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PG&E Letter DCL-15-150

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

10 CFR 54.21(b)

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) Annual Update to the Diablo Canyon Power Plant License Renewal Application (LRA), Amendment 51

References:

1. PG&E Letter DCL-09-079, "License Renewal Application," dated November 23, 2009

2. PG&E Letter DCL-14-103, "10 CFR 54.21(b) Annual Update to the Diablo Canyon Power Plant License Renewal Application (LRA), Amendment 48 and LRA Appendix E, Applicant's Environmental Report - Operating License Renewal Stage, Amendment 1," dated December 22, 2014

3. PG&E Letter DCL-15-121, "Response to NRC Letter dated September 24, 2015, Request for Additional Information for the Review of the Diablo Canyon Power Plant, Units 1 and 2, License Renewal Application - Set 38," dated October 21, 2015

Dear Commissioners and Staff:

By Reference 1, 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 LRA Appendix E, "Applicant's Environmental Report – Operating License Renewal Stage." As required by 10 CFR 54.21(b), each year following submittal of the LRA, an update to the LRA must be submitted that identifies any change to the current licensing basis (CLB) that materially affects the contents of the LRA, including the Final Safety Analysis Report Supplement. PG&E has not made a decision to move forward with the State licensing review process at this time. A schedule for potential coastal consistency review has not been established and will be provided if a decision is made to resume State licensing.

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Enclosure 1 identifies DCPP LRA changes that are being made to reflect CLB that materially affect the LRA. Enclosure 2 contains the affected LRA pages with changes shown as electronic markups (deletions crossed out and insertions italicized). The LRA update covers the period from October 1, 2014, through September 30, 2015.

By Reference 2, PG&E committed to provide the NRC with responses to the applicable aging management program plant-specific action items, conditions, and limitations identified in the NRC Safety Evaluation (SE), Revision 1, on MRP-227 by December 2015. Enclosures 3 and 4 submit WCAP-17462-NP, Revision 1, "Program Plan for Aging Management of Reactor Vessel Internals at Diablo Canyon Power Plant Unit 1,"and WCAP-17463-NP, Revision 1, "Program Plan for Aging Management of Reactor Vessel Internal at Diablo Canyon Power Plant Unit 2," respectively. WCAP-17462-NP, Revision 1, and WCAP-17463-NP, Revision 1, contain responses to the applicable plant-specific action items, conditions, and limitations identified in the NRC SE, Revision 1, on MRP-227. In addition, although not specifically required by the NRC SE, Revision 1, PG&E is aware that the NRC is requesting licensees to provide additional plant specific information to address NRC expectations and concerns regarding responses to plant-specific action items 1 and 2. Enclosure 5 provides additional plant-specific information that the NRC has been requesting from other licensees to address actions items 1 and 2 for DCPP Units 1 and 2. As noted in Enclosures 3 and 4, PG&E will be providing an evaluation of the DCPP Units 1 and 2 reactor internals components with regard to fuel designs and fuel management by March 31, 2016.

By Reference 3, PG&E committed to update the cathodic protection licensing basis by December 31, 2015. The design process for upgrading the cathodic protection system on buried, in-soil auxiliary saltwater system piping is ongoing. PG&E is revising this commitment to update the cathodic protection design and installation action plan and associated licensing basis by March 31, 2016.

New and revised regulatory commitments (as defined by NEI 99-04) are provided in Enclosure 6. Changes to existing LRA Table A4-1 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) 458-0534.

I have been delegated the authority of Edward D. Halpin, Senior Vice President – Power Generation and Chief Nuclear Officer, during his absence. I declare under penalty of perjury that the foregoing is true and correct.

Executed on December 21, 2015.

Sincerely,

L. Jearl Strickland, P.E.
Director, Technical Services

gwh/50668099
Enclosures

cc: Diablo Distribution

cc/enc: Marc L. Dapas, NRC Region IV Administrator
Siva P. Lingam, NRC Project Manager
Richard A. Plasse, NRC Project Manager, License Renewal
John P. Reynoso, Acting NRC Senior Resident Inspector
Michael J. Wentzel, NRC Project Manager, License Renewal (Environmental)

Diablo Canyon Power Plant License Renewal Application (LRA) Changes Reflected in the Annual LRA Update Amendment 51

Diablo Canyon Power Plant License Renewal Application (LRA) Changes Reflected in the Annual LRA Update Amendment 51

Affected LRA Section	Reason for Change
Section 2.4.11	Updated to add a newly installed steel plate security-related enclosure to the scope of License Renewal. This enclosure was analyzed to ensure that failure of the enclosure will not impact Design Class I structures, systems, and components.
Table 2.3.3-12 Table 3.3.2-12	LRA tables were updated to reflect the fire protection hose stations as-built configuration. In the Diablo Canyon Power Plant (DCPP) LRA, hose stations represent the hose reel and not the associated isolation valve or piping; thus, there is no pressure boundary function. Hose reels only provide a structural support function. While verifying the hose reel material, it was noted that the copper alloy orifices and piping leading from the isolation valve to the fire hose were not addressed in the LRA. These components were added.
Table 3.3.2-12	Errata. Updated to change aging effect of copper alloy (> 15% zinc) fire protection spray nozzles with an internal environment of plant indoor air from none to loss of material, consistent with NUREG-1801, table item VII.G-9, as revised in LR-ISG-2012-02.
Table 3.3.2-8	PG&E responded to license renewal request for additional information 2.3-2 related to whether the guard pipe enclosing the hydrogen piping was in the scope of license renewal in PG&E Letter DCL-10-067, "Response to License Renewal Application (LRA) Request for Additional Information and LRA Errata," dated June 18, 2010, and PG&E Letter DCL-10-128, "Response to NRC Letter dated September 13, 2010, Request for Additional Information (Set 23) for the Diablo Canyon License Renewal Application," dated October 12, 2010. Table 3.3.2-8 is updated to clarify that the guard pipe enclosing the hydrogen piping is not a fire barrier since it does not prevent the spread of fire. The most-appropriate and most-conservative license renewal intended function of fire barrier was chosen to support its inclusion as being in-scope because it is relied upon in the Fire Hazards Analysis.

Affected LRA Section	Reason for Change
Section 3.1.2.1.2 Section 3.3.2.1.3 Section 3.3.2.1.4 Section 3.3.2.1.5 Section 3.3.2.1.8 Section 3.3.2.1.12 Section 3.3.2.1.13 Section 3.3.2.1.17 Section 3.3.2.1.19 Section 3.4.2.1.1 Section 3.4.2.1.4	Errata. Updated to add the Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks Program to manage the aging effects for the system component types.
Section 3.2.2.1.1 Section 3.2.2.1.3 Section 3.3.2.1.3 Section 3.3.2.1.8 Section 3.4.2.1.3 Section 3.4.2.1.5	Errata. Updated to add wall thinning due to erosion as an aging effect requiring management, and where applicable, the Flow Accelerated Corrosion Program to manage the aging effects for the system component types.
Section 3.2.2.1.2 Section 3.3.2.1.14 Table 3.2.2-2 Table 3.3.2-3 Table 3.3.2-9 Table 3.3.2-14 Table 3.4.2-1	Updated to reflect plant modifications (review of equipment changes).
Section 4.7.5 Section 4.9 Section A3.5.3	Updated to reflect a new stress and fracture mechanics evaluation completed to support request for alternative REP-SI, Revision 2, "Proposed Alternative to Requirements for Repair/Replacement Activities for Certain Safety Injection Pump Welded Attachments." Updated to reflect new inservice flaw growth analyses that were performed to address weld flaw indications identified in the Unit 2 structural weld overlays for the pressurizer safety spray nozzle welds. Updated to reflect a new inservice flaw growth analysis that was performed to address weld flaw indications identified in the Unit 1 pressurizer spray line pipe weld.
Section A1.14	Updated to add fuel tank for portable diesel electric generators credited during fire protection events for supporting safe shutdown.

Affected LRA Section	Reason for Change
Section A1.13 Table A4-1, Item 3	In PG&E Letter DCL-14-103, "10 CFR 54.21(b) Annual Update to the Diablo Canyon Power Plant License Renewal Application (LRA), Amendment 48 and LRA Appendix E, Applicant's Environmental Report – Operating License Renewal Stage, Amendment 1," dated December 22, 2014, PG&E stated that following the enhancements listed in Enclosure 1, Attachment 7C, and with the exceptions listed, the DCPP Fire Water System Program would be consistent with LR-ISG-2012-02, Section C. PG&E has determined that an enhancement not mentioned in DCL-14-103 was necessary for the deluge testing to be consistent with LR-ISG-2012-02, Section C. PG&E is revising the licensing basis to enhance the testing of deluge system nozzles consistent with the 2011 edition of NFPA 25, Section 10.3.4.3.1, as recommended by LR-ISG-2012-02, Appendix D, Table 4a. Following this enhancement and those listed in DCL-14-103, Enclosure 1, Attachment 7C, and with the exceptions listed, PG&Es Fire Water System Program will be consistent with LR-ISG-2012-02, Section C.
Table A4-1, Items 34 and 60	Updated the status of these items to show them as completed.
Table A4-1, Item 73	Enclosure 3 of this letter provides DCPP Unit 1 and 2 responses to the applicable aging management program plant-specific action items, conditions, and limitations identified in the NRC SE, Revision 1, on MRP-227. Table A4-1, item 73 is revised to show this commitment is complete.

License Renewal Application (LRA) Amendment 51 Affected LRA Sections and Tables, and Figures

License Renewal Application (LRA) Amendment 51 Affected LRA Sections and Tables, and Figures
Table 2.3.3-12
Section 2.4.11
Section 3.1.2.1.2
Section 3.2.2.1.1
Section 3.2.2.1.2
Section 3.2.2.1.3
Table 3.2.2-2
Section 3.3.2.1.3
Section 3.3.2.1.4
Section 3.3.2.1.5
Section 3.3.2.1.8
Section 3.3.2.1.12
Section 3.3.2.1.13
Section 3.3.2.1.14
Section 3.3.2.1.17
Section 3.3.2.1.19
Table 3.3.2-3
Table 3.3.2-8
Table 3.3.2-9
Table 3.3.2-12
Table 3.3.2-14
Section 3.4.2.1.1
Section 3.4.2.1.3
Section 3.4.2.1.4
Section 3.4.2.1.5
Table 3.4.2-1
Section 4.7.5
Section 4.9
Section A1.13
Section A1.14
Section A3.5.3
Table A4-1, Items 3, 34, 60, and 73

Table 2.3.3-12 *Fire Protection System*

Component Type	Intended Function
Bellows	Pressure Boundary
Closure Bolting	Pressure Boundary
Flow Element	Pressure Boundary
Flow Indicator	Pressure Boundary
Hose Station	~~Pressure Boundary~~*Structural Support*
Hydrant	Pressure Boundary
Orifice	Pressure Boundary *Throttle*
Piping	Leakage Boundary (spatial) Pressure Boundary Structural Support
Pump	Pressure Boundary Structural Support
RCP Oil Collection Reservoir	Pressure Boundary
Solenoid Valve	Pressure Boundary
Spray Nozzle	Spray
Strainer	Pressure Boundary
Tank	Pressure Boundary Structural Support
Test Connection	Pressure Boundary
Trailer	Structural Support
Tubing	Leakage Boundary (spatial) Pressure Boundary
Valve	Leakage Boundary (spatial) Pressure Boundary
Vessel	Pressure Boundary

2.4.11 Earthwork and Yard Structures

Structure Description

The earthwork and yard structures include the circulating water conduits, auxiliary saltwater (ASW) vacuum breaker vaults, ASW thrust blocks and anchors, *a security-related enclosure,* raw water storage reservoirs 1A and 1B, east and west breakwaters, and the earth slopes east of the auxiliary building and over the ASW line east of the intake structure.

The seismically qualified portions of the circulating water conduits and ASW vacuum breaker vaults are reinforced concrete structures founded on compacted fill. The Design Class I ASW supply piping is supported by reinforced concrete thrust blocks, compacted backfill, and concrete anchors attached to the circulating water conduits.

The seismically qualified security-related enclosure is a steel structure and was analyzed to ensure that failure of the enclosure will not impact Design Class I SSCs. The security-related enclosure does not perform any (a)(1) intended functions and does not contain components required by the five License Renewal regulated events (a)(3).

The raw water reservoir, located east of the power block, has reinforced concrete-walls. The reservoir is primarily intended to serve as fresh water storage for fire protection and long term cooling.

The breakwater structures, which are constructed of precast reinforced concrete blocks and rip-rap, protect the intake structure from tsunami loads. The earth slopes east of auxiliary building and over the ASW line east of the intake structure were analyzed for design basis seismic loads to ensure that such loading will not produce any significant slope failure that can impact Design Class I SSCs. The ASW system buried piping and electrical conduits are protected from tsunami/storm conditions by wave protection measures, which include concrete covers, revetments, roadway slabs, and pavement. Gabion mattresses embedded within the slopes are covered with grass for additional erosion control.

For the purposes of license renewal and aging management, the breakwaters and earth slope protection structures are evaluated as barriers.

Structure Intended Functions

The earthwork and yard structures*, except the security-related enclosure,* provide structural support, shelter, and protection for components relied upon to provide the capability to shutdown the reactor and maintain it in a safe shutdown condition. The raw water reservoir provides fresh water storage for long term cooling.

The earthwork and yard structures*, except the security-related enclosure,* also provide structural support, shelter, and protection for nonsafety-related SSCs whose

failure could prevent performance of a safety-related function. Therefore, the structures are within the scope of license renewal based on the criterion of 10 CFR 54.4(a)(2).

The security-related enclosure provides structural support whose failure could prevent performance of a safety-related function. Therefore, the structure is within the scope of license renewal based on the criterion of 10 CFR 54.4(a)(2).

The earthwork and yard structures*, except the security-related enclosure,* provide structural support, shelter, and protection for components required to support fire protection and SBO requirements. Therefore, the earthwork and yard structures are within the scope of license renewal based on the criteria of 10 CFR 54.4(a)(3).

3.1.2.1.2 Reactor Coolant System

Aging Management Programs

The following aging management programs manage the aging effects for the reactor coolant system component types:

- *Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks (B2.1.42)*

3.2.2.1.1 Safety Injection System

Aging Effects Requiring Management

The following saltwater and chlorination system aging effects require management:

- *Wall thinning due to erosion*

Aging Management Programs

The following aging management programs manage the aging effects for the saltwater and chlorination system component types:

- *Flow-Accelerated Corrosion (B2.1.6)*

3.2.2.1.2 Containment Spray System

Materials

The materials of construction for the containment spray system component types are:

- ~~Copper Alloy~~

3.2.2.1.3 Residual Heat Removal System

Aging Effects Requiring Management

The following saltwater and chlorination system aging effects require management:

- *Wall thinning due to erosion*

Aging Management Programs

The following aging management programs manage the aging effects for the saltwater and chlorination system component types:

- *Flow-Accelerated Corrosion (B2.1.6)*

Table 3.2.2-2 Engineered Safety Features – Summary of Aging Management Evaluation – Containment Spray System

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
~~Valve~~	~~PB~~	~~Copper Alloy~~	~~Dry Gas (Int)~~	~~None~~	~~None~~	~~V.F-4~~	~~3.2.1.56~~	~~A~~
~~Valve~~	~~PB~~	~~Copper Alloy~~	~~Plant Indoor Air (Ext)~~	~~None~~	~~None~~	~~V.F-3~~	~~3.2.1.53~~	~~A~~

3.3.2.1.3 Saltwater and Chlorination System

Materials

The materials of construction for the saltwater and chlorination system component types are:

- *Ductile Iron*

Aging Effects Requiring Management

The following saltwater and chlorination system aging effects require management:

- *Wall thinning due to erosion*

Aging Management Programs

The following aging management programs manage the aging effects for the saltwater and chlorination system component types:

- *Flow-Accelerated Corrosion (B2.1.6)*
- *Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks (B2.1.42)*

3.3.2.1.4 Component Cooling Water System

Aging Management Programs

The following aging management programs manage the aging effects for the component cooling water system component types:

- *Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks (B2.1.42)*

3.3.2.1.5 Makeup Water System

Aging Management Programs

The following aging management programs manage the aging effects for the makeup water system component types:

- *Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks (B2.1.42)*

3.3.2.1.8 Chemical and Volume Control System

Aging Effects Requiring Management

The following saltwater and chlorination system aging effects require management:

- *Wall thinning due to erosion*

Aging Management Programs

The following aging management programs manage the aging effects for the chemical and volume control system component types:

- *Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks (B2.1.42)*

3.3.2.1.12 Fire Protection System

Aging Management Programs

The following aging management programs manage the aging effects for the fire protection system component types:

- *Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks (B2.1.42)*

3.3.2.1.13 Diesel Generator Fuel Oil System

Aging Management Programs

The following aging management programs manage the aging effects for the diesel generator fuel oil system component types:

- *Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks (B2.1.42)*

3.3.2.1.14 Diesel Generator System

Materials

The materials of construction for the diesel generator system component types are:

- *Copper Alloy (> 15 percent Zinc)*

3.3.2.1.17 Liquid Radwaste System

Aging Management Programs

The following aging management programs manage the aging effects for the liquid radwaste system component types:

- *Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks (B2.1.42)*

3.3.2.1.19 Oily Water and Turbine Sump System

Aging Management Programs

The following aging management programs manage the aging effects for the oily water and turbine sump system component types:

- *Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks (B2.1.42)*

Table 3.3.2-3 Auxiliary Systems – Summary of Aging Management Evaluation – Saltwater and Chlorination System

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
Separator	*LBS*	*Nickel Alloys*	*Plant Indoor Air (Ext)*	*None*	*None*	*VII.J-14*	*3.3.1.94*	*A*
Separator	*LBS*	*Nickel Alloys*	*Raw Water (Int)*	*Loss of material*	*Open-Cycle Cooling Water System (B2.1.9)*	*VII.C1-13*	*3.3.1.78*	*A*
Valve	*LBS*	*Ductile Iron*	*Plant Indoor Air (Ext)*	*Loss of material*	*External Surfaces Monitoring Program (B2.1.20)*	*VII.I-8*	*3.3.1.58*	*B*
Valve	*LBS*	*Ductile Iron*	*Raw Water (Int)*	*Loss of material*	*Open-Cycle Cooling Water System (B2.1.9)*	*VII.C1-19*	*3.3.1.76*	*A*
Valve	*LBS*	*Ductile Iron*	*Raw Water (Int)*	*Wall thinning due to erosion*	*Flow-Accelerated Corrosion (B2.1.6)*	*None*	*None*	*H, 5*

Table 3.3.2-8 Auxiliary Systems – Summary of Aging Management Evaluation – Chemical and Volume Control System

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
Piping	FB, LBS, PB, SIA	Carbon Steel	Plant Indoor Air (Ext)	Loss of material	External Surfaces Monitoring Program (B2.1.20)	VII.I-8	3.3.1.58	B, *12*
Piping	FB, LBS, SIA	Carbon Steel	Plant Indoor Air (Int)	Loss of material	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B2.1.22)	V.A-19	3.2.1.32	B, *12*

Notes for Table 3.3.2-8:

Plant Specific Notes:

12 *The guard pipe enclosing the hydrogen piping is credited in the Fire Hazards Analysis and is thus in the scope of License Renewal per 10 CFR 54.4(a)(3). The guard pipe is not a fire barrier since it does not prevent the spread of fire, as defined in Regulatory Guide 1.120. The most-appropriate and most-conservative license renewal intended function of "fire barrier" was chosen to support its inclusion as being in-scope because it is relied upon in the Fire Hazards Analysis.*

Table 3.3.2-9 Auxiliary Systems – Summary of Aging Management Evaluation – Miscellaneous HVAC Systems

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
Valve	*SIA,* SS	Copper Alloy	Plant Indoor Air (Ext)	None	None	VIII.I-2	3.4.1.41	C
Valve	*SIA,* SS	Copper Alloy	Ventilation Atmosphere (Int)	Loss of material	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B2.1.22)	VII.G-9	3.3.1.28	E

Table 3.3.2-12 Auxiliary Systems – Summary of Aging Management Evaluation – Fire Protection System

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
Hose Station	~~PB~~*SS*	Carbon Steel	Atmosphere/ Weather (Ext)	Loss of material	External Surfaces Monitoring Program (B2.1.20)	VII.I-9	3.3.1.58	B, *16*
Hose Station	~~PB~~*SS*	Carbon Steel	Plant Indoor Air (Ext)	Loss of material	External Surfaces Monitoring Program (B2.1.20)	VII.I-8	3.3.1.58	B, *16*
~~Hose Station~~	~~PB~~	~~Carbon Steel~~	~~Raw Water (Int)~~	~~Flow blockage~~	~~Fire Water System (B2.1.13)~~	~~None~~	~~None~~	~~H, 9~~
~~Hose Station~~	~~PB~~	~~Carbon Steel~~	~~Raw Water (Int)~~	~~Loss of material~~	~~Fire Water System (B2.1.13)~~	~~VII.G-24~~	~~3.3.1.68~~	~~B~~
~~Hose Station~~	~~PB~~	~~Carbon Steel~~	~~Raw Water (Int)~~	~~Loss of material; recurring internal corrosion~~	~~Fire Water System (B2.1.13)~~	~~None~~	~~None~~	~~H, 7~~
Orifice	*PB, TH*	*Copper Alloy (> 15% Zinc)*	*Atmosphere/ Weather (Ext)*	*Loss of material*	*Selective Leaching of Materials (B2.1.17)*	*None*	*None*	*G*
Orifice	*PB, TH*	*Copper Alloy (> 15% Zinc)*	*Plant Indoor Air (Ext)*	*None*	*None*	*VIII.I-2*	*3.4.1.41*	*A*
Orifice	*PB, TH*	*Copper Alloy (> 15% Zinc)*	*Plant Indoor Air (Int)*	*Loss of material*	*Fire Water System (B2.1.13)*	*VII.G-9*	*3.3.1.28*	*E, 8*
Piping	*PB*	*Copper Alloy (> 15% Zinc)*	*Atmosphere/ Weather (Ext)*	*Loss of material*	*Selective Leaching of Materials (B2.1.17)*	*None*	*None*	*G*
Piping	*PB*	*Copper Alloy (> 15% Zinc)*	*Plant Indoor Air (Ext)*	*None*	*None*	*VIII.I-2*	*3.4.1.41*	*A*
Piping	*PB*	*Copper Alloy (> 15% Zinc)*	*Plant Indoor Air (Int)*	*Loss of material*	*Fire Water System (B2.1.13)*	*VII.G-9*	*3.3.1.28*	*E, 8*
Spray Nozzle	SP	Copper Alloy (> 15% Zinc)	Plant Indoor Air (Int)	*Loss of material*~~None~~	*Fire Water System (B2.1.13)*~~None~~	*VII.G-9*~~None~~	*3.3.1.28*~~None~~	*E, 8*~~G~~

Notes for Table 3.3.2-12:

Plant Specific Notes:

8 The Fire Water System program (B2.1.13) is used to monitor copper alloy piping, piping components and piping elements exposed to condensation (internal) for loss of material in the fire protection system. Reference LR-ISG-2012-02, Appendix C, Line VII.G.A-143, ~~and~~ PG&E Letter DCL-14-103, Enclosure 1, Attachment 7C*, and DCL-15-150, Enclosure 2*.

16. This line item only represents the hose reel and not the associated isolation valve, piping, or fittings.

Table 3.3.2-14 Auxiliary Systems – Summary of Aging Management Evaluation – Diesel Generator System

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
Valve	*PB*	*Aluminum*	*Plant Indoor Air (Int)*	*Loss of material*	*Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B2.1.22)*	*VII.F2-12*	*3.3.1.27*	*E*
Valve	*PB*	*Aluminum*	*Plant Indoor Air (Ext)*	*None*	*None*	*VII.J-1*	*3.3.1.95*	*A*
Valve	*PB*	*Copper Alloy (> 15% Zinc)*	*Lubricating Oil (Int)*	*Loss of material*	*Lubricating Oil Analysis (B2.1.23) and One-Time Inspection (B2.1.16)*	*VII.H2-10*	*3.3.1.26*	*B*
Valve	*PB*	*Copper Alloy (> 15% Zinc)*	*Plant Indoor Air (Ext)*	*None*	*None*	*V.F-3*	*3.2.1.53*	*A*

3.4.2.1.1 Turbine Steam Supply System

Aging Management Programs

The following aging management programs manage the aging effects for the turbine steam supply system component types:

- *Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks (B2.1.42)*

3.4.2.1.3 Feedwater System

Aging Effects Requiring Management

The following saltwater and chlorination system aging effects require management:

- *Wall thinning due to erosion*

3.4.2.1.4 Condensate System

Aging Management Programs

The following aging management programs manage the aging effects for the condensate system component types:

- *Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks (B2.1.42)*

3.4.2.1.5 Auxiliary Feedwater System

Aging Effects Requiring Management

The following saltwater and chlorination system aging effects require management:

- *Wall thinning due to erosion*

Aging Management Programs

The following aging management programs manage the aging effects for the saltwater and chlorination system component types:

- *Flow-Accelerated Corrosion (B2.1.6)*

Table 3.4.2-1 Steam and Power Conversion System – Summary of Aging Management Evaluation – Turbine Steam Supply System

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
Expansion Joint	*LBS*	*Nickel Alloys*	*Plant Indoor Air (Ext)*	*None*	*None*	*VIII.I-9*	*3.4.1.41*	*A*
Expansion Joint	*LBS*	*Nickel Alloys*	*Steam (Int)*	*Loss of material*	*Water Chemistry (B2.1.2) and One-Time Inspection (B2.1.16)*	*VIII.B1-1*	*3.4.1.37*	*E, 4*

4.7.5 Inservice Flaw Growth Analyses that Demonstrate Structural Stability for 40 Years

Summary Description

The ISI procedure states that a fracture mechanics analysis, in accordance with ASME Section XI Code, Subsection IWB-3600, must be completed if a flaw acceptance criterion is not met as outlined in the corresponding test procedure. These analyses depend on a specified number of operating years, and thus may be TLAAs for DCPP.

Analysis

Unit 2 RHR Piping Weld RB-119-11

During a routine inservice inspection prior to DCPP Unit 2 Refueling Outage 13 (2R13) in 2006, a circumferential flaw was identified in Weld RB-119-11 of the residual heat removal (RHR) system.

The observed flaw did not meet the Section XI acceptance standards of Table IWB-3514-2. Consequently, the indication was evaluated per the guidelines of Section XI, IWB-3640. A conservative fatigue crack growth evaluation was then performed to determine the adequacy of the piping system for continued operation. The evaluation was submitted to the NRC for review, as required by the Code, in PG&E Letter DCL-06-069. The 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 cycle assumptions used in the analysis are conservative compared to the DCPP original design cycles described in Section 4.3.1.1. The DCPP licensing basis assumes 250 heatups and 250 cooldowns for a 50 year plant life.

Since the analysis indicates that the allowable flaw depth will not be reached for the next 40 years of plant operation beginning in October 2006, the flaw evaluation of RHR Weld RB-119-11 will remain valid for the period of extended operation in accordance with 10 CFR 54.21l(1)(i).

Unit 2 Auxiliary Feedwater Piping Line 567

During Unit 2 Refueling Outage 8 (2R8), while performing a non-routine surface examination prior to maintenance, DCPP identified a flaw indication in the auxiliary feedwater pump recirculation header Line 567, that exceeds Section XI, Table IWB-3410-1 criteria. The flaw has been accepted by analysis by meeting the

allowable size criteria of IWB-3620 and IWB-3610 and was submitted to the NRC in PG&E Letter DCL-99-136.

The numbers of thermal and seismic cycles used in the analysis are consistent with or more conservative than the DCPP 50-year design basis described in FSAR Table 5.2-4. The assumed transients are consistent with or bounded by the 50 year licensing basis. The number of transients will be monitored by the enhanced Fatigue Management Program. The enhanced Fatigue Management Program provides assurance that the fatigue crack growth analysis will be managed for the period of extended operation in accordance with 10 CFR 54.21l(1)(iii).

Unit 1 RHR Piping Weld WIC-95

During Unit 1 Refueling Outage 9 (1R9), while performing an inservice inspection, DCPP identified a weld flaw indication located in an ASME Class 2 portion of the RHR injection Line 985 to hot legs 1 and 2 at weld WIC-95. The indication exceeded the Section XI, Table IWC-3410-1 criteria. The flaw has been accepted by analysis in accordance with IWB-3410 and was submitted to the NRC in PG&E Letter DCL-97-086.

The number of seismic cycles used in the analysis is consistent with the DCPP 50-year design basis described in FSAR Table 5.2-4. There have been no occurrences of a DE, DDE, or Hosgri seismic event at 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 is sufficient to the end of the period of extended operation. Therefore, the analysis is valid for the period of extended operation, in accordance with 10 CFR 54.21l(1)(i).

Units 1 and 2 Safety Injection Pumps Vent and Drain Socket Welds

In 2014, PG&E requested NRC approval of Inservice Inspection Request for Alternative REP-SI for DCPP, Units 1 and 2. To support this request, a stress and fracture mechanics evaluation was performed to determine the adequacy of the socket welds associated with ASME Class 2 Safety Injection (SI) Pumps 1-1, 1-2, and 2-1 vent and drain connections. The evaluation was submitted to the NRC for review, as part of the relief request, in PG&E Letter DCL-14-060, dated July 21, 2014. Postulated flaws were evaluated using a fracture mechanics approach analogous to the methods of ASME Code Section XI. This relief request was approved for the remaining life of the subject SI Pumps, including the duration of the current operating licenses plus a license renewal period of 20 years [Reference 42]. Since the evaluation is based on the 60-year operating period, the TLAA covers the period of extended operation and is dispositioned under 10 CFR 54.21l(1)(i).

For the postulated crack analysis, 7,000 thermal transient cycles (pump starts), 400 DE cycles (20 events with 20 cycles per event), and 20 Hosgri earthquake cycles (1 event with 20 cycles) were assumed. Using a conservative projection of 1,400 SI Pump start cycles for a 60 year plant life, the 7,000 thermal transient cycles assumed in the postulated crack analysis during 60 years of operation is conservative. The number of seismic cycles used in the analysis is consistent with the DCPP 50-year design basis described in FSAR Table 5.2-4. There have been no occurrences of a DE or Hosgri seismic event at DCPP during the first 20 plus years of operation. Therefore, the seismic cycles in the postulated crack analysis for the 50-year design basis number of DE and Hosgri events is sufficient to the end of the period of extended operation. The analysis is valid for the period of extended operation in accordance with 10 CFR 54.21l(1)(i).

Unit 2 Pressurizer Safety and Spray Nozzle Welds

As stated in LRA Section 4.7.2, during Unit 2 Refueling Outage 14 (2R14, Spring 2008), Alloy 690 structural weld overlays were completed on Alloy 82/182 welds attaching the surge, spray, and relief valve nozzles to the safe ends, and the safe ends to the connecting piping. During the seventeenth Unit 2 Refueling Outage (2R17), while performing inservice inspections, DCPP identified weld flaw indications located at Unit 2 structural weld overlays for the pressurizer safety nozzles A and B, and pressurizer spray nozzle.

Conservative fatigue crack growth evaluations were then performed to determine the adequacy of the piping system for continued operation. The evaluations were submitted to the NRC for review, as part of a relief request, in PG&E Letter DCL-14-028, dated April 7, 2014. This relief request was approved for the service life of the structural weld overlays [Reference 43]. The service life for the pressurizer safety and spray nozzle structural weld overlays is based on operating for 38 years from the date the structural weld overlays were completed, i.e. until 2046. 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 cycle assumptions used in the analyses are consistent with those transients used in the pressurizer structural weld overlay (LRA Section 4.7.2 and PG&E Letter DCL-10-120). Per LRA Table A4-1, Commitment 38, the plant transient cycles related to the structural weld overlay fatigue crack growth analyses are included in the existing plant transient monitoring program.

Since the analyses indicate that the allowable flaw depth will not be reached for the remaining plant life, the flaw evaluations of the pressurizer safety and spray nozzle structural weld overlays will remain valid for the period of extended operation in accordance with 10 CFR 54.21l(1)(i).

Unit 1 Pressurizer Spray Line Pipe Weld WIB-378

During Unit 1 Refueling Outage 19 (1R19), while performing an inservice inspection, DCPP identified a weld flaw indication located in an ASME Code Class 1 pressurizer spray line pipe weld WIB-378. The indication exceeded the Section XI, Table IWB-3514-2 criteria. The flaw has been accepted by analysis in accordance with IWB-3600 and was submitted to the NRC in PG&E Letter DCL-15-131, dated November 3, 2015.

A fatigue crack growth evaluation was performed to determine the adequacy of the weld for continued operation. The indication was a planar flaw oriented circumferentially and was assumed to be ID connected for conservatism. The service life for Weld WIB-378 is based on operating through the period of extended operation. Thus, the evaluation encompassed a 60-year plant life and the analysis will remain valid for the Unit 1 period of extended operation.

The number of transient cycles used in the analysis is consistent with or more conservative than the DCPP 50-year design basis described in FSAR Table 5.2-4.

Because the evaluation indicates that the allowable flaw depth will not be reached for the remaining plant life and the assumed transients are consistent with or more conservative than the DCPP 50-year design basis described in FSAR Table 5.2-4, the flaw evaluation of the pressurizer spray line pipe weld WIB-378 will remain valid for the period of extended operation in accordance with 10 CFR 54.21l(1)(i).

Disposition: Validation, 10 CFR 54.21l(1)(i); and Aging Management, 10 CFR 54.21l(1)(iii)

Validation – Flaw Evaluation of Unit 2 RHR Piping Weld RB-119-11

The result indicates that the allowable flaw depth will not be reached for the next 40 years of plant operation beginning in October 2006. Therefore, the flaw evaluation of RHR Weld RB-119-11 will remain valid for the period of extended operation in accordance with 10 CFR 54.21l(1)(i).

Validation – Flaw Evaluation of Unit 1 RHR Weld WIC-95

There have been no occurrences of a DE, DDE, or Hosgri seismic event at 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 is sufficient to the end of the period of extended operation. Therefore, the analysis is valid for the period of extended operation, in accordance with 10 CFR 54.21l(1)(i).

Validation – Units 1 and 2 Safety Injection Pumps Vent and Drain Socket Welds

Using a conservative projection of 1,400 SI Pump start cycles for a 60 year plant life, the 7,000 thermal transient cycles assumed in the postulated crack analysis during 60 years of operation is conservative. There have been no occurrences of a DE or Hosgri seismic event at DCPP during the first 20 plus years of operation. Therefore, the seismic cycles in the postulated crack analysis for the 50-year design basis number of DE and Hosgri events is sufficient to the end of the period of extended operation. The analysis is valid for the period of extended operation in accordance with 10 CFR 54.21l(1)(i).

Validation – Flaw Evaluation of Unit 2 Pressurizer Safety and Spray Nozzle Welds

The results indicate that the allowable flaw depth will not be reached for the remaining plant life. Therefore, the flaw evaluation of Unit 2 Pressurizer Safety and Spray Nozzle Welds will remain valid for the period of extended operation in accordance with 10 CFR 54.21l(1)(i).

Validation – Flaw Evaluation of Unit 1 Pressurizer Spray Line Pipe Weld WIB-378

The results indicate that the allowable flaw depth will not be reached for the remaining plant life. Therefore, the flaw evaluation of Unit 1 Pressurizer Spray Line Pipe Weld WIB-378 will remain valid for the period of extended operation in accordance with 10 CFR 54.21l(1)(i).

Aging Management – Unit 2 Auxiliary Feedwater Piping Line 567

The Metal Fatigue of the Reactor Coolant Pressure Boundary program (B3.1) monitors fatigue design transients including the transients assumed in the fatigue crack growth analyses for the Unit 2 auxiliary feedwater piping Line 567. The program provides assurance that the fatigue crack growth analysis will be managed for the period of extended operation in accordance with 10 CFR 54.21l(1)(iii).

4.9 REFERENCES

42. US NRC Letter. From Michael T. Markley, Chief, Plant Licensing Branch IV-1, Division of Operating Reactor Licensing, Office of Nuclear Reactor Regulation; to Mr. Edward D. Halpin, Senior Vice President and Chief Nuclear Officer, DCPP. "Diablo Canyon Power Plant, Unit Nos. 1 and 2 – Request for Alternative REP-SI, Revision 2, Proposed Alternative to Requirements for Repair/Replacement Activities for Certain Safety Injection Pump Welded Attachments (TAC Nos. MF4476 and MF4477)." 15 July 2015. (ML15187A035)

43. US NRC Letter. From Michael T. Markley, Chief, Plant Licensing Branch IV-1, Division of Operating Reactor Licensing, Office of Nuclear Reactor Regulation; to Mr. Edward D. Halpin, Senior Vice President and Chief Nuclear Officer, DCPP. "Diablo Canyon Power Plant, Unit No. 2 – Inservice Inspection Program Relief Request SWOL-REP-1 U2 for Approval of an Alternative to the ASME Code, Section XI, for Preemptive Full Structural Weld Overlays (TAC No. MF3891)." 14 October 2014. (ML14255A232)

A1.13 FIRE WATER SYSTEM

The Fire Water System program manages loss of material due to corrosion, including MIC, fouling, flow blockage because of fouling, and loss of integrity for water-based fire protection systems and internal coatings/linings for the fire water storage tank within the scope of license renewal. Internal and external inspections and tests of fire protection equipment are performed consistent, with exceptions identified in PG&E Letters DCL-14-103, Enclosure 1, Attachment 7C, and DCL-15-121 with NFPA-25 (2011 edition). Testing or replacement of sprinklers that have been in place for 50 years is performed in accordance with NFPA-25 (2011 edition). Portions of the deluge systems that are normally dry but periodically subjected to flow and cannot be drained or allow water to collect will undergo augmented testing beyond that in NFPA-25 consisting of volumetric wall thickness examinations. The fire water system is managed by performing routine preventive maintenance, inspections and testing; operator rounds, performance monitoring, and reliance on the corrective action program; and system improvements to address aging and obsolescence issues. The fire water system is normally maintained at required operating pressure and is monitored such that loss of system pressure is immediately detected and corrective actions are initiated.

The Fire Water System program will conduct a flow test with air, water, or other medium through each open spray nozzle to verify that deluge systems nozzles are unobstructed. Water flow tests will verify that the deluge system provide full coverage of the equipment it protects. Visual inspections will be performed on firewater piping. Non-intrusive follow-up volumetric examinations will be performed if internal visual inspections detect surface irregularities to determine if wall thickness is within acceptable limits. Visual inspections will evaluate for the presence of sufficient foreign material to obstruct fire water pipe or sprinklers.

Inspections of the firewater tank will be performed to detect loss of material.

As discussed in PG&E Letter DCL-15-027, Enclosure 1, in response to LR-ISG-2013-01, the program consists of periodic visual inspections of the internal liner of the fire water storage tank exposed to raw water where loss of lining integrity could impact the components' and downstream components' current licensing basis intended function(s). For coated surfaces determined to not meet the acceptance criteria, physical testing is performed where physically possible (i.e., sufficient room to conduct testing) in conjunction with repair, replacement, or removal of the lining. The training and qualification of individuals involved in coating inspections are conducted in accordance with ASTM International Standards endorsed in RG 1.54 including guidance from the NRC associated with a particular standard.

The Fire Water program implements the recommendations in LR-ISG-2012-02, as discussed in PG&E Letters DCL-14-103, Enclosure 1, Attachments 7C, ~~and~~ DCL-15-121*, and DCL-15-150*, the recommendations in LR-ISG-2013-01, as discussed in PG&E Letter DCL-15-027

A1.14 FUEL OIL CHEMISTRY

The Fuel Oil Chemistry program manages loss of material on the internal surface of components in the emergency diesel fuel oil storage and transfer system, *portable diesel electric generator fuel oil tanks,* portable diesel driven fire pump fuel oil tanks, and portable caddy fuel oil tanks. The program includes (a) surveillance and monitoring procedures for maintaining fuel oil quality by controlling contaminants in accordance with applicable ASTM Standards, (b) periodic draining of water from fuel oil tanks, (c) visual inspection of internal surfaces during periodic draining and cleaning, (d) one-time ultrasonic wall thickness measurements of accessible portions of fuel oil tank bottoms , (e) sampling and analysis of new fuel oil before it is introduced into the fuel oil tanks, and (f) supplemental one-time inspections of a representative sample of components in systems that contain fuel oil by the One-Time Inspection program (A1.16).

A3.5.3 Inservice Flaw Growth Analyses that Demonstrate Structural Stability for 40 Years

The ISI procedure states that a fracture mechanics analysis, in accordance with ASME Code, Section XI, Subsection IWB-3600, must be completed if flaw acceptance criterion is not met as outlined in the corresponding test procedure. These analyses depend on a specified number of operating years, and thus may be TLAAs.

Unit 2 RHR Piping Weld RB-119-11

In 2006, a circumferential flaw was identified in DCPP Unit 2 Weld RB-119-11 of the residual heat removal (RHR) system. The observed flaw did not meet the Section XI acceptance standards of Table IWB-3514-2. Consequently, the indication was evaluated per the guidelines of Section XI, IWB-3640. A conservative fatigue crack growth evaluation was performed to determine the adequacy of continued operation of the piping system. The analysis is based on operating for 40 years from the date the flaw was identified and will be valid beyond the end of the period of extended operation for Unit 2 in accordance with 10 CFR 54.21l(1)(i).

Unit 2 Auxiliary Feedwater Piping Line 567

DCPP identified a flaw indication in the Unit 2 auxiliary feedwater pumps recirculation header Line 567, that exceeds Section XI, Table IWB-3410-1 criteria. The flaw has been accepted by analysis by meeting the allowable size criteria of IWB-3620 and IWB-3610.

The Metal Fatigue of Reactor Coolant Pressure Boundary program described in Section A2.1 monitors fatigue design transients including the transients assumed in the fatigue crack growth analyses and therefore will be managed for the period of extended operation in accordance with 10 CFR 54.21l(1)(iii).

Unit 1 RHR Piping Weld WIC-95

DCPP identified a weld flaw indication located in an ASME Class 2 portion of the Unit 1 residual heat removal injection Line 985 to hot legs 1 and 2 at weld WIC-95. The indication exceeded the Section XI, Table IWC-3410-1 criteria. The flaw has been accepted by analysis in accordance with IWB-3410.

The number of seismic cycles assumed in the analysis is sufficient for the period of extended operation. Therefore, the analysis is valid for the period of extended operation, in accordance with 10 CFR 54.21l(1)(i).

Units 1 and 2 Safety Injection Pumps Vent and Drain Socket Welds

In support of a relief request, a stress and fracture mechanics evaluation was performed to determine the adequacy of socket welds associated with ASME Class 2 Safety Injection (SI) Pumps 1-1, 1-2, and 2-1 vent and drain connections. Postulated flaws were evaluated using a fracture mechanics approach analogous to the methods of ASME Code Section XI.

The number of cycles used in the analysis is sufficient for the period of extended operation. Using a conservative projection of 1,400 SI Pump start cycles for a 60 year plant life, the 7,000 thermal transient cycles assumed in the postulated crack analysis during 60 years of operation is conservative. Therefore, the analysis is valid for the period of extended operation, in accordance with 10 CFR 54.21l(1)(i).

Unit 2 Pressurizer Safety and Spray Nozzle Welds

In 2013, laminar flaws were identified in DCPP Unit 2 structural weld overlays for pressurizer safety nozzles A and B, and pressurizer spray nozzle. Conservative fatigue crack growth evaluations were performed to determine the adequacy of continued operation of the piping system. The analyses are based on operating for 38 years from the date the structural weld overlays were completed and will be valid beyond the end of the period of extended operation for Unit 2 in accordance with 10 CFR 54.21l(1)(i).

Unit 1 Pressurizer Spray Line Pipe Weld WIB-378

In 2015, a circumferential flaw was identified in DCPP Unit 1 Pressurizer Spray Line Pipe Weld WIB-378. The observed flaw did not meet the Section XI acceptance standards of Table IWB-3514-2. Consequently, the indication was evaluated per the guidelines of Section XI, IWB-3600. A conservative fatigue crack growth evaluation was performed to determine the adequacy of continued operation of the weld. The analysis is based on operating through the period of extended operation and will remain valid Unit 1 in accordance with 10 CFR 54.21l(1)(i).

Table A4-1 License Renewal Commitments

Item #	Commitment	LRA Section	Implementation Schedule
3	Enhance the Fire Water System program: (a) Sprinkler heads in service for 50 years will be replaced or representative samples from one or more sample areas will be tested consistent with NFPA 25, *Inspection, Testing and Maintenance of Water-Based Fire Protection Systems, 2011 Edition* guidance. Test procedures will be repeated at 10-year intervals during the period of extended operation, for sprinkler heads that were not replaced prior to being in service for 50 years, to ensure that signs of degradation, such as corrosion, are detected prior to the loss of intended function, and (b) To perform non-intrusive follow-up volumetric examinations if internal visual inspections detect surface irregularities to determine if wall thickness is within acceptable limits. Visual inspections will evaluate for the presence of sufficient foreign material to obstruct fire water pipe or sprinklers (c) To be in conformance with LR-ISG-2012-02, Section C as discussed in PG&E Letter DCL-14-103, Enclosure 1, Attachment 7C. (d) To be in conformance with LR-ISG-2013-01 as discussed in PG&E Letter DCL-15-027, Enclosure 1. *(e) Test deluge system nozzles in accordance with the 2011 Edition of NFPA 25, Section 10.3.4.3.1.*	B2.1.13	Program is implemented 5 years before the period of extended operation. Inspections of wetted normally dry piping segments that cannot be drained or that allow water to collect begin 5 years before the period of extended operation. Internal linings inspections begin no later than the last refueling outage before the period of extended operation. The program's remaining inspections begin during the period of extended operation
34	The DCPP work control procedure will be revised to include evaluation of reinforced concrete exposed during excavations.	B1.2.32	*Complete. PG&E Letter DCL-15-150.* ~~Prior to the period of extended operation~~

Table A4-1 License Renewal Commitments

Item #	Commitment	LRA Section	Implementation Schedule
60	PG&E will enhance provisions in the HVAC ducting from the 480V switchgear room that allow water to drain from the exhaust ducting so water cannot enter the 480V switchgear room.		*Complete. PG&E Letter DCL-15-150.*. ~~Prior to the period of extended operation~~
73	The NRC SE for MRP-227 contains eight action items for applicants/licensees to consider. Responses to the applicable aging management program plant-specific action items, conditions, and limitations identified in the NRC SE, Revision 1, on MRP-227 will be submitted to the NRC by December 2015. Reference DCL-14-103, Enclosure 1, Attachment 4.	B2.1.41	*Complete. PG&E Letter DCL-15-150.* ~~December 2015~~

WCAP-17462-NP, Revision 1

Program Plan for Aging Management of Reactor Vessel Internals at Diablo Canyon Power Plant Unit 1

MRP-227-A Applicability Guideline for Diablo Canyon Power Plant Westinghouse Pressurized Water Reactor Design

Background

The Nuclear Regulatory Commission (NRC) staff has determined that additional information, as discussed in References 1 and 2, should be provided by licensees to verify the applicability of MRP-227-A (Reference 6). The two specific generic issues that need to be addressed are summarized as follows:

1. Do the reactor internals have any non-weld or bolting austenitic stainless steel components with 20 percent cold work or greater, and if so, do the affected components have operating stresses greater than 30 ksi?

2. Does the plant have atypical fuel design or fuel management that could render the assumptions of MRP-227-A, regarding core loading/core design, non-representative?

PG&E Response to Question 1

Diablo Canyon Power Plant (DCPP) Units 1 and 2 reactor internals components have been evaluated according to industry guideline MRP 2013-025 (Reference 3), as well as to the MRP-191 (Reference 4) industry generic component listings and screening criteria (including consideration of cold work as defined in MRP-175 (Reference 5), noting the requirements of Section 3.2.3). In addition to consideration of the material fabrication, forming, and finishing process, a general screening definition of "severe cold work" [a resulting reduction in wall thickness (material stock thickness) of 20 percent] was applied as an evaluation limit.

The evaluation included a review of all plant modifications affecting reactor internals and the plant operating history. The components were procured according to American Society for Testing and Materials International of American Society of Mechanical Engineers material specifications that were callouts in the original plant construction drawings. Thus, material identification based on the material callouts and notes in the component drawings was an efficient and reasonable approach to identify the material of construction of components for DCPP Units 1 and 2.

Based on the specifications used in the DCPP Units 1and 2 plant component drawings, it was possible to bin the reactor internals components into five material categories identified in MRP 2013-025. DCPP Units 1 and 2 components were binned according to the following categories for the materials used in the component fabrication.

Categories based on MRP 2013-025 include:

- Cast austenitic stainless steel (CASS) (Category 1)
- Hot-formed austenitic stainless steel (Category 2)
- Annealed austenitic stainless steel (Category 3)
- Fasteners austenitic stainless steel (Category 4)
- Cold-formed austenitic stainless steel without subsequent solution annealing (Category 5)

The potential for cold work is directly controlled by the materials specifications. Essentially, all of the components that are binned (based on their specified materials) as Categories 1, 2, and 3 are non-cold worked; therefore, they have less than 20 percent cold work according to NRC criterion. Similarly, any component binned under Category 5 has the potential to contain greater than 20 percent cold work. Category 4 materials are fasteners that may have been intentionally strain-hardened.

The strain hardening according to guidelines should have been intentionally restricted to less than 20 percent. Material definitions in drawings identify maximum yield stress restrictions on these materials, which allows for the identification of the cold work level. In some cases, however, these restrictions are not present on drawings. Restrictions or limitations on the material yield stress (e.g., a maximum of 90 ksi) would indicate that the material cold work would be limited to less than 20 percent. In the absence of a maximum restriction yield stress of strain-hardened material, a conservative approach was taken to indicate the potential for greater than 20 percent cold work.

Where multiple options existed for a component or assembly, the bounding condition was taken as the option that had the greater potential to include greater than 20 percent cold work. This option was then employed in the assessment of the component and was selected for the purposes of the Westinghouse evaluation. In some instances, sequential fabrication would appear to mitigate any potential for cold work; however, since the historical record was not detailed, the potential is noted, but a conservative approach was selected for the Westinghouse evaluation.

The evaluation, performed consistently with MRP 2013-025, concluded that the reactor internals Categories 1, 2, and 3 (non-bolting) components at DCPP Units 1 and 2 contain no cold work greater than 20 percent as a result of material specification and controlled fabrication construction. Category 4 components were already assumed to have the potential for cold work in the MRP-191 generic assessments. No Category 5 components with severe cold work were identified for DCPP Units 1 and 2.

The detailed evaluation for the DCPP Units 1 and 2 cold work assessments concluded that the plant-specific fabrication and design was consistent with the

MRP-191 basis, and that the MRP-227-A (Reference 6) sampling inspection aging management requirements, as related to cold work, are directly applicable to DCPP Units 1 and 2.

The inspection sampling requirements for aging management outlined in MRP-232 (Reference 7) are based on the assumptions of MRP-191 and MRP-227-A. Therefore, MRP-232 calls for the demonstration that the plant specific materials, fabrication, and design meet the assumptions inherent in MRP-191 and MRP-227-A. The detailed Westinghouse evaluation of DCPP Units 1 and 2 material fabrication and design has concluded that no non-fastener materials of greater than 20 percent cold work were used in construction, and that the inspection sampling approach of MRP-232 is applicable to DCPP Units 1 and 2.

PG&E Response to Question 2

As stated in MRP 2013-025, to demonstrate plant-specific applicability of the MRP-227-A sampling inspection strategy for managing aging in reactor internals, licensees must demonstrate that the criteria of MRP-227-A, Section 2.4 are met, and that the neutron fluence and heat generation rates are within the range of the following variables summarized.

As detailed in WCAP-17462-NP, Revision 1, "Program Plan for Aging Management of Reactor Vessel Internals at Diablo Canyon Power Plant Unit 1," and WCAP-17463-NP, Revision 1, "Program Plan for Aging Management of Reactor Vessel Internals at Diablo Canyon Power Plant Unit 2," for DCPP Units 1 and 2, respectively, the criteria specified in MRP-227-A, Section 2.4 has been demonstrated as follows. The MRP-227-A, Section 2.4 assumptions are stated first, followed by a description of how the assumptions are addressed at DCPP Units 1 and 2.

The assumptions from MRP-227-A, Section 2.4 are as follows:

- *30 years of operation with high leakage core loading patterns (fresh fuel assemblies loaded in peripheral locations) followed by implementation of a low leakage fuel management strategy for the remaining 30 years of operation;*

  DCPP Units 1 and 2 fuel management programs changed from a high- to low-leakage core loading pattern prior to 30 years of operation.

- *Base load operation, i.e., typically operates at fixed power levels and does not usually vary power on a calendar or load demand schedule.*

  DCPP Units 1 and 2 operate as base-load units.

- *No design changes beyond those identified in general industry guidance or recommended by the original vendors.*

  MRP-227-A states that the recommendations are applicable to all U.S. PWR operating plants as of May 2007 for the three designs considered. There have been no modifications to reactor internals components at DCPP Units 1 or 2 since May 2007.

Based on the applicability, as stated for DCPP Units 1 and 2, the criteria of MRP-227-A, Section 2.4 are met for DCPP Units 1 and 2.

In addition to the requirement to demonstrate that the criteria of MRP-227-A, Section 2.4 are met, MRP 2013-025 requires that the neutron fluence and heat generation rates for DCPP Units 1 and 2 are within the range of the limiting threshold values defined in MRP 2013-025. The limiting threshold values defined for Westinghouse plants are:

- Average core power density less than 124 Watts/cm^3
- Heat generation figure of merit (F) less than or equal to 68 Watts/cm^3
- Active fuel to upper core plate distance greater than 12.2 inches

PG&E is currently in the process of evaluating the DCPP Units 1 and 2 reactor internals components with regard to fuel designs and fuel management according to guidance provided in MRP 2013-025. PG&E is currently scheduled to complete and submit to the NRC the results of this evaluation for DCPP Units 1 and 2 by March 31, 2016 (see Enclosure 4).

References

1. U.S. Nuclear Regulatory Commission Letter, "Summary of January 22-23, 2013, Closed Meeting with the Electric Power Research Institute and Westinghouse," February 21, 2013. (ADAMS: ML13042A048/ml13043A062).

2. U.S. Nuclear Regulatory Commission Letter, "Summary of February 25, 2013, Telecom with the Electric Power Research Institute and Westinghouse Electric Company," March15, 2013. (ADAMS: ML13067A262).

3. EPRI Letter, MRP 2013-025, "MRP-227-A Applicability Template Guideline," October 14, 2013.

4. Materials Reliability Program: Screening, Categorization and Ranking of PWR Internals Components for Westinghouse and Combustion Engineering PWR Design (MRP-191). EPRI, Palo Alto, CA: 2006. 1013234.

5. Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175). EPRI, Palo Alto, CA: 2005. 1012081.

6. Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A). EPRI, Palo Alto, CA: 2011. 1022863.

7. Materials Reliability Program: Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internal Components (MRP-232, Rev. 1). EPRI, Palo Alto, CA: 2012. 1021029.

Regulatory Commitments

Pacific Gas and Electric Company (PG&E) is making the following new and revised regulatory commitments (as defined by NEI 99-04) in this submittal:

Commitment	Due Date
PG&E will update the cathodic protection design and installation action plan and associated licensing basis by March 31, 2016.	March 31, 2016
PG&E is currently scheduled to complete and submit to the NRC an evaluation of the Units 1 and 2 reactor internals components with regard to fuel designs and fuel management according to guidance provided in MRP 2013-025.	March 31, 2016