Pacific Gas and Electric Company®

James R. Becker
Site Vice President

Diablo Canyon Power Plant
Mail Code 104/5/601
P. O. Box 56
Avila Beach, CA 93424

805.545.3462
Internal: 691.3462
Fax: 805.545.6445

August 12, 2010

PG&E Letter DCL-10-098

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20852

Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
Response to NRC Letter dated July 15, 2010, Request for Additional Information (Set 10) for the Diablo Canyon License Renewal Application

Dear Commissioners and Staff:

By letter dated November 23, 2009, Pacific Gas and Electric Company (PG&E) submitted an application to the U.S. Nuclear Regulatory Commission (NRC) for the renewal of Facility Operating Licenses DPR-80 and DPR-82, for Diablo Canyon Power Plant (DCPP) Units 1 and 2, respectively. The application included the license renewal application (LRA), and Applicant's Environmental Report – Operating License Renewal Stage.

By letter dated July 15, 2010, the NRC staff requested additional information needed to continue their review of the DCPP LRA.

PG&E's response to the request for additional information is included in Enclosure 1. LRA Amendment 9, resulting from the responses, is included in Enclosure 2 showing the changed pages with line-in/line-out annotations.

PG&E makes no regulatory commitments (as defined in NEI 99-04) in this letter.

If you have any questions regarding this response, please contact Mr. Terence L. Grebel, License Renewal Project Manager, at (805) 545-4160.

A139
NRR

I declare under penalty of perjury that the foregoing is true and correct.

Executed on August 12, 2010.

Sincerely,

James R. Becker
Site Vice President

pns/50327747
Enclosures
cc: Diablo Distribution
cc/enc: Elmo E. Collins, NRC Region IV Regional Administrator
Nathanial B. Ferrer, NRC Project Manager, License Renewal
Kimberly J. Green, NRC Project Manager, License Renewal
Michael S. Peck, NRC Senior Resident Inspector
Alan B. Wang, NRC Project Manager, Office of Nuclear Reactor Regulation

**PG&E Response to NRC Letter dated July 15, 2010
Request for Additional Information (Set 10) for the
Diablo Canyon License Renewal Application**

RAI 2.4.4-1

Based on the staff's review of Section 2.4.4, "Turbine Building," and Table 2.4-4 of the license renewal application (LRA), it is not clear if the roof and roofing membranes have been included within the scope of license renewal and subject to an aging management review (AMR). If the components are not included due to oversight, please discuss whether these components are within the scope of license renewal and subject to an AMR. If they are excluded from the scope of license renewal, please provide the basis for their exclusion.

PG&E Response to RAI 2.4.4-1
By letter dated June 18, 2010, PGE submitted license renewal application (LRA) Amendment 1, which revised the turbine building screening and aging management reports and updated LRA Section 2.4.4 and Table 2.4-4, "Turbine Building," to add additional details regarding the turbine building. The response explains that the turbine building roof structure and materials are within the scope of license renewal but the design does not have a membrane.

RAI 2.5-1

NEI 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule," Appendix B indicates that elements, resistance temperature detectors (RTDs), sensors, thermocouples and transducers should be included in the list of component or commodity groups subject to an AMR if a pressure boundary is applicable. LRA Section 2.1.3.1 states that instrument and control (I&C) components with mechanical functions such as flow elements, flow indicators, flow orifices, and sight gauges were evaluated in their respective mechanical systems. However, it is not clear if I&C components such as RTDs, thermocouples and transducers that may have a pressure boundary function or other "mechanical function" have been included within the scope of license renewal. Please explain how such components were evaluated. Provide examples or references to such components and indicate where in the LRA these components have been listed as component types subject to an AMR, if required in accordance with 10 CFR 54.4(a).

PG&E Response to 2.5-1

License renewal application Section 2.1.3.1, "Mechanical System Scoping Methodology," indicates that instrument and control components with mechanical functions such as flow elements, flow indicators, flow orifices, and sight gauges are evaluated in their respective mechanical systems. Resistance temperature detectors (RTDs), sensors, thermocouples, transducers, and various elements at Diablo Canyon Power Plant do not have a pressure boundary since they are not in-line components. Thermowells or mounting brackets that may provide a pressure boundary for the sensing devices are evaluated as part of the piping system. Therefore, an aging management program for RTDs, sensors, thermocouples, transducers, and various elements is not required in accordance with 10 CFR 54.4(a).

DCPP RAI 3.1.2.1-1

The Generic Aging Lessons Learned (GALL) Report item IV.B2-21 recommends the Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program to manage loss of fracture toughness due to thermal aging and neutron irradiation embrittlement for the CASS lower internal assembly (lower support casting and lower support plate columns) exposed to reactor coolant (>250°C) and neutron flux. In addition, GALL Report item IV.B2-37 recommends the Thermal Aging and Neutron Irradiation Embrittlement of CASS Program to manage loss of fracture toughness due to thermal aging and neutron irradiation embrittlement for the CASS upper internals assembly, upper support columns exposed to reactor coolant (>250°C) and neutron flux.

Diablo Canyon Power Plant (DCPP) LRA Table 3.1.2-1, citing GALL Report items IV.B2-21 and IV.B2-37, indicates that the loss of fracture toughness of CASS Reactor Vessel Internal (RVI) lower core support structure (core support casing (U1)) and RVI upper core support structure (upper support columns) exposed to reactor coolant is managed by the Water Chemistry Program and the Final Safety Analysis Report (FSAR) Supplement commitment to (1) participate in industry RVI aging programs, (2) implement applicable results, and (3) submit for NRC approval >24 months before the extended period an RVI inspection plan, based on industry recommendation. The LRA uses Note E, along with a plant-specific note that states "Consistent with the GALL Report for material, environment and aging effect, but a different aging management program is credited or the GALL Report identifies a plant specific aging management program."

Clarify why the Water Chemistry Program with the commitment, which is different from the recommendation of the GALL Report, is adequate to manage the aging effect of the CASS components.

PG&E Response to RAI 3.1.2.1-1

In license renewal application (LRA) Table A4-1, "License Renewal Commitments," PG&E commits to the following activities for managing the aging effects of reactor vessel internals (RVI) components: (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.

EPRI's Material Reliability Program (MRP) has been in the process of developing guidance on managing RVIs, including cast austenitic stainless steel (CASS) components susceptible to thermal aging and neutron irradiation embrittlement. PG&E Pressurized Water Reactor (PWR) Vessel Internals Program will rely on the results of the ongoing MRP initiative to develop a comprehensive aging management program (AMP) for PWR reactor internals. As indicated in the EPRI report, MRP-227, the screening process includes the evaluations of aging effect of loss of fracture toughness in the CASS RVIs components as a result of thermal aging embrittlement and neutron irradiation embrittlement.

Implementation of the PWR Vessel Internals Program will provide reasonable assurance that aging effects will be managed such that components within the scope of this program will continue to perform their intended function(s) during the period of extended operation and is therefore an acceptable substitution for NUREG-1 801, AMP XI.M13, "Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)."

LRA Table 3.1.2-1, "Reactor Vessel, Internals, and Reactor Coolant System – Summary of Aging Management Evaluation – Reactor Vessel and Internals," has been revised to remove water chemistry as part of the AMP and to indicate that the loss of fracture toughness of CASS reactor vessel internal (RVI) lower core support structure (core support casting (U1)) and RVI upper core support structure (upper support columns) exposed to reactor coolant is managed by the Final Safety Analysis Report supplement commitment to: (1) participate in industry RVI aging programs; (2) implement applicable results; and (3) submit for NRC approval greater than 24 months before the extended period an RVI inspection plan, based on industry recommendation.

See revised LRA Tables 3.1.1 and 3.1.2-1 in Enclosure 2.

RAI 3.5.2.2.1.7-1

DCPP LRA Section 3.5.2.2.1.7 addresses the further evaluation of cracking due to stress corrosion cracking (SSC) of stainless steel penetration sleeves, penetration bellows, and dissimilar metal welds, as not applicable to DCPP.

LRA Section 3.5.2.2.1.7 indicates that this further evaluation is not applicable because DCPP has no in-scope stainless steel penetration sleeves, penetration bellows, or dissimilar metal welds subject to stress corrosion cracking. The FSAR, on Page 3.8-33, indicates that penetration sleeves for containment structures are made of carbon steel and that the flued heads, which appear to be welded to the penetrations sleeves, are made of stainless steel as shown in FSAR Figure 3.8-6. The staff noted that this information of the FSAR potentially contradicts the basis of the applicant's claim that the further evaluation described in LRA Section 3.5.2.2.1.7 is not applicable to DCPP.

Clarify why there are no in-scope stainless steel penetration sleeves, penetration bellows, or dissimilar metal welds subject to SCC, taking into consideration the stainless steel flued heads which are apparently welded to the penetration sleeves.

PG&E Response to RAI 3.5.2.2.1.7-1

Diablo Canyon Power Plant (DCPP) aging evaluation of the flued head dissimilar welds on containment penetrations concludes that the design and plant configuration preclude a corrosive environment from affecting the dissimilar welds. Therefore the welds should not be subject to stress corrosion cracking and do not require an aging management program. Final Safety Analysis Report, Figure 3.8-6 shows the configuration of the penetration sleeves. Inside containment the environment is plant indoor air, which is non-corrosive. Also the welds are protected from spray and drip by a leak chase, which was installed for testing purposes during plant construction. On the auxiliary building side, the flued head welds are isolated by insulation and a fire protection metal cover. Although the configuration on the auxiliary building side is not hermetically sealed, it does limit the free transfer of air from the auxiliary building environment to the dissimilar metal welds. The dissimilar metal welds are tested during the integrated leak rate test, which was recently conducted in 2008 for Unit 2 and 2009 for Unit 1. DCPP operating experience has not shown any failures of dissimilar metal welds for this unique plant configuration.

LRA Amendment 9

LRA Section	RAI
Table 3.1.1	RAI 3.1.2.1-1
Table 3.1.2-1	RAI 3.1.2.1-1

Table 3.1.1 Summary of Aging Management Evaluations in Chapter IV of NUREG-1801 for Reactor Vessel, Internals, and Reactor Coolant System (continued)

Item Number	Component Type	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.1.1.80	Cast austenitic stainless steel reactor vessel internals (e.g., upper internals assembly, lower internal assembly, CEA shroud assemblies, control rod guide tube assembly, core support shield assembly, lower grid assembly)	Loss of fracture toughness due to thermal aging and neutron irradiation embrittlement	Thermal Aging and Neutron Irradiation Embrittlement of CASS (B2.1.39)	No	Consistent with NUREG-1801 for material, environment, and aging effect, but a different aging management program ~~Water Chemistry (B2.1.2) and~~ FSAR supplement commitment to (1) participate in industry RVI aging programs (2) implement applicable results (3) submit for NRC approval > 24 months before the extended period an RVI inspection plan based on industry recommendations is credited.

Table 3.1.2-1 Reactor Vessel, Internals, and Reactor Coolant System – Summary of Aging Management Evaluation – Reactor Vessel and Internals

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
RVI Lower Core Support Structure (Core Support Casting (U1))	SS	Stainless Steel Cast Austenitic	Reactor Coolant (Ext)	Loss of fracture toughness	~~Water Chemistry (B2.1.2) and~~ FSAR supplement commitment to (1) participate in industry RVI aging programs (2) implement applicable results (3) submit for NRC approval > 24 months before the extended period an RVI inspection plan based on industry recommendation	IV.B2-21	3.1.1.80	E, 3
RVI Upper Core Support Structure (Upper Support Columns)	SS	Stainless Steel Cast Austenitic	Reactor Coolant (Ext)	Loss of fracture toughness	~~Water Chemistry (B2.1.2) and~~ FSAR supplement commitment to (1) participate in industry RVI aging programs (2) implement applicable results (3) submit for NRC approval > 24 months before the extended period an RVI inspection plan based on industry recommendation	IV.B2-37	3.1.1.80	E, 3