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PG&E Letter DCL-11-020

U.S. Nuclear Regulatory Commission
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Washington, DC 20555-0001

Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
10 CFR 54.21(b) Update to the DCPP License Renewal Application

Dear Commissioners and Staff:

By Pacific Gas and Electric Company (PG&E) Letter DCL-09-079, "License Renewal Application," dated November 23, 2009, 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 Applicant's Environmental Report – Operating License Renewal Stage.

As required by 10 CFR 54.21(b), at least 3 months before the scheduled completion of the NRC review of the license renewal application (LRA), an amendment to the renewal application must be submitted that identifies any change to the current licensing basis (CLB) of the facility that materially affects the contents of the license LRA, including the Final Safety Analysis Report supplement.

By PG&E Letter DCL-10-158, "10 CFR 54.21(b) Annual Update to the DCPP License Renewal Application and License Renewal Application Amendment No. 34," dated December 29, 2010, PG&E submitted its first 10 CFR 54.21(b) update to the LRA, which covered the period from July 1, 2009, through September 30, 2010.

Pursuant to the requirements of 10 CFR 54.37(b), PG&E has completed a review of regulatory correspondence and plant modifications covering the period from October 1, 2010, through November 30, 2010. Enclosure 1 identifies LRA changes that are being made to reflect: (1) CLB that materially affects the LRA; and (2) completed enhancements and commitments. Enclosure 2 contains the affected LRA pages with changes shown as electronic mark-ups (deletions crossed out and insertions underlined). As a reviewer aid, all pages of the Appendix B aging management program section are provided, including unchanged pages, when there is a change on any of the pages in that section.

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Changes to existing commitments are contained in the changes to LRA Table A4-1 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 March 25, 2011.

Sincerely,

James R. Becker
Site Vice President

gwh/50375929
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 Licensing Project Manager

DCPP License Renewal Application (LRA) Changes Reflected in the LRA Update Amendment 41

Affected LRA Section	Reason for Change
Section 2.3.3.2	Reflect removal of the Unit 2 temporary cask pit rack.
Table 2.3.3-5 Table 3.3.2-5	PG&E Letter DCL-11-002, dated January 21, 2011, stated that in-scope piping ends at valve MU-0-881. Components downstream of valve MU-0-881 were removed from license renewal scope to reduce the number of buried in-scope components.
Section 3.3.2.2.14 Table A4-1 #23	PG&E replaced the carbon steel with stainless steel clad CCP 1-1 pump casing in the CVCS with a completely stainless steel pump casing.
Section 4.2.1 Section 4.9 Appendix A3.1.1 Appendix B2.1.15	Reflect a change to the withdrawal date of Unit 1 Capsule B from 21 EFPY to 23.2 EFPY as approved by an NRC Safety Evaluation dated October 29, 2010. (ML103010159)
Appendix B2.1.15	Changed the number of standby capsules that will remain inside the Unit 1 reactor vessel throughout the vessel lifetime from "five" to "four" per PG&E Letter DCL-10-131, dated October 21, 2010.
Table A4-1 #38 and 50	Commitments have been completed.
Table A4-1 #49	Aligned the commitment statement with the implementation schedule.

LRA Amendment 41 Affected LRA Sections and Tables
Section 2.3.3.2
Table 2.3.3-5
Section 3.3.2.2.14
Table 3.3.2-5
Section 4.2.1
Section 4.9
Appendix A3.1.1
Table A4-1, #23, 38, 49, and 50
Appendix B2.1.15

2.3.3.2 Spent Fuel Pool Cooling System

System Description

The spent fuel pool (SFP) cooling and cleanup system removes decay heat from fuel stored in the SFP. Heat is transferred through the SFP heat exchanger to the component cooling water system. The refueling water purification (RWP) subsystem is included as part of this system to maintain water clarity and purity. The system also includes the new fuel racks, the spent fuel racks and cask pit storage cask restraint fixtures. The permanent spent fuel racks do not credit boron-absorbing panels but instead credit soluble boron in the SFP. ~~A temporary cask pit spent fuel rack is installed in Unit 2 and is authorized for use until the end of cycle 16.~~ The new fuel racks are not located in the SFP but are evaluated as part of this system for license renewal.

When the SFP cooling and cleanup system is in operation, water flows from the SFP to the SFP pump suction, is pumped through the tube side of the heat exchanger, and is returned to the pool. The suction line is located below the normal SFP water level, while the return line contains an anti-siphon hole near the surface of the water to prevent gravity drainage of the pool. While the heat removal operation is in process, a portion of the SFP water may be diverted away from the heat exchanger through the RWP subsystem to maintain water clarity and purity.

During refueling outages, connections are provided such that the refueling water may be pumped from either the RWST or the refueling cavity, through the RWP subsystem and discharged to either the refueling cavity or the RWST. In addition to this flowpath, it is possible to manually align the SFP cleanup system with the RWP system to clean the refueling canal water during fuel movement. The RWP pump may also be utilized to pump down the refueling canal by pumping water to the liquid hold-up tanks, located in the chemical and volume control system, through the RWP filter. To further assist in maintaining SFP water clarity, the water surface is cleaned by a skimmer loop.

Demineralized makeup water can be added directly to the SFP. Water from the condensate storage tank is pumped to the SFP using the makeup water transfer pumps and appropriate interconnecting piping and valves.

Table 2.3.3-5 Makeup Water System

Component Type	**Intended Function**
Trap	Leakage Boundary (spatial) ~~Pressure Boundary~~

3.3.2.2.14 Loss of Material due to Cladding Breach

The Water Chemistry program (B2.1.2) and the One-Time Inspection program (B2.1.16) manage loss of material due to cladding breach for steel clad with stainless steel pump casings exposed to treated borated water. The one-time inspection includes each of the affected components and will address the full internal cladding surface exposed to treated borated water.

NRC Information Notice 80-38 and Information Notice 94-63 address loss of material due to cladding breach for CVCS pumps fabricated of steel with stainless steel cladding. DCPP identifies pumps CCP 1-1 and CCP 2-2 as fabricated of steel with stainless steel cladding. NRC Information Notice 80-38 advises that the condition presents a "potential source of degradation over long term operations" and recommends a "non-destructive examination of this pump type." Information Notice 94-63 provides additional information about the condition described based on the analysis of industry operating experience and concludes that "corrosion of the base metal due to cladding cracks is usually relatively easy to identify through visual inspection". The One-Time Inspection program (B2.1.16) provides a non-destructive visual examination consistent with the guidance of Information Notice 80-38 and 94-63.

Prior to the period of extended operation, DCPP will replace the current carbon steel with stainless steel clad ~~CCP 1-1 and~~ *CCP* 2-2 pump casings with *a* completely stainless steel pump casing~~s~~.

Table 3.3.2-5 Auxiliary Systems – Summary of Aging Management Evaluation – Makeup Water System

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
~~Piping~~	~~PB~~	~~Carbon Steel~~	~~Atmosphere/ Weather (Int)~~	~~Loss of material~~	~~Inspection of internal Surfaces in Miscellaneous Piping and Ducting Components (B2.1.22)~~	~~VIII.B1-6~~	~~3.4.1.30~~	~~B~~
~~Piping~~	~~PB~~	~~Carbon Steel~~	~~Buried (Ext)~~	~~Loss of material~~	~~Buried Piping and Tanks Inspection (B2.1.18)~~	~~VII.G-25~~	~~3.3.1.19~~	~~B~~
~~Trap~~	~~LBS, PB~~	~~Cast Iron (Gray Cast Iron)~~	~~Buried (Ext)~~	~~Loss of material~~	~~Selective Leaching of Materials (B2.1.17)~~	~~VII.G-15~~	~~3.3.1.85~~	~~A~~
~~Trap~~	~~LBS, PB~~	~~Cast Iron (Gray Cast Iron)~~	~~Buried (Ext)~~	~~Loss of material~~	~~Buried Piping and Tanks Inspection (B2.1.18)~~	~~VII.G-25~~	~~3.3.1.19~~	~~B~~
~~Trap~~	~~LBS, PB~~	~~Cast Iron (Gray Cast Iron)~~	~~Raw Water (Int)~~	~~Loss of material~~	~~Selective Leaching of Materials (B2.1.17)~~	~~VII.G-14~~	~~3.3.1.85~~	~~A~~
~~Trap~~	~~LBS, PB~~	~~Cast Iron (Gray Cast Iron)~~	~~Raw Water (Int)~~	~~Loss of material~~	~~Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B2.1.22)~~	~~VII.G-24~~	~~3.3.1.68~~	~~E, 5~~

4.2.1 Neutron Fluence Values

Summary Description

Loss of fracture toughness is an aging effect caused by the neutron embrittlement aging mechanism that results from prolonged exposure to neutron radiation. This process results in increased tensile strength and hardness of the material with reduced toughness. The rate of neutron exposure is defined as neutron flux, and the cumulative degree of exposure over time is defined as neutron fluence. As neutron embrittlement progresses, the toughness/temperature curve shifts down (lower fracture toughness as indicated by Charpy upper-shelf energy or C_V USE), and the curve shifts to the right (brittle/ductile transition temperature increases). Neutron fluence projections are made in order to estimate the effect on these reactor vessel material properties (Section 4.2.2 and Section 4.2.3), and to determine if additional reactor vessel materials will be exposed to fluence greater than 1×10^{17} n/cm^2 (E>1.0 MeV) as a result of license renewal (extended beltline).

Analysis

Unit 1

The last capsule withdrawn and tested from Unit 1 was Capsule V at the end-of-cycle (EOC) 11. At that point, Unit 1 had operated for 14.27 EFPY. This capsule had a lead factor of 2.26 resulting in an exposure equivalent to 32.25 EFPY of operation. The results were documented in WCAP-15958 [Reference 2].

This exposure is less than that expected at EOLE. In PG&E Letter DCL-08-021, PG&E requested a change to the withdrawal date of Unit 1 Capsule B from 20.7 EFPY to 21.9 EFPY in order to capture enough fluence data for EOLE. The change was approved by the NRC in a Safety Evaluation dated September 24, 2008, *Diablo Canyon Nuclear Power Plant, Unit No. 1 – Approval of Proposed Reactor Vessel Material Surveillance Capsule Withdrawal Schedule (TAC No. MD8371)* [Reference 13].

During the scheduled Unit 1 Sixteenth Refueling Outage (1R16), refueling personnel were not able to remove the Capsule B access plug on the reactor core barrel flange. In PG&E Letter DCL-10-141, dated October 25, 2010, PG&E requested a change to the withdrawal date of Unit 1 Capsule B from 21.9 EFPY to 23.2 EFPY. The change was approved by the NRC in a Safety Evaluation dated October 29, 2010, Diablo Canyon Nuclear Power Plant, Unit No. 1 – Approval of Proposed Reactor Vessel Material Surveillance Program Withdrawal Schedule (TAC No. ME4924) [Reference 38].

4.9 References

38. *US NRC Letter. Carl F. Lyon, Project Manager, Plant Licensing Branch IV, Division of Operating Reactor Licensing, Office of Nuclear Reactor Regulation; to Mr. John Conway, Senior Vice President – Energy Supply and Chief Nuclear Officer, DCPP. "Diablo Canyon Nuclear Power Plant, Unit No. 1 – Approval of Proposed Reactor Vessel Material Surveillance Program Withdrawal Schedule (TAC No. ME4924)." 29 October 2010.*

A3.1.1 Neutron Fluence Values

Loss of fracture toughness is an aging effect caused by the neutron embrittlement aging mechanism that results from prolonged exposure to neutron radiation. This process results in increased tensile strength and hardness of the material with reduced toughness. The rate of neutron exposure is defined as neutron flux, and the cumulative degree of exposure over time is defined as neutron fluence. As neutron embrittlement progresses, the toughness/temperature curve shifts down (lower fracture toughness as indicated by Charpy upper shelf energy or C_V USE), and the curve shifts to the right (brittle/ductile transition temperature increases). Neutron fluence projections are made in order to estimate the effect on these reactor vessel material properties at the end-of-license extended (EOLE). The basis for EOLE is assumed to be 54 effective full power years (EFPY) based on a lifetime capacity factor of 90 percent for 60 years.

The last capsule withdrawn and tested from Unit 1 was Capsule V at the end-of-cycle (EOC) 11, which yielded an exposure less than that expected at EOLE. Capsule B will be withdrawn at *23.2*~~21.9~~ EFPY in order to capture enough fluence data for EOLE. The last remaining capsule withdrawn and tested from Unit 2 was Capsule V at EOC 9, which yielded an exposure comparable to that expected at EOLE.

The fluence values for EOLE were projected using ENDF/B-VI cross sections, and they comply with Regulatory Guide 1.190, *Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence.*

The DCPP reactor vessel EOLE fluence projections account for use of lower-leakage cores, and the Unit 1 power uprate. The fluence projections were revised to quantify expected fluence at the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

Neutron fluence will also be monitored and its effects managed for the period of extended operation by the DCPP Reactor Vessel Surveillance program, described in Section A1.15. The validity of these parameters and the analyses that depend upon them will therefore be managed to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

Table A4-1 License Renewal Commitments

Item #	Commitment	LRA Section	Implementation Schedule
23	DCPP will replace the current carbon steel with stainless steel clad ~~CCP 1-1 and~~ *CCP* 2-2 pump casings in the CVCS with *a* completely stainless steel pump casing~~s~~.	3.3.2.2.14	Prior to the period of extended operation
38	~~The actual plant transient cycles related to the SWOL and Model 93A Reactor Coolant Pumps fatigue crack growth analyses will be included in the existing plant transient monitoring program by January 31, 2011 to ensure that the actual plant transients do not exceed the L fatigue analysis limits.~~*Completed. Reference PG&E Letter DCL-11-020.*	~~4.3~~	~~Prior to January 31, 2011~~ *Completed*
49	DCPP will update the PM basis documents for strainers and screens in the makeup water system that support long term cooling and firewater inventory to require that they are cleaned and inspected on a 24 month frequency *prior to* ~~during~~ the period of extended operation.	B2.1.13	Prior to the period of extended operation.
50	~~Procedures will be enhanced to provide specific valves that need to be repositioned to provide Class I makeup to the spent fuel pool including the correct position of any normally open code break valves. Reference PG&E Letter DCL-10-133.~~*Completed. Reference PG&E Letter DCL-11-020.*	~~2.3.3.5~~	~~03/01/2011~~ *Completed*

B2.1.15 Reactor Vessel Surveillance

Program Description

The Reactor Vessel Surveillance program manages loss of fracture toughness due to neutron embrittlement in reactor materials exposed to neutron fluence exceeding $1.0E^{17}$ n/cm^2 (E>1.0 MeV). The program is consistent with ASTM E 185-70 and ASTM E 185-73 for Units 1 and 2, respectively. Capsules are periodically removed during the course of plant operating life. Neutron embrittlement is evaluated through surveillance capsule testing and evaluation, ex-vessel neutron fluence calculations, and monitoring of reactor vessel neutron fluence. The testing program and reporting conform to requirements of 10 CFR 50 Appendix H, *Reactor Vessel Material Surveillance Program Requirements.* Data resulting from the program is used to:

- Determine pressure-temperature limits, minimum temperature requirements, and end-of-life Charpy upper-shelf energy (C_V USE) in accordance with the requirements of 10 CFR 50 Appendix G, *Fracture Toughness Requirements*; and,

- Determine end-of-life RT_{PTS} values in accordance with 10 CFR 50.61, *Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock.*

The Reactor Vessel Surveillance program provides guidance for removal and testing or storage of material specimen capsules. All capsules that have been withdrawn and tested were stored.

For 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 remaining ~~five~~ *four* standby capsules have low lead factors, will remain inside the vessel throughout the vessel lifetime, and will be available for future testing.

There are no capsules remaining in the Unit 2 vessel. All capsules were removed because high lead factors produced exposures comparable to the fluence expected at the end of the period of extended operation.

DCPP Units 1 and 2 currently use ex-vessel monitoring dosimetry, which consists of four gradient chains with activation foils outside the reactor vessel, which will be used to monitor the neutron fluence environment within the beltline region.

NUREG-1801 Consistency

The Reactor Vessel Surveillance program is an existing program that is consistent with NUREG-1801, Section XI.M31, Reactor Vessel Surveillance.

Exceptions to NUREG-1801

None

Enhancements

None

Operating Experience

Reactor Vessel Surveillance program experience at DCPP is evaluated and monitored to maintain an effective program. This is accomplished by promptly identifying and documenting (using the Corrective Action Program) any conditions or events that could compromise the program. In addition, industry operating experience provides input to ensure that the program is maintained. The DCPP operating experience findings for this program identified no unique plant specific operating experience; therefore DCPP operating experience is consistent with NUREG-1801.

The Reactor Vessel Surveillance program has provided materials data and dosimetry for the monitoring of irradiation embrittlement since plant startup. The use of this program has been reviewed and approved by the NRC during the period of current operation. Surveillance capsules have been withdrawn during the period of current operation, and the data from these surveillance capsules have been used to verify and predict the performance of DCPP reactor vessel beltline materials with respect to neutron embrittlement. Calculations have been performed as required to project the reference temperature for pressurized thermal shock (RT_{PTS}) and Charpy upper-shelf energy (C_V USE) values to the end-of-license-extended (EOLE). DCPP pressure-temperature limit curves are valid up to a stated vessel fluence limit, and must be revised prior to operating beyond that limit.

Neutron Fluence

The last capsule withdrawn and tested from Unit 1 was Capsule V at the end-of-cycle (EOC) 11, which yielded an exposure less than that expected at EOLE. Capsule B will be withdrawn at ~~21.9~~*23.2* EFPY in order to capture enough fluence data for EOLE. The last capsule withdrawn and tested from Unit 2 was Capsule V at EOC 9, which yielded an exposure comparable to that expected at EOLE. The EOLE fluence projections include the use of lower-leakage cores and the Unit 1 power uprate.

Pressurized Thermal Shock

The projected Unit 1 RT_{PTS} values did not meet the 10 CFR 50.61 screening criteria for beltline and extended beltline materials. The Unit 2 RT_{PTS} was projected to the end of the period of extended operation. The Unit 1 reactor vessel fluence will continue to be monitored as part of the Reactor Vessel Surveillance program.

Charpy Upper-Shelf Energy

The most recent coupon examination results for both units demonstrate that the DCPP reactor vessel material ages consistently with Regulatory Guide 1.99 predictions and provides a conservative means to satisfy the requirements of 10 CFR 50, Appendix G. The C_V USE values were revised with projections to the end of the period of extended operation.

The Reactor Vessel Surveillance program operating experience information provides objective evidence to support the conclusion that the effects of aging will be adequately managed so that the component intended functions will be maintained during the period of extended operation.

Conclusion

Continued implementation of the Reactor Vessel Surveillance program provides reasonable assurance that the aging effects will be managed so that the systems and components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.