Pacific Gas and Electric Company®

James R. Becker
Site Vice President

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

805.545.3462
Internal: 691.3462
Fax: 805.545.6445

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PG&E Letter DCL-10-120

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

Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
Response to NRC Letter dated August 26, 2010, Request for Additional Information (Set 17) for the Diablo Canyon License Renewal Application

Dear Commissioners and Staff:

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

By letter dated August 26, 2010, the NRC staff requested additional information needed to continue their review of the DCPP LRA.

PG&E's response to the request for additional information is included in Enclosure 1. LRA Amendment 14 resulting from the responses is included in the Enclosure 2 showing the changed pages with line-in/line-out annotations.

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

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

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

Sincerely,

James R. Becker

A139
NRR

pns/50338724
Enclosure
cc: Diablo Distribution
cc/enc: Elmo E. Collins, NRC Region IV Regional Administrator
Nathanial Ferrer, NRC Project Manager, License Renewal
Kimberly J. Green, NRC Project Manager, License Renewal
Michael S. Peck, NRC Senior Resident Inspector
Fred Lyon, NRC Project Manager, Office of Nuclear Reactor Regulation

PG&E Response to NRC Letter dated August 26, 2010, Request for Additional Information (Set 17) for the Diablo Canyon License Renewal Application

RAI 4.7.2-1

In license renewal application (LRA) section 4.7.2, within the "Pressurizer" section, the applicant states that the fatigue crack growth analyses were projected to the end of the period of extended operation and are therefore valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

1. *Discuss how the actual plant transient cycles are monitored to ensure that they are bounded by the number assumed in the fatigue crack growth analysis.*

2. *Discuss the transient cycles used in the crack growth analyses, including the number of cycles.*

PG&E Response to RAI 4.7.2-1

1. The fatigue crack growth analyses associated with the Diablo Canyon Power Plant (DCPP) Unit 2 pressurizer structural weld overlays (SWOL) confirm that crack growth due to fatigue would remain within ASME Section XI, Appendix C, acceptable crack size criteria limits for 38 years after installation. The analyses are based on design basis numbers of transients. The SWOL were installed in 2008, therefore the analyses are valid through 2046, which encompasses the period of extended operation. Since the analyses are valid through the end of period of extended operation, the TLAA for the SWOL fatigue crack growth is dispositioned in accordance with 10CFR54.21(c)(1)(i).

   The actual plant transient cycles related to the SWOL 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 SWOL fatigue analysis limits. See revised LRA Table A4-1 in Enclosure 2.

2. Transients used in the fatigue crack growth analysis are shown below with the number of cycles analyzed.

Transient	Unit 2 Pressurizer Spray Nozzle	Unit 2 Pressurizer Safety & Relief Nozzle	Unit 2 Pressurizer Surge Nozzle
Heatups/Cooldowns	250	250	310*
Unit Loading/ Unloading at 5 percent/min	41,950	--	39,600
Red. Temp. Return to Power	--	--	4,470
Large Step Load Decrease w/Steam Dump	250	250	200
10 percent Step Load Increase/Decrease	2,500 each	2,500	4,000
Boron Equalization	32,000	--	32,000
Loss of Load	100	100	80
Loss of Power	50	50	40
Loss of Flow	100	100	220
Reactor Trip	500	500	400
Inadvertent Auxiliary Spray	12	12	24
Operation Basis Earthquake Load Cycles	400	400	400
Turbine Roll	10	10	20

* Combines heatup and cooldowns with 60 leak test transients.

Two transients used in the fatigue crack growth analysis have been deemed nonsignificant: (1) Reduced temperature return to power, and (2) Boron equalization per the Westinghouse system standard. These transients are associated with load following. The current operating strategy for the DCPP units is continuous base-load power generation. Therefore, the actual number of reduced temperature return to power and boron equalization occurrences is expected to be a small fraction of the cycles assumed in the fatigue analyses.

<u>*RAI 4.7.2-2*</u>

In LRA section 4.7.2, within the "Pressurizer" section, the applicant states that "[n]o base-metal corrosion analyses exist for the pressurizers, since no half-nozzle or similar repairs have exposed the base metal to reactor coolant." The applicant also states that "[t]he Unit 1 pressurizer and its nozzles and safe ends contain no Alloy 600 or Alloy 82/182 weld material." The above statements are not clear regarding whether the half nozzle method was used in repairing heater sleeves in the pressurizer in both units.

1. *For each unit, list all the pressurizer nozzles (e.g., pressurizer safety valve nozzle and heater sleeve nozzle). Identify the materials used to fabricate the nozzles. If a nozzle is welded to a safe end, identify the material of the safe end.*
2. *Discuss whether a fatigue crack growth calculation was performed for the remnant Alloy 82/182 welds. If so, discuss how the transient cycles used in the fatigue crack growth calculation are monitored to ensure they bound the actual plant cycles. If no fatigue crack growth calculation was performed, justify the structural integrity of the pressurizer shell.*
3. *Discuss any flaws that remained in service in the heater sleeves and in the attachment welds in both units. If so, discuss how these flaws are monitored and evaluated for the period of extended operation.*

<u>PG&E Response to RAI 4.7.2-2</u>

1. Pressurizer nozzles and materials of fabrication are provided below. If a nozzle is welded to a safe end, the material of the safe end is also identified.

Component	Unit 1	Unit 2
Surge Nozzle and Safe End Material	Nozzle – SA-216 WCC Safe End – SA-182 Type 316	Nozzle – SA-508 Cl. 2 Safe End – SA-182 Type 316L
Spray Nozzle and Safe End Material	Nozzle – SA-216 WCC Safe End – SA-182 Type 316	Nozzle – SA-508 Cl. 2 Safe End – SA-182 Type 316L
Safety and Relief Nozzle and Safe End Material	Nozzle – SA-216 WCC Safe End – SA-182 Type 316	Nozzle – SA-508 Cl. 2 Safe End – SA-182 Type 316L
Instrument Tube	SA-213 Type 316	SA-213 Type 316
Heater Well	SA-213 Type 316	SA-213 Type 316

2. Westinghouse performed an assessment of primary water stress corrosion cracking susceptibility for Alloy 82/182 welds in Diablo Canyon Power Plant (DCPP) Units 1 and 2 as discussed in License Renewal Application (LRA) Section 4.7.2. The only pressurizer repair or mitigation work that has been completed at DCPP is the Unit 2 pressurizer structural weld overlays as discussed in LRA Section 4.7.2 and in PG&E's response to Request for Additional Information 4.7.2-1.

3. No flaws have been identified in DCPP Units 1 or 2 pressurizers.

RAI 4.7.2-3

Discuss whether reactor vessel internals contain any nickel-based Alloy 600 components or nickel-based Alloy 82/182 welds. If so, discuss how these components are monitored for primary water stress corrosion cracking.

PG&E Response to RAI 4.7.2-3

The Diablo Canyon Power Plant reactor vessel internals do not contain any Alloy 600 components or nickel-based Alloy 82/182 welds.

RAI 4.7.2-4

In LRA section 4.7.2, within the "Steam Generators" section, the applicant states that "[r]eplacement steam generators contain no Alloy 600 components or Alloy 82/182 welds."

1. *Identify the material specification of the welds that join the replacement steam generator nozzles to the piping.*
2. *Identify the material specification of the safe ends that are welded to the steam generator nozzles.*

PG&E Response to RAI 4.7.2-4

The table below provides the requested materials associated with the Diablo Canyon Power Plant Units 1 and 2 replacement steam generators (RSGs).

RSG Nozzle	**Material specification of the welds that join the RSG nozzles to the piping**	**Material specification of the safe ends that are welded to the RSG nozzles**
Primary Nozzles	ER316L	SA-336 Class F316LN
Feedwater Nozzles	ER70S-6 & E7018	SFA-5.18 Class ER70S-X
Steam Nozzles	ER70S-6 & E7018	SA-508 Grade 1A

<u>RAI 4.7.5-1</u>

In LRA Section 4.7.5, within the "Unit 2 RHR Piping Weld RB-119-11" section, the applicant states that "[t]he DCPP licensing basis assumes 250 heatups and 250 cooldowns for a 50 year plant life."

1. *Discuss why only heatup and shutdown cycles are applied for flaw evaluation of weld RB-119-11 in the June 6, 2006 letter but other transient cycles such as seismic, temperature, and pressure were not mentioned in the flaw evaluation for weld RB-119-11.*

2. *It is not clear in LRA Section 4.7.5 or in the flaw evaluation that the cycles used in the flaw evaluation for weld RB-119-11 bounds the accumulated transient cycles at the end of 60 years. LRA section 4.7.5 states that "[t]he 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 above statements do not provide a clear reasoning as to how the flaw evaluation for 40 years encompasses 60 years of plant life. Clarify how the flaw evaluation encompassed a 60 year plant life in terms of cycle counting (e.g., are the 500 startup and shutdown cycles bound the actual plant cycles at the end of 60 years?).*

3. *Discuss how you ensure that transient cycles used in the flaw evaluation for the Unit 2 residual heat removal (RHR) piping weld RB-119-11 do not exceed the actual operating cycles at the end of 60 years without the enhanced fatigue management program.*

4. *(a) Provide the material specification of weld RB-119-11 (e.g., E308L or Alloy 82/182). (b) Discuss whether the indication in weld RB-119-11 is surface-connected or embedded. (c) Discuss the degradation mechanism of the indication. (d) If the weld is fabricated with Alloy 82/182 metal or if the flaw is embedded in the pipe/weld wall thickness, discuss any mitigation measures applied to the flaw in Weld RB-119-11.*

5. *Discuss whether weld RB-119-11 will be examined in the future ASME 10-year inservice inspection (ISI) intervals. If not, provide justifications.*

PG&E Response to RAI 4.7.5-1

1. Only heatup and shutdown cycles were discussed in the License Renewal Application (LRA) for the Diablo Canyon Power Plant (DCPP) Unit 2 residual heat removal (RHR) piping weld RB-119-11 flaw evaluation because the flaw evaluation only used heatup and shutdown cycles. Maximum stresses (not cycles) due to pressure, deadweight, seismic loadings, and thermal expansion were also used in the evaluation.

   The Unit 2 RHR piping weld RB-119-11 flaw evaluation was submitted to the NRC in PG&E Letter DCL-06-069, "Residual Heat Removal Weld RB-119-11 – Flaw Analytical Evaluation Results," dated June 6, 2006. The flaw evaluation was performed based on the guidelines of ASME Code, Section XI, IWB-3640, to calculate the allowable flaw size for the RHR pipe weld, specifically using the procedures and acceptance criteria of IWB-3641.

2. As shown in LRA Table 4.3-2, the number of heatup and cooldown cycles that DCPP projects for 60 years of operation (based on actual plant operating history) is 65 and 63 for Unit 2, respectively. This is less than the 500 heatup and cooldown cycles that were used in the Unit 2 RHR piping weld RB-119-11 flaw evaluation. Thus, the flaw evaluation cycles are bounded by projected actual plant cycles at the end of 60 years.

   Additionally, as shown in LRA Table 4.3-2, the transient cycles used in the flaw evaluation for the Unit 2 RHR piping weld RB-119-11 (plant heatup and cooldown cycles) are monitored by the Metal Fatigue of Reactor Coolant Pressure Boundary Program, as summarized in LRA Section B3.1. The Metal Fatigue of Reactor Coolant Pressure Boundary Program will ensure that transient cycles used in the flaw evaluation are not exceeded by the actual operating cycles.

3. Since the Unit 2 RHR piping weld RB-119-11 flaw evaluation states that it is valid through October 2046, the time limited aging analysis (TLAA) has been dispositioned in accordance with 10 CFR 54.21(c)(1)(i). Additionally, as described in part 2 of this response, it has been shown (based on actual plant operating history) that the flaw evaluation cycles are bounded by projected actual plant cycles at the end of 60 years.

   As required by the Metal Fatigue of Reactor Coolant Pressure Boundary Program (as summarized in LRA Section B3.1), if DCPP reaches one of the cycle count action limits, acceptable corrective actions are implemented.

4. (a) The weld filler material that is used for RB-119-11 is ER308.

   (b) The indication in weld RB-119-11 is embedded.

   (c) The flaw was characterized as a lack of fusion from original fabrication and was not service induced.

   (d) No mitigation measures were applied to the flaw in Weld RB-119-11.

5. As required by IWA-2420 of ASME Code, Section XI, one successive examination was completed for the Unit 2 RHR piping weld RB-119-11 flaw. The ultrasonic examination concluded that there were no apparent changes in the indication and that the results were satisfactory. As required by the ASME Code, Section XI, Unit 2 RHR piping weld RB-119-11 will be examined in the future ASME 10-year in service inspection intervals.

<u>*RAI 4.7.5-2*</u>

LRA section 4.7.5 discusses the flaw evaluation of an indication detected in weld WIC-95 of the RHR injection line 985 to hot legs 1 and 2 as shown in Pacific Gas and Electric Company (PG&E) Letter DCL-97-086 dated May 7, 1997. LRA Section 4.7.5 states further that "[t]here have been no occurrences of a DE, DDE, or Hosgri seismic event at Diablo Canyon Power Plant (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 are sufficient to the end of the period of extended operation."

1. *LRA section 4.7.5 states that "[t]he number of seismic cycles used in the analysis [flaw evaluation] is consistent with the DCPP 50-year design basis described in FSAR Table 5.2-4..." Final Safety Analysis Report (FSAR) Table 5.2-4 specifies one cycle for the Hosgri earthquake, 20 cycles for the design earthquake (DE), and 1 cycle for the double design earthquake (DDE). In the flaw evaluation for weld WIC-95 in the applicant's letter dated May 7, 1997, none of these seismic cycles were discussed. The applicant's flaw evaluation discussed only "400 cycles of future loading for the governing pipe stress load case". Clarify whether the seismic cycles were included in the flaw evaluation of the indication at weld WIC-95.*

2. *FSAR Table 5.2-4 provides several transients that have more occurrences/cycles than 400 cycles used in the flaw evaluation for weld WIC-95. For example, Unit loading and unloading at 5% of full power has 18,300 occurrences (cycles), hot standby operation/feedwater cycling has 18,300 occurrences. (a) Identify the transients that are included in the 400 cycles. (b) Provide basis for those transients shown in Table 5.2-4 but were not included in the flaw evaluation for weld WIC-95.*

3. *FSAR Table 5.2-4 specifies 250 occurrences for reactor coolant system heatup and cooldown transients. The total cycles for heatup and shutdown transients would be 500. However, the flaw evaluation used only 400 cycles. The staff notes that 500 cycles were used in the flaw evaluation of the indication in weld RB-119-11. The cycles in FSAR Table 5.2-4 are for the design life of the plant which presumably is 50 years. It appears that the 400 cycles used in the flaw evaluation for weld WIC-95 are for 50 years, not 60 years, of plant operation. LRA section 4.7.5 states that the seismic cycles in the weld WIC-95 fatigue crack growth evaluation for the 50-year design basis number of DE, DDE, and Hosgri events are sufficient to the end of the period of extended operation. Clarify whether (a) the seismic cycles in the flaw evaluation in the May 7, 1997 letter, are sufficient to cover the seismic cycles at the end of extended operation, (b) the 400 cycles cover all the transient cycles at the end of extended operation, and (c) why a total of 500 cycles for heatup and cooldown were not used.*

4. *(a) Provide the pipe diameter and wall thickness at weld WIC-95 of the Unit 1 RHR injection line 985 where an indication was detected in refueling outage 9. (b) In the flaw evaluation dated May 7, 1997, the applicant stated that it will re-examine the indication in weld WIC-95 in refueling outage 1R10. Discuss the inspection result of weld WIC-95 during refueling outage 1R10. Confirm that the indication was detected in 1997 and was re-examined in 1999. (c) Provide the material specification of weld WIC-95 (e.g., Alloy 82/182 weld or E308L). (d) Discuss whether the subject indication is surface-connected or embedded. (e) Discuss the degradation mechanism of the indication. (f) Discuss the orientation of the indication (i.e., a circumferential or an axial indication). (g) Provide operating temperature and pressure of the subject pipe line at weld WIC-95.*

5. *Discuss whether weld WIC-95 will be examined in the future ASME 10-year ISI inspection intervals. If not, provide justifications.*

6. *It is not clear to the staff that the applicant has demonstrated that the cycles used in the flaw evaluation for weld WIC-95 bounds the cycles at the end of 60 years. Discuss how you ensure that transient cycles used in the flaw evaluation for the RHR piping weld WIC-95 do not exceed the actual operating cycles.*

<u>PG&E Response to RAI 4.7.5-2</u>

1. Cycles for the design earthquake were included in the Unit 1 residual heat removal (RHR) Weld WIC-95 flaw evaluation. As stated in PG&E Letter DCL-97-086, "Inservice Inspection Evaluation Analysis of Flaw Indication for Weld WIC-95 (Reference A0430829)," dated May 7, 1997, "400 cycles of future loading for the governing pipe stress load case" were assumed. The flaw evaluation further clarifies that these "400 cycles of future loading" are seismic cycles. This is consistent with Final Safety Analysis Report (FSAR) Table 5.2-4, which states that the 50-year design basis for design earthquakes is 20 events, with 20 cycles per event (a total of 400 cycles).

2. (a) As stated in part 1 of this response, the "400 cycles of future loading" are seismic cycles.

   (b) The Unit 1 RHR Weld WIC-95 flaw evaluation was performed based on the guidelines of ASME Code, Section XI, IWB-3640, to calculate the allowable flaw size for the RHR weld.

   RHR injection line 985 to hot legs 1 and 2 only operates during plant refueling (i.e., during heatups and cooldowns). When not in a plant refueling mode, the RHR injection line is not in service. Thus, those additional transients listed in FSAR Table 5.2-4 have no significant impact on the line and do not contribute any thermal cycles. The Unit 1 RHR Weld WIC-95 flaw evaluation states that the seismic events, plus pressure and deadload, envelops the thermal stress both in magnitude and number of cycles. Additionally, thermal and seismic stresses are not combined per ANSI B31.1 code.

3. (a) The 400 seismic cycles used in the flaw evaluation are adequate for the period of extended operation because no seismic cycles have occurred at Diablo Canyon Power Plant (DCPP) since operation began. As with all other transients, seismic cycles are projected to 60 years of operation by using the actual plant operating history and projecting it to 60 years. As shown in License Renewal Application (LRA), Table 4.3-2, the projected number of design earthquakes (and thus the number of seismic cycles) is less than the 400 cycles used in the flaw evaluation.

   (b) As stated in part 2(a) of this response, the 400 cycles used in the flaw evaluation are seismic cycles. The flaw evaluation did not address other transient cycles because, as stated in Request for Additional Information Response 4.7.5-2, part 2(b), those additional transients listed in FSAR Table 5.2-4 have no significant impact on the line and do not contribute any thermal cycles.

   (c) Heatup and cooldown cycles were not included in the Unit 1 RHR Weld WIC-95 flaw evaluation. Rather, the flaw evaluation used 400 future loading cycles because seismic events, plus pressure and deadload, enveloped the thermal stress (which would be associated with heatups and cooldowns) both in magnitude and number of cycles.

4. (a) The pipe diameter and wall thickness at weld WIC-95 of the Unit 1 RHR injection line 985 where an indication was detected was 12.750 inches outside diameter and 0.410 inches, respectively.

   (b) As required by IWA-2420 of ASME Code, Section XI, one successive examination was completed for the Unit 1 RHR Weld WIC-95 flaw in October 1999. The ultrasonic examination concluded that there were no apparent changes in the indication and that the results were satisfactory.

   (c) The material specification of Weld WIC-95 is ER308.

   (d) As stated in PG&E Letter DCL-97-086, the subject indication is inside diameter connected.

   (e) The flaw was characterized as construction-related flaw and was not service induced.

   (f) The orientation of the indication is circumferential.

(g) The maximum operating temperature and pressure of the subject line at weld WIC-95 are 350°F and 700 psig, respectively.

5. As required by the ASME Code, Section XI, Weld WIC-95 will be examined in the future ASME 10-year in service inspection intervals.

6. Since the Unit 1 RHR Weld WIC-95 flaw evaluation shows that the flaw is valid after 400 seismic cycles, the time limited aging analysis has been dispositioned in accordance with 10 CFR 54.21(c)(1)(i). Additionally, as described in LRA Section 4.7.5, it has been shown (based on actual plant operating history) that the flaw evaluation seismic cycles are bounded by projected actual plant cycles at the end of 60 years.

   As required by the Metal Fatigue of Reactor Coolant Pressure Boundary Program, (as summarized in LRA Section B3.1), if DCPP reaches one of the cycle count action limits (such as for seismic cycles), acceptable corrective actions are implemented.

<u>*RAI 4.7.5-3*</u>

LRA Section 4.7.5 discussed the indication detected in Unit 2 Auxiliary feedwater piping line 567. The applicant submitted a flaw evaluation in PG&E letter DCL-99-136, dated October 22, 1999.

1. *In the flaw evaluation for piping line 567, the applicant stated that it will re-examine the indication during the Unit 2 tenth refueling outage (2R10). Discuss the inspection results of the re-examination.*

2. *The applicant stated in the flaw evaluation that the indication is believed to be a fabrication defect (a lap in the pipe). Confirm that the indication is embedded in the pipe wall. As stated in the flaw evaluation, the flaw was characterized as 0.1 inch deep (approximately 46 percent through wall) and 12 feet in length. Describe in detail how the indication is modeled in the flaw growth calculation.*

3. *The flaw evaluation dated October 22, 1999 states that the 250 cycles of future seismic and thermal loading corresponding to the remaining plant life. In LRA Section 4.7.5, the applicant stated that the assumed transients are consistent with or bounded by the 50 year design basis described in FSAR Table 5.2-4. It is not clear to the staff that 250 cycles used in the flaw evaluation bound the cycles in Table 5.2-4 in FSAR. Identify the transients that are included in the 250 cycles. Discuss in detail how 250 cycles in the flaw evaluation bound the cycles in the licensing basis.*

4. *Discuss whether the indication in Unit 2 Auxiliary feedwater piping line 567 will be examined in the future ASME 10-year ISI inspection intervals. If not, provide justification.*

<u>PG&E Response to RAI 4.7.5-3</u>

1. One successive examination was completed for the Unit 2 auxiliary feedwater piping line 567. The ultrasonic examination concluded that there were no apparent changes in the indication and that the results were satisfactory.

2. The indication in the Unit 2 auxiliary feedwater piping line 567 was surface-connected, not embedded.

   Because the piping material is carbon steel with stresses in the elastic range, the associated flaw evaluation used linear elastic fracture mechanics to evaluate the flaw growth. This approach is conservative since the carbon steel material has some ductility. The methodology is similar to ASME Section XI Appendix A, except that the Appendix A crack growth relations are based on a flat plate, while the analysis is performed for cylindrical geometry and is thus more accurate for a pipe.

The flaw model used was a longitudinal crack in a cylinder with t/R=0.2 (i.e., the ratio of pipe thickness to pipe mean radius). All of the stresses were conservatively applied as membrane stresses. Using the crack growth law for ferritic steel in an air environment and the material fracture toughness of carbon steel, the crack growth was determined for the given number of cycles. It was determined that the growth in the flaw was below the critical flaw size.

3. The Unit 2 auxiliary feedwater line 567 flaw evaluation considered 250 Hosgri seismic loads (5 seismic events with 50 cycles per event). This is more conservative than the licensing basis described in Final Safety Analysis Report, Table 5.2-4, because it is based on 5 Hosgri events while the licensing basis only anticipates 1 event.

4. Diablo Canyon Power Plant (DCPP) evaluated the Class 3 Unit 2 auxiliary feedwater line 567 flaw to Class 1 requirements since the 1989 ASME Code then in effect did not have Class 3 acceptance criteria. Although there is no applicable requirement, DCPP committed to perform one successive exam, which yielded satisfactory results. There are no plans to conduct any further inspections on the Unit 2 auxiliary feedwater line 567 because; (1) it is not required, (2) the flaw is a fabrication defect and is not service-related, and (3) a follow-up examination showed there was no change in the flaw.

<u>RAI B2.1.39-1</u>

In LRA Section B2.1.39; the applicant states that the Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program will be implemented as part of the ASME Code, Section XI ISI program and will be completed within the 10-year inspection interval before the period of extended operation.

1. *The NRC staff notes that ultrasonic testing (UT) has not yet been qualified to examine CASS material via the ASME Code, Section XI, Appendix VIII. Discuss how components fabricated with CASS material are inspected under the current licensing basis. Discuss whether the current inspection practices (methods, frequencies and acceptance criteria) will be applied in the future CASS aging management program (AMP).*

2. *In light of the limitation of UT of CASS material, discuss how volumetric examination of CASS components will be accomplished during the period of extended operation. Specifically, clarify whether the qualified UT will only be used in the CASS AMP, if a qualified UT method becomes available.*

<u>PG&E Response to RAI B2.1.39-1</u>

1. Components fabricated with cast austenitic stainless steel (CASS) material that are in scope of Aging Management Program (AMP) B2.1.39, reactor coolant loop elbow fittings, are currently pressure tested every refueling outage per the current ASME Section XI Code edition in effect. Current inspection practices will continue during the period of extended operation as required per the ASME Code editions in effect during the period of extended operation. In addition, as indicated in License Renewal Application (LRA) Section B2.1.39, for in-scope CASS components that are determined to be susceptible to the aging effect of thermal embrittlement, aging management would be accomplished through a qualified volumetric examination, provided one is demonstrated to be adequate for CASS inspection in accordance with criteria identified in ASME Section XI, Appendix VIII, or a component-specific flaw tolerance evaluation will be performed. Additional inspection or evaluations to demonstrate that the material has adequate fracture toughness will not be required for components that have been determined to not be susceptible to thermal aging embrittlement.

2. For the CASS AMP, DCPP will either; (1) use a qualified ultrasonic testing (UT) method for enhanced volumetric examination, if one becomes available, or (2) perform a component-specific flaw tolerance evaluation. As indicated in LRA Section B2.1.39, this AMP is a new program and if a viable volumetric examination method is developed, it will be implemented as part of the Section XI In Service Inspection Program. The qualified UT method will be demonstrated to be adequate for CASS inspection in accordance with criteria specified in ASME Section XI, Appendix VIII.

RAI B2.1.39-2

(1) Discuss whether Diablo Canyon units 1 and 2 have implemented the risk-informed ISI program. (2) If yes, discuss how the CASS components will be inspected under the risk-informed ISI program considering the requirements of the CASS aging management program (e.g., whether the CASS AMP will increase the inspection frequency of the CASS components in the risk-informed ISI program and whether thermal aging embrittlement will be a degradation mechanism considered in the risk-informed ISI program).

PG&E Response to RAI B2.1.39-2

1. For the current 10-year in service inspection (ISI) interval, Diablo Canyon Power Plant, Units 1 and 2, have implemented the risk-informed ISI Program for piping welds.

2. Current inspection practices (pressure tests) will continue during the period of extended operation as required per the ASME Code editions in effect during the period of extended operation. In addition, regardless of whether the ISI Program for the period of extended operation is risk informed, as indicated in License Renewal Application, Section B2.1.39, for in-scope cast austenitic stainless steel components that are determined to be susceptible to the aging effect of thermal embrittlement, aging management will be accomplished through a qualified volumetric examination, if one becomes available, once every 10 years. Alternatively, a component-specific flaw tolerance evaluation will be performed.

LRA Amendment 14

LRA Section	RAI
Table A4-1	RAI 4.7.2-1

Table A4-1 License Renewal Commitments

Item #	Commitment	LRA Section	Implementation Schedule
38	The actual plant transient cycles related to the SWOL 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 SWOL fatigue analysis limits.	4.3	Prior to January 31,2011