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PG&E Letter DCL-10-123

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 August 30, 2010, Request for Additional Information (Set 21) 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 August 30, 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 15 resulting from the responses is included in Enclosure 2 showing the changed pages with line-in/line-out annotations.

PG&E makes a commitment in revised LRA Table A4-1, License Renewal Commitments, shown in Enclosure 2.

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

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

Executed on September 29, 2010.

Sincerely,

James R. Becker

pns/50340646
Enclosure
cc: Diablo Distribution
cc/enc: Elmo E. Collins, NRC Region IV Regional Administrator
Nathanial Ferrer, NRC Project Manager, License Renewal
Kimberly J. Green, NRC Project Manager, License Renewal
Michael S. Peck, NRC Senior Resident Inspector
Fred Lyon, NRC Project Manager, Office of Nuclear Reactor Regulation

PG&E Response to NRC Letter dated August 30, 2010
Request for Additional Information (Set 21) for the
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RAI 3.1.2.3.2-2

In LRA Table 3.1.2-2, the applicant stated that for calcium silicate insulation exposed to borated water leakage (external) there is no aging effect and no AMP is proposed. The AMR line item cites generic note F. The 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.*

PG&E Response to RAI 3.1.2.3.2-2

1. The only insulation in scope for the reactor coolant system (RCS) is on the pressurizer loop seals. Units 1 and 2 do not have calcium silicate insulation on the pressurizer loop seals. Units 1 and 2 have reflective mirror insulation (RMI). The RMI has reflective steel foils encapsulated within a stainless steel enclosure. In addition to the RMI, fiberglass insulation has been installed at various locations. Aluminum tape is currently installed on the seams of the Unit 1 RMI insulation panels of the pressurizer loop seals to minimize the heat flow out of the pressurizer loop seals. This aluminum tape is currently scheduled to be removed during the Unit 1 sixteenth refueling outage (1R16) outage, October 2010. Aluminum is not being added to LRA Section 3.1.2.1.2 and Table 3.1.2-2 since it is currently scheduled to be removed during the 1R16 outage. The aluminum tape on the Unit 2 pressurizer loop seal insulation panels have been removed during the Unit 2 fifteenth refueling outage (2R15) outage, October 2009. See revised LRA Section 3.1.2.1.2 and Tables 3.1.2-2 and A4-1 in Enclosure 2.

2. For the stainless steel RMI in the environment of borated water leakage, there are no aging effects that require aging management. This is consistent with generic aging lessons learned (GALL) line item V.F-13. Fiberglass insulation exposed to borated water leakage also has no aging effects that require aging management. The fiberglass insulation is a composite material comprised of glass fibers and a polyester or epoxy resin. Fiberglass composites have excellent moisture resistance and chemical resistance to many corrosive materials, including acids. LRA Table 3.1.2-2 is revised to include the aging management review lines for these insulation materials. See revised LRA Section 3.1.2.1.2 and Table 3.1.2-2 in Enclosure 2.

<u>*RAI 3.3.2.3.5-1*</u>

<u>*Background:*</u>
In LRA Tables 3.3.2-5 and 3.4.2.1, the applicant stated 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 AMR line items cite generic note G and a plant specific note, which states "The 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." The AMR line item cites generic note G. 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 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 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 sodium hydroxide in reactor water because it is a contaminant that can cause stress corrosion cracking, but 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.

<u>*Issue:*</u>
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 sodium hydroxide. 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 sodium hydroxide; 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 sodium hydroxide.

Request:

1. *Describe the parameters (e.g., temperature, concentration) being monitored and acceptance criteria for the sodium hydroxide solution being monitored by the Water Chemistry Program.*
2. *Describe whether carbon steel components exposed to sodium hydroxide 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 sodium hydroxide 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.*

PG&E Response to RAI 3.3.2.3.5-1

The source of the sodium hydroxide, the caustic storage tank, is no longer in service. The caustic storage tank is shown on boundary drawing LR-DCPP-16-106716-20. Lines connecting the tank to other components are shown on boundary drawings LR-DCPP-16-106716-14, LR-DCPP-04-106704-16 and LR-DCPP-04-107704-16. The associated systems, structures, and components (SSCs) on these boundary drawings were inservice at one time but are no longer in service. The connections are flanged and the temperature of these components is below 118°F.

SSCs that used to contain sodium hydroxide solution when the caustic storage tank was in use are in scope of license renewal because they are shown on the boundary drawings as in-plant equipment and have not been formally abandoned-in-place. For scoping purposes these components were assumed to contain fluid, and some abandoned-in-place lines were also assumed to contain fluid. However these SSCs are no longer in service and the temperature of these components is below 118°F.

Since the conditions necessary for stress corrosion cracking (SCC) in carbon steel exposed to sodium hydroxide do not exist at Diablo Canyon Power Plant, SCC is not an applicable aging effect for the carbon steel components, and the aging effects that are applicable to these components are managed by the Water Chemistry Program.

<u>RAI 3.3.2.3.11-1</u>

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 by the External Surfaces Monitoring Program.

LRA Section B2.1.20, 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.

<u>PG&E Response to RAI 3.3.2.3.11-1</u>

The Diablo Canyon license renewal application only identifies the aging effects that require aging management, and therefore there is no reference to the applicable aging mechanisms referenced in NUREG-1801 Section IX.F.

As stated in License Renewal Application (LRA) Tables 3.3.2-11 and LRA Section B2.1.20, the aging effect of hardening and loss of strength will be managed by the External Surfaces Monitoring Program. The External Surfaces Monitoring Program will use physical manipulation of elastomers to augment visual inspections to confirm the absence of hardening or loss of strength. The physical manipulation in conjunction with the visual inspection would allow for the visual indications of cracking and changes in surface conditions to be identified that could be associated with the aging effect of hardening and loss of strength. Therefore the External Surfaces Monitoring Program is adequate to manage the aging effects of hardening and loss of strength.

<u>RAI 4.1-1</u>

<u>Background</u>: 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 TLAA in 10 CFR 54.3.

<u>Issue 1</u>: In LRA Section 4.7.3, the applicant states that the analysis in WCAP-15338-A qualifies reactor pressure vessels for the 60-year operating period rather than the current licensed operating period (40 years), and based on this, the flaw growth analysis for underclad cracks in low-alloy steel RV forgings is not a TLAA under 10 CFR 54.3(a)(3) criterion. The staff noted that the 60-year basis for concluding that the generic analysis in WCAP-15338-A does not need to be identified as a TLAA appears contrary to the basis in the SOC on 10 CFR Part 54 that identifies that 60-year analyses should be identified as TLAAs.

<u>Request 1</u>: Clarify whether WCAP-15338-A is being credited in the 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.

<u>Issue 2</u>: Non-proprietary Westinghouse 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 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 RVP 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 underclad cracking, provide responses to the LRAAIs.

PG&E Response to RAI 4.1-1

1. To date, DCPP has found no indications in the reactor vessel flange welds or nozzles requiring engineering analysis or monitoring. In the absence of underclad cracks, WCAP-15338-A is not being credited in the DCPP current licensing basis (CLB) for analysis of underclad flaws in the SA-508 forging materials used to fabricate the reactor vessels, nor is the report being credited for aging management during the period of extended operation. Since the WCAP is not part of the DCPP CLB, it is not a TLAA under 10 CFR 54.3(a)(3) criterion 6.

2. Should DCPP find indications of reactor vessel underclad flaws in the future and implement WCAP-15338-A, the License renewal applicant action items (LRAAIs) would be addressed as follows:

   (1) If WCAP-15338-A is relied upon for future evaluations, then as required by the NRC's safety evaluation for WCAP-15338-A, the number of design cycles and transients used in WCAP-15338-A would be evaluated to ensure that these are bounded by the Metal Fatigue of Reactor Coolant Pressure Boundary Components Aging Management Program (AMP) that is being relied upon for the period of extended operation.

   (2) If WCAP-15338-A is relied upon for future reactor vessel underclad flaw evaluations, as required by 10 CFR 54.37(b), DCPP would submit an Final Safety Analysis Report supplement update containing a summary description of how the WCAP-15338-A TLAA would be managed for the period of extended operation.

<u>*RAI 4.1-2*</u>

<u>*Background*</u>
In LRA Table 4.1-1, TLAA Category 6, "Plant-specific Time-limited Aging Analyses," the applicant stated that reactor coolant pump (RCP) flywheel fatigue crack growth analysis is not a TLAA. In LRA Section 4.7.4 the applicant further stated 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).

<u>*Issue*</u>
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.

<u>*Request*</u>
Clarify whether 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.

<u>PG&E Response to RAI 4.1-2</u>

The 60-year flaw growth analysis in WCAP-14535-A for the reactor coolant pump (RCP) flywheel is being relied upon in the DCPP current licensing basis to support the relaxation of the inservice inspection interval for the RCP flywheels. The LRA has been revised in Enclosure 2 to identify the analysis for the RCP flywheel as a TLAA. The analysis is for a period greater than 60 years and is dispositioned under 10 CFR 54.21(c)(1)(i). See revised LRA Sections 4.7.4, A3.5.4 and Table 4.1-1.

LRA Amendment 15

LRA Section	RAI
Section 3.1.2.1.2	3.1.2.3.2-2
Section 4.7.4	4.1-2
Section A3.5.4	4.1-2
Table 3.1.2-2	3.1.2.3.2-2
Table 4.1-1	4.1-2
Table 4.1-2	4.1-2
Table A4.1	3.1.2.3.2-2

3.1.2.1.2 Reactor Coolant System

Materials

The materials of construction for the reactor coolant system component types are:

- Carbon Steel
- ~~Insulation Calcium Silicate~~
- *Insulation Fiberglass*
- Stainless Steel
- Stainless Steel Cast Austenitic

4.7.4 ~~Absence of a TLAA for a~~ Reactor Coolant Pump Flywheel Fatigue Crack Growth Analysis

Summary Description

NUREG-1800 identifies “Fatigue analysis of the reactor coolant pump flywheel” as a potential TLAA.

During normal operation, the reactor coolant pump flywheel possesses sufficient kinetic energy to potentially produce high-energy missiles inside containment and could also damage pump seals or other pressure boundary components in the unlikely event of failure. Conditions that may result in overspeed of the reactor coolant pump increase both the potential for failure and the kinetic energy. The aging effect of concern is fatigue crack initiation in the flywheel bore keyway. This concern is the subject of Regulatory Guide 1.14, *Reactor Coolant Pump Flywheel Integrity*. At DCPP, flywheel fatigue is a recognized aging effect, but the aging effect is not the subject of a TLAA.

The original DCPP SER, NUREG-0675, states that the RCP motor flywheel is designed to meet the guidelines of Regulatory Guide 1.14. The DCPP flywheel design and its compliance with Regulatory Guide 1.14 is described in the FSAR Section 5.2.6. The inspection recommendations are incorporated in the DCPP ISI Program and are required by the TS.

To reduce the inspection frequency and scope, DCPP amended its initial compliance with Regulatory Guide 1.14 by implementing WCAP-14535-A [Reference 9], which supports relaxation of the inspection required by Regulatory Guide 1.14 Position C.4.b(1) and (2). The NRC has reviewed and accepted this topical report for use in license renewal applications. This relaxation was approved for DCPP with the Improved TS conversion [Reference 12] and was incorporated into the DCPP ISI Program and the TS.

Analysis

WCAP-14535-A [Reference 9] performed an evaluation of the probability of failure over the period of extended operation for all operating Westinghouse plants. It 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 6,000 pump starts~~)~~ *over a 60 year life*. Since the evaluation is based on the 60-year operating period*, the TLAA covers the period of extended operation and is dispostioned under 10 CFR 54.21(c)(1)(i).* ~~rather than the current licensed operating period (40 years), it is not a TLAA under 10 CFR 54.3 Criterion 3.~~

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

Using a conservative projection of 1,000 cycles for a 60 year plant life, the 6,000 events assumed in the fatigue crack growth analysis for the reactor coolant pump flywheels during 60 years of operation is conservative. The analysis is valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A3.5.4 Reactor Coolant Pump Flywheel Fatigue Crack Growth Analysis

The reactor coolant pump flywheels are supported by a fatigue crack growth analysis. Using a conservative projection of 1,000 cycles for a 60 year plant life, the 6,000 events assumed in the fatigue crack growth analysis for the reactor coolant pump flywheels during 60 years of operation is significantly conservative. The analysis is therefore valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

Table 3.1.2-2 Reactor Vessel, Internals, and Reactor Coolant System – Summary of Aging Management Evaluation – Reactor Coolant System

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
Heat Exchanger (RPV Support Cooler Plate)	LBS, SS	Carbon Steel	Plant Indoor Air (Ext)	Loss of material	External Surfaces Monitoring Program (B2.1.20)	V.C-1	3.2.1.31	D
Insulation	*INS*	*Insulation Fiberglass*	*Borated Water Leakage (Ext)*	*None*	*None*	*None*	*None*	*F*
Insulation	INS	~~Insulation Calcium Silicate~~*Stainless Steel*	Borated Water Leakage (Ext)	None	None	~~None~~ *V.F-13*	~~None~~*3.2.1.57*	~~F~~*C*
Orifice	PB, TH	Stainless Steel	Borated Water Leakage (Ext)	None	None	IV.E-3	3.1.1.86	A
Orifice	PB, TH	Stainless Steel	Reactor Coolant (Int)	Cracking	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD for Class 1 components (B2.1.1) and Water Chemistry (B2.1.2)	IV.C2-2	3.1.1.68	A
Orifice	PB, TH	Stainless Steel	Reactor Coolant (Int)	Loss of material	Water Chemistry (B2.1.2) and One-Time Inspection (B2.1.16)	IV.C2-15	3.1.1.83	E, 3
Piping	PB, SIA	Carbon Steel	Dry Gas (Int)	None	None	V.F-18	3.2.1.56	A
Piping	PB	Carbon Steel	Plant Indoor Air (Ext)	Loss of material	External Surfaces Monitoring Program (B2.1.20)	V.C-1	3.2.1.31	B

Table 4.1-1 List of TLAAs

TLAA Category	Description	Disposition Category[(a)]	Section
6.	**Plant-Specific Time-Limited Aging Analyses**	NA	**4.7**
	~~Absence of a TLAA for a~~ Reactor Coolant Pump Flywheel Fatigue Crack Growth Analysis	~~NA~~*i*	4.7.4

Table 4.1-2 Review of Analyses Listed in NUREG-1800 Tables 4.1-2 and 4.1-3

NUREG-1800 Examples	Disposition Category[a]	Section
Fatigue analysis for the reactor coolant pump flywheel	~~No – No explicit analysis based on plant life applies.~~ *Yes*	4.7.4

Table A4-1 License Renewal Commitments

Item #	**Commitment**	**LRA Section**	**Implementation Schedule**
47	*Aluminum tape currently installed on the seems of the Unit 1 RMI insulation panels of the pressurizer loop seals is currently scheduled to be removed during the Unit 1 sixteenth refueling outage (1R16) outage, October 2010.*	3.1.2.1.2	*Prior to the period of extended operation*