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PG&E Letter DCL-14-074

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

10 CFR 50.55a

Docket No. 50-275, OL-DPR-80
Diablo Canyon Unit 1

ASME Section XI Inservice Inspection Program Request for Alternative RPV-U1-Extension to Allow Use of Alternate Reactor Inspection Interval Requirements

References:

1. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 2001 Edition with Addenda through 2003.

2. Westinghouse Report WCAP-16168-NP-A, Revision 3, "Risk-Informed Extension of the Reactor Vessel In-service Inspection Interval," October 2011 (ADAMS Accession Number ML113060207).

3. PWROG Letter OG-10-238, "Revision to the Revised Plan for Plant Specific Implementation of Extended Inservice Inspection Interval per WCAP-16168-NP, Revision 1, 'Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval.' PA-MSC-0120," July 12, 2010 (ADAMS Accession Number ML11153A033).

Dear Commissioners and Staff:

In accordance with Title 10 of the Code of Federal Regulations (10 CFR), Part 50.55a, "Codes and Standards," paragraph (a)(3)(i), Pacific Gas and Electric Company (PG&E) proposes an alternative to the requirements of the ASME B&PV Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," (Reference 1) as applicable to Diablo Canyon Power Plant (DCPP), Unit 1.

PG&E's Request for Alternative (RFA) RPV-U1-Extension pertains to the requirements of ASME Code, Section XI, Paragraph IWB-2412, "Inspection Program B," requiring volumetric examination of essentially 100 percent of reactor pressure-retaining welds identified in Table IWB-2500-1 once each 10-year interval.

PG&E proposes to extend the DCPP Unit 1 reactor vessel inspection interval from 10 years to 20 years in accordance with WCAP-16168-NP-A, Revision 3, "Risk-Informed Extension of Reactor Vessel In-Service Inspection Interval" (Reference 2).

The proposed interval extension is a deviation from the latest revised implementation plan, OG-10-238 (Reference 3). The impact to the intent of the implementation plan in OG-10-238 is considered by PG&E to be minimal since a majority of the reactor volumetric inspections scheduled for the DCPP Unit 1 third ISI interval were completed to the extent feasible in the recent Unit 1 eighteenth refueling outage (1R18), including approximately 84 percent of the linear footage of beltline region welds. The results of the 1R18 beltline inspections performed are essentially identical to those of the previous examinations with only one indication in the beltline region detected. This indication was found to be acceptable per the applicable ASME code. Unresolved inspection robot control issues in 1R18 led to the deferral of the remainder of the examinations to Unit 1 nineteenth refueling outage (1R19), the last outage in the third inspection interval for DCPP Unit 1.

PG&E has concluded that the proposed alternative provides an acceptable level of quality and safety, in accordance with 10 CFR 50.55a(a)(3)(i). The requisite supporting information and basis for use are provided in the enclosed RFA RPV-U1-Extension for DCPP, Unit 1.

PG&E requests approval of the RFA by August 1, 2015, to support 1R19 planning.

This RFA is similar to recent alternatives granted for Duke Energy's McGuire Nuclear Station, Unit 2, by letter dated September 6, 2012 (ADAMS Accession No. ML12249A175), and Dominion's Surry Power Station, Units 1 and 2, by letter dated April 30, 2013 (ADAMS Accession No. ML13106A140).

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

If you have any questions regarding the information enclosed, or other ISI program activities, please contact Mr. Tom Baldwin at (805) 545-4720.

Sincerely,

Barry S. Allen

Barry S. Allen
Site Vice President

rntt/4231/50621143
Enclosure
cc: Diablo Distribution
cc/enc: Marc L. Dapas, NRC Region IV Administrator
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Proposed Alternative for Diablo Canyon Power Plant Unit 1 In Accordance with 10 CFR 50.55a(a)(3)(i) -Alternative Provides Acceptable Level of Quality and Safety-

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Proposed Alternative for Diablo Canyon Power Plant Unit 1
In Accordance with 10 CFR 50.55a(a)(3)(i)
-Alternative Provides Acceptable Level of Quality and Safety-

1. ASME Code Component(s) Affected

The affected component is the Diablo Canyon Power Plant (DCPP) Unit 1 reactor vessel (RV); specifically, the following American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code, Section XI (Reference 1) examination categories and item numbers covering examinations of the RV. These examination categories and item numbers are from IWB-2500 and Table IWB-2500-1 of the ASME BPV, Code Section XI.

Category B-A welds are defined as "Pressure Retaining Welds in Reactor Vessel." Category B-D welds are defined as "Full Penetration Welded Nozzles in Vessels."

Examination Category	Item No.	Description
B-A	B1.11	Circumferential Shell Welds
B-A	B1.12	Longitudinal Shell Welds
B-A	B1.21	Circumferential Head Welds
B-A	B1.22	Meridional Head Welds
B-A	B1.30	Shell-to-Flange Weld
B-D	B3.90	Nozzle-to-Vessel Welds

(Throughout this request the above examination categories are referred to as "the subject examinations" and the ASME BPV Code, Section XI, is referred to as "the Code.")

2. Applicable Code Edition and Addenda

The DCPP Unit 1 third 10-year inservice inspection (ISI) interval is based on the ASME Code Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 2001 Edition through 2003 Addenda (Reference 1). ASME Code Section XI, 2001 Edition without Addenda applies to ultrasonic examinations performed per ASME Code Section XI, Appendix VIII requirements. The applicable ASME Section XI Code for the fourth 10-year ISI interval for DCPP will be the 2007 Edition with 2008 Addenda.

3. Applicable Code Requirement

IWB-2412, Inspection Program B, requires volumetric examination of essentially 100 percent of reactor vessel pressure-retaining welds identified in Table IWB-2500-1 once each 10-year interval. The DCPP Unit 1 third 10-year ISI interval is scheduled to end on May 6, 2015.

4. Reason for Request

An alternative is requested from the requirement of IWB-2412, Inspection Program B, that volumetric examination of essentially 100 percent of reactor vessel pressure-retaining Examination Category B-A and B-D welds be performed once each 10-year interval. Extension of the interval between examinations of Category B-A and B-D welds from 10 years to up to 20 years will result in a reduction in man-rem exposure and examination costs.

5. Proposed Alternative and Basis for Use

Pacific Gas and Electric Company (PG&E) proposes not to perform the ASME Code required volumetric examination of the DCPP Unit 1 reactor vessel full penetration pressure-retaining Examination Category B-A and B-D welds for the third ISI, which was scheduled for 2014. PG&E will perform the third ASME Code required volumetric examination of the DCPP Unit 1 reactor vessel full penetration pressure-retaining Examination Category B-A and B-D welds in the fourth ISI interval in 2025 plus or minus one refuelling outage. The proposed inspection date is a deviation from the latest revised implementation plan, OG-10-238 (Reference 2). The impact to the implementation plan in OG-10-238 would increase the number of inspections in 2025 from two to three and decrease the number of inspections in 2015 from six to five. Based on Figures 3 and 4 of OG-10-238, this proposed inspection schedule is considered to have a minor impact on the future inspection plan and the distribution of inspections over time.

In accordance with 10 CFR 50.55a(a)(3)(i), an alternate inspection interval is requested on the basis that the current interval can be revised with negligible change in risk by satisfying the risk criteria specified in Regulatory Guide 1.174 (Reference 3).

The methodology used to conduct this analysis is based on that defined in the study WCAP-16168-NP-A, Revision 3, "Risk-Informed Extension of the Reactor Vessel In-service Inspection Interval" (Reference 4). This study focuses on risk assessments of materials within the beltline region of the RV wall. The results of the calculations for DCPP Unit 1 were compared to those obtained from the Westinghouse pilot plant evaluated in WCAP-16168-NP-A, Revision 3. Appendix A of the WCAP identifies the parameters to be compared. Demonstrating that the parameters for DCPP Unit 1 are bounded by the results of the Westinghouse pilot plant qualifies DCPP Unit 1 for an ISI interval extension. Table 1 below lists the critical parameters investigated in the WCAP and compares the results of the Westinghouse pilot plant to those of DCPP Unit 1. Tables 2 and 3 provide additional information that was requested by the NRC and included in Appendix A of Reference 4.

Table 1: Critical Parameters for the Application of Bounding Analysis for DCPP Unit 1

Parameter	Pilot Plant Basis	Plant-Specific Basis	Additional Evaluation Required?
Dominant Pressurized Thermal Shock (PTS) Transients in the NRC PTS Risk Study are Applicable	NRC PTS Risk Study (Reference 5)	PTS Generalization Study (Reference 6)	No
Through-Wall Cracking Frequency (TWCF)	1.76E-08 Events per year (Reference 4)	5.56E-09 Events per year (Calculated per Reference 4)	No
Frequency and Severity of Design Basis Transients	7 heatup/cooldown cycles per year (Reference 4)	Bounded by 7 heatup/cooldown cycles per year	No
Cladding Layers (Single/Multiple)	Single Layer (Reference 4)	Single Layer	No

Table 2 below provides a summary of the latest reactor vessel inspection for DCPP Unit 1 and an evaluation of the recorded indications. This information confirms that satisfactory examinations have been performed on the DCPP Unit 1 RV.

Table 2: Additional Information Pertaining to Reactor Vessel Inspection for DCPP Unit 1	
Inspection methodology:	The 2005 ISI was conducted in accordance with the ASME Code, Section XI and Section V, 1989 Edition, with no Addenda, as modified by 10 CFR 50.55a(b)(2)(xiv, xv, and xvi). Examinations of Category B-A and B-D welds were performed to ASME Section XI, Appendix VIII, 1995 Edition with 1996 Addenda, as modified by 10 CFR 50.55a(b)(2)(xiv, xv, and xvi). Future ISIs will be performed to ASME Section XI, Appendix VIII requirements.
Number of past inspections:	Two complete 10-year ISIs have been performed.
Number of indications found:	There was one indication identified in the beltline region during the most recently completed ISI (2005). This subsurface indication is located in the intermediate shell to lower shell circumferential weld (Item 20 in Table 3). The indication is acceptable per Table IWB-3510-1 of Section XI of the ASME Code. This indication is not within the inner $1/10^{th}$ or 1 inch of the reactor vessel thickness; therefore, it is inherently acceptable per the requirements of the Alternate PTS Rule, 10 CFR 50.61a (Reference 7).
Proposed inspection schedule for balance of plant life:	The third ISI was scheduled for 2014. This inspection will be performed in 2025 plus or minus one refuelling outage. The proposed inspection date for DCPP Unit 1 is a deviation from the schedule presented in OG-10-238 (Reference 2). The impact to the implementation plan in OG-10-238 would increase the number of inspections in 2025 from two to three and decrease the number of inspections in 2015 from six to five. Based on Figures 3 and 4 of OG-10-238, this proposed inspection schedule is considered to have a minor impact on the future inspection plan and the distribution of inspections over time.

Table 3 summarizes the inputs and outputs for the calculation of TWCF.

Table 3: Details of TWCF Calculation for DCPP Unit 1 at 54 Effective Full-Power Years (EFPY)

Inputs								
Reactor Coolant System Temperature, T_C [°F]:			N/A	Inter. & Lower Shell T_{wall} [inches]:				8.85
				Upper Shell T_{wall} [inches]:				10.97
No.	Region and Component Description	Material Heat No.	Cu[(1)] [wt%]	Ni[(1)] [wt%]	R.G. 1.99 Pos.[(1)]	CF[(1)] [°F]	$RT_{NDT(u)}$[(1)] [°F]	Fluence [10^{19} n/cm^2, E>1.0 MeV][(1)]
1	Upper Shell Plate B4105-1	C-2624-1	0.12	0.56	1.1	82.2	28	0.0341
2	Upper Shell Plate B4105-2	C-2624-2	0.12	0.57	1.1	82.4	9	0.0341
3	Upper Shell Plate B4105-3	C-2608-2B	0.14	0.56	1.1	98.2	14	0.0341
4	Intermediate Shell Plate B4106-1	C-2884-1	0.125	0.53	1.1	85.3	-10	2.02
5	Intermediate Shell Plate B4106-2	C-2854-2	0.120	0.50	1.1	81.0	-3	2.02
6	Intermediate Shell Plate B4106-3	C-2793-1	0.086	0.476	1.1	55.2	30	2.02
7	Lower Shell Plate B4107-1	C-3121-1	0.13	0.56	1.1	89.8	15	2.01
8	Lower Shell Plate B4107-2	C-3131-2	0.12	0.56	1.1	82.2	20	2.01
9	Lower Shell Plate B4107-3	C-3131-1	0.12	0.52	1.1	81.4	-22	2.01
10	Upper Shell Long. Weld 1-442A	27204/12008	0.19	0.97	1.1	215.7	-20	0.0245
11	Upper Shell Long. Weld 1-442B	27204/12008	0.19	0.97	1.1	215.7	-20	0.0149
12	Upper Shell Long. Weld 1-442C	27204/12008	0.19	0.97	1.1	215.7	-20	0.0306
13	Intermediate Shell Long. Weld 2-442A	27204	0.203	1.018	2.1	214.1	-56	1.49
14	Intermediate Shell Long. Weld 2-442B	27204	0.203	1.018	2.1	214.1	-56	1.49
15	Intermediate Shell Long. Weld 2-442C	27204	0.203	1.018	2.1	214.1	-56	0.768
16	Lower Shell Long. Weld 3-442A	27204	0.203	1.018	2.1	214.1	-56	1.19
17	Lower Shell Long. Weld 3-442B	27204	0.203	1.018	2.1	214.1	-56	1.19
18	Lower Shell Long. Weld 3-442C	27204	0.203	1.018	2.1	214.1	-56	2.01
19	Upper To Inter. Shell Circ. Weld 8-442	13253	0.25	0.73	1.1	197.5	-56	0.0341
20	Inter. To Lower Shell Circ. Weld 9-442	21935	0.183	0.704	1.1	172.2	-56	2.01

Outputs						
Methodology Used to Calculate ΔT_{30}:			Regulatory Guide 1.99, Revision 2[(2)]			
	Controlling Material Region No. (From Above)	$RT_{MAX\text{-}XX}$ [°R]	Fluence [10^{19} Neutron/cm^2, E > 1.0 MeV]	FF (Fluence Factor)	ΔT_{30} [°F]	$TWCF_{95\text{-}XX}$
Limiting Axial Weld - AW	18	658.54	2.01	1.190	254.87	2.42E-09
Limiting Plate - PL	7	581.57	2.01	1.190	106.90	4.22E-13
Circumferential Weld - CW	20	608.66	2.01	1.190	204.99	0.00E+00
$TWCF_{95\text{-}TOTAL}(\alpha_{AW}TWCF_{95\text{-}AW} + \alpha_{PL}TWCF_{95\text{-}PL} + \alpha_{CW}TWCF_{95\text{-}CW})$:						5.56E-09

(1) Per Reference 8. The Position 2.1 chemistry factor (CF) value for the intermediate and lower shell longitudinal welds (heat 27204) incorporates all surveillance capsule data, including sister plant data from Palisades. Per Reference 8, the surveillance data for weld heat 27204 was deemed credible.

(2) Reference 9

6. Duration of Proposed Alternative

This request is applicable to the DCPP Unit 1 ISI Program for the third and fourth 10-year inspection intervals.

7. Precedents

- "Palo Verde Nuclear Generating Station, Units 1, 2, and 3 – Relief Request No. RR-40, Reactor Vessel Weld Examination Interval Extension (TAC Nos. ME1634, ME1635, and ME1636)," dated February 22, 2010 (ADAMS Accession Number ML100290415).
- "Safety Evaluation of Relief Requests to Extend the Inservice Inspection Interval for Reactor Vessel Examinations for Salem Nuclear Generating Station, Unit Nos. 1 and 2 (TAC Nos. ME1478, ME1479, ME1480 and ME1481)," dated February 22, 2010 (ADAMS Accession Number ML100491550).
- "Arkansas Nuclear One, Unit 2 – Request for Alternative ANO2-ISI-004, to Extend the Third 10-Year Inservice Inspection Interval for Reactor Vessel Weld Examinations (TAC No. ME2508)," dated September 21, 2010 (ADAMS Accession Number ML102450654).
- "Joseph M. Farley Nuclear Plant, Unit 2 (Farley Unit 2) – Relief Request for Extension of the Reactor Vessel Inservice Inspection Date to the Year 2020 (Plus or Minus One Outage) (TAC No. ME3010)," dated July 12, 2010 (ADAMS Accession Number ML101750402).
- "Three Mile Island Nuclear Station, Unit 1 (TMI-1) – Request to Extend the Inservice Inspection Interval for Reactor Vessel Weld and Internal Examinations, Proposed Alternative Request Nos. RR-09-01 and RR-09-02 (TAC Nos. ME2483 and ME2484)," dated September 21, 2010 (ADAMS Accession Number ML102390018).
- "Surry Power Station Units 1 and 2 – Relief Implementing Extended Reactor Vessel Inspection Interval (TAC Nos. ME8573 and ME8574)," dated April 30, 2013 (ADAMS Accession Number ML13106A140).
- "McGuire Nuclear Station, Unit 2, Relief 10-MN-002 to Extend the Inservice Inspection Interval for Reactor Vessel Category B-A and B-D Welds (TAC Nos. ME7329 and ME 7330)," dated September 6, 2012 (ADAMS Accession Number ML12249A175).

8. References

1. ASME Boiler and Pressure Vessel Code, Section XI, 2001 Edition through 2003 Addenda, American Society of Mechanical Engineers, New York
2. PWROG Letter OG-10-238, "Revision to the Revised Plan for Plant Specific Implementation of Extended Inservice Inspection Interval per WCAP-16168-NP, Revision 1, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval." PA-MSC-0120, July 12, 2010 (ADAMS Accession Number ML11153A033)
3. NRC Regulatory Guide 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," U.S. Nuclear Regulatory Commission, November 2002
4. Westinghouse Report WCAP-16168-NP-A, Revision 3, "Risk-Informed Extension of the Reactor Vessel In-service Inspection Interval," October 2011 (ADAMS Accession Number ML113060207)
5. NUREG-1874, "Recommended Screening Limits for Pressurized Thermal Shock (PTS)," U.S. Nuclear Regulatory Commission, March 2010
6. NRC Letter Report, "Generalization of Plant-Specific Pressurized Thermal Shock (PTS) Risk Results to Additional Plants," U.S. Nuclear Regulatory Commission, December 14, 2004 (ADAMS Accession Number ML042880482)
7. Code of Federal Regulations, 10 CFR Part 50.61a, "Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," U.S. Nuclear Regulatory Commission, Washington D. C., Federal Register, Volume 75, No. 1, dated January 4, 2010, and No. 22 with corrections to part (g) dated February 3, 2010, March 8, 2010, and November 26, 2010
8. Westinghouse Report WCAP-17315-NP, Revision 0, "Diablo Canyon Units 1 and 2 Pressurized Thermal Shock and Upper-Shelf Energy Evaluations," July 2011
9. NRC Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, May 1988