of the limitation of UT of CASS material, discuss how volumetric examination of CASS components will be accomplished during the period of extended operation. Specifically, clarify whether the qualified UT will only be used in the CASS AMP, if a qualified UT method becomes available.

RAI B2.1.39-2

(1) Discuss whether DCPP Units 1 and 2 have implemented the risk-informed ISI program. (2) If yes, discuss how the CASS components will be inspected under the risk-informed ISI program considering the requirements of the CASS AMP (e.g., whether the CASS AMP will increase the inspection frequency of the CASS components in the risk-informed ISI program and whether thermal aging embrittlement will be a degradation mechanism considered in the risk-informed ISI program).

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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 August 26, 2010

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