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

August 17, 2010

PG&E Letter DCL-10-100

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 July 20, 2010, Request for Additional Information (Set 11) 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 July 20, 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.

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

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

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

Executed on August 17, 2010.

Sincerely,

James R. Becker



A139
NRR

pns/50329540
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
Alan B. Wang, NRC Project Manager, Office of Nuclear Reactor Regulation

# PG&E Response to NRC Letter dated July 20, 2010, Request for Additional Information (Set 11) for the Diablo Canyon License Renewal Application

*<u>RAI 3.1.2.2.3-1</u>*

*In license renewal application (LRA) Table 3.1.2-1, for component type "RV (Reactor Vessel) Nozzles (Inlet/Outlet Nozzles)," material "Carbon Steel with Stainless Steel Cladding," environment "reactor coolant (int)," and aging effect "loss of fracture toughness," there are two line items. For the first line item, aligned with GALL Report item IV.A2.16, the aging management program (AMP) is "Time-Limited Aging Analysis (TLAA) evaluated for the period of extended operation." For the second line item, aligned with GALL Report item IV.A2.17, the AMP is "Reactor Vessel Surveillance (82.1.15)." The first line item aligns with LRA Table 3.1.1, item 3.1.1.17, which states, in the AMP column:*

> *TLAA, evaluated in accordance with Appendix G of 10 CFR Part 50 and RG 1.99. The applicant may choose to demonstrate that the materials of the nozzles are not controlling for the TLAA evaluations.*

*The second line item aligns with LRA Table 3.1.1, item 3.1.1.18, which states, under the AMP column, "Reactor Vessel Surveillance Program (82.1.15)." Further evaluation is recommended for both Table 3.1.1 items.*

*However, the neutron embrittlement TLAA evaluations, described in LRA Section 4.2, do not discuss how the nozzles were demonstrated not to be controlling, nor are any nozzle materials included among the extended beltline materials that are included in the pressurized thermal shock (PTS) TLAA evaluation (LRA Section 4.2.2) and the Upper Shelf Energy (USE) evaluation (LRA Section 4.2.3). Further, the RV Surveillance Program description does not address any nozzle materials.*

1. *Describe how the Diablo Canyon Power Plant (DCPP) RV nozzle materials were demonstrated to not be controlling with respect to the neutron embrittlement related TLAAs. This description should include a summary of any fluence evaluations and/or neutron embrittlement projections using the nozzle fluence and chemistry to project the PTS reference temperature ($RT_{PTS}$) and USE of the nozzle materials.*

2. *Indicate any changes this may have to the LRA.*

3. *Describe how the RV Surveillance Program manage loss of fracture toughness of the RV nozzles if specimens of the nozzle materials are not included in the surveillance program.*

PG&E Response to RAI 3.1.2.2.3-1

PG&E evaluated reactor vessel materials, such as the nozzles, to determine if these materials would be exposed to fluences greater than $1 \times 10^{17}$ n/cm$^2$ (E>1.0 Mev) as a result of license renewal. This evaluation was performed by Westinghouse as part of the analysis, which formed the basis for the fluences provided in License Renewal Application Tables 4.2-1, 4.2-2, and 4.2-3. The Westinghouse letter, "Neutron Fluence Evaluation for the Diablo Canyon Units 1 and 2 Reactor Pressure Vessel Extended Beltline Materials" (LTR-REA-09-90), dated May 21, 2009, analysis showed that the nozzles, as well as the nozzle-to-nozzle-shell welds, remain below $1 \times 10^{17}$ n/cm$^2$ through 60 effective full-power years for Diablo Canyon Units 1 and 2. Consequently, nozzle materials are not included in the surveillance program, and changes to the license renewal application are not necessary.

*RAI 3.1.2.2.7-1*

*LRA Section 3.1.2.2.7 and LRA Table 3.1.2 indicate that material of the RV flange O-ring leak monitoring line is nickel alloy. However, in most pressurized-water reactors (PWRs) only the vessel penetration is nickel alloy while the adjoining piping is stainless steel.*

*Clarify whether the adjoining piping of the RV flange O-ring leak monitoring line is stainless steel. If so, clarify whether this piping is included within the scope of license renewal such as under another Table 2 line item such as Class 1 Piping <= 4in, GALL Report item IV.C2-1 in LRA Table 3.1.2-2.*

PG&E Response to RAI 3.1.2.2.7-1

The adjoining piping of the reactor vessel flange O-ring leak monitoring line is stainless steel. It is included within the scope of license renewal with the component type Class 1 piping less than or equal to four inches in License Renewal Application Table 3.1.2-2.

*RAI 3.1.2.2.7-2*

*In LRA Section 3.1.2.2.7.1, the applicant indicated that for managing aging due to stress corrosion cracking of stainless steel high pressure conduits (flux thimble guide-tubes-to seal table) exposed to reactor coolant, the applicant's Water Chemistry (B2.1.2) AMP will be augmented by their American Society of Mechanical Engineers (ASME) Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1) AMP. For stainless steel flux thimble tubes exposed to reactor coolant, cracking due to stress corrosion cracking (SCC) is managed by the DCPP Water Chemistry (B2.1.2) AMP. The staff notes that in LRA Table 3.1.2, the flux thimble tubes are included as a subcomponent of the RV Bottom Mounted Instrument Guide Tube, which aligns to GALL Report item IV.A2-1(RP-13) for the aging effect cracking.*

*The staff reviewed LRA Section 3.1.2.2.7.1 against the criteria in NUREG-1800, 'The Standard Review Plan for Review of License Renewal Applications (SRP-LR)," Section 3.1.2.2.7.1, which states cracking due to SCC could occur in the PWR stainless steel reactor vessel flange leak detection lines and bottom-mounted instrument guide tubes. The GALL Report recommends further evaluation to ensure that these aging effects are adequately managed. The SRP-LR further states that the GALL Report recommends that a plant-specific AMP be evaluated because existing programs may not be capable of mitigating or detecting cracking due to SCC. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of the SPR-LR).*

*Branch Technical Position RLS8-1 states that a plant-specific AMP should include a "detection of aging effects" program element. The DCPP Water Chemistry Program provides mitigation of cracking through control of impurities, but does not provide for detection of aging effects. The ASME Section XI Inservice Inspection Program, Subsections IWB, IWC, and IWD AMP, provides for inspections of components. The standard examination requirements for flux thimble tubes under the ASME Section XI Inservice Inspection Program, Subsections IWB, IWC, and IWD AMP, is a VT-2 visual inspection per ASME Code Section XI, Table IWB-2500-1, Examination Category B-P, which would not generally be capable of detecting cracking unless a leak is already present, producing visible water and/or boric acid. The program description in LRA Section B.2.1.1 does not describe any augmented inspections for the flux thimble tubes which would be capable of early detection of cracking.*

1. *Identify any specific examinations included in the ASME Section XI Inservice Inspection Program, Subsections IWB, IWC, and IWD AMP, which would be capable of detecting cracking in the RV Bottom Mounted Instrument Guide Tubes (High Pressure Conduits, Seals) before a throughwall crack and leakage occurs.*

2. *If the ASME Section XI Inservice Inspection Program, Subsections IWB, IWC, and IWD AMP does not provide for detection of cracking prior to a leak, provide a plant-specific AMP or combination of existing AMPs that include a "detection of aging effect" program element for managing the aging effect of cracking due to SCC*

*in the RV Bottom Mounted Instrument Guide Tubes (High Pressure Conduits, Seals); and*

*3. Describe what examination techniques will be used to detect (or confirm the absence of) the aging effect of cracking in the RV Bottom Mounted Instrument Guide Tubes (High Pressure Conduits, Seals), either as part of the ASME Section XI Inservice Inspection Program, Subsections IWB, IWC, and IWD, or an additional plant-specific program.*

PG&E Response to RAI 3.1.2.2.7-2

The Diablo Canyon Power Plant (DCPP) Water Chemistry Program (LRA Section B2.1.2) provides mitigation of stress corrosion cracking through control of impurities. In addition, DCPP's ASME XI Inservice Inspection (ISI) Program (LRA Section B2.1.1) examines the reactor vessel (RV) bottom-mounted instrument (BMI) guide tubes on a refueling frequency.

Industry operating experience has provided two events of RV BMI guide tube leakage. In a 1989 event at Turkey Point, transgranular stress corrosion cracking in several BMI guide tubes was found. The cracking occurred above the seal table due to chloride contamination and intermittent wetting. In 2003, boron deposits were found on the South Texas Project's Unit 1 BMI. The cause of this leak was determined to be a result of weld defects on the highly stressed interface between the Alloy 600 tube and the connecting weld to the pressure vessel. In both events the initiators of stress corrosion cracking (SCC) were due to nonaging related factors.

Westinghouse Technical Bulletin NSID-TB-90-02 recommended that the outside surface of the guide tubes at the seal table would be visually examined each outage. The RV BMI guide tubes are examined under ASME Section XI IWB-2500 item B15.10. DCPP inspection procedures contain acceptance criteria based on the ASME Code. Due to ASME Section XI Code inspection requirement size limitations, the RV BMI guide tubes are not specified to have any inspections other than normal operating pressure tests in ISI programs. There are no nondestructive examinations required, as piping of nominal pipe size of one inch and smaller is exempted from volumetric and surface exams by IWB-1220.

PG&E believes the current ISI testing by pressurizing the BMI guide tubes and visual examination is sufficient. PG&E's Water Chemistry Program also maintains the environment to prevent a similar event as Turkey Point. The industry operating experience shows that SCC of the BMI guide tubes was due to nonaging related factors.

<u>RAI 4.2.2-1</u>

*As part of its independent evaluation of the $RT_{PTS}$ values, the staff checked the copper, nickel, and initial (unirradiated) reference temperature ($RT_{NDT}$) values against the corresponding values from the Reactor Vessel Integrity Database (RVID) for each beltline material. The applicant provided lower copper values for DCPP, Unit 1 intermediate shell plates B4106-1 (Heat No. C2884-1), B4106-2 (Heat No. C2854-2), and 84106-3 (Heat No. C2793-1), and lower nickel values for plates 84106-3 and B4107-3 (Heat No. C3131-1), than the corresponding copper and nickel values in RVID. The copper and nickel values provided in the LRA for plate B4106-3 match the best estimate values provided in the most recent surveillance capsule report for DCPP, Unit 1, and are therefore acceptable.*

*Provide data for the copper and nickel content for DCPP, Unit 1 Intermediate Shell Plates B4106-1, B4106-2, and B4107-3 as given in LRA Tables 4.2-4 and 4.2-6.*

<u>PG&E Response to RAI 4.2.2-1</u>

The copper and nickel values provided in License Renewal Application Tables 4.2-4 and 4.2-6 are from the Diablo Canyon Power Plant Final Safety Analysis Report, Table 5.2-17A. They are taken from Westinghouse WCAP-13771, "Evaluation of Pressurized Thermal Shock for Diablo Canyon Unit 1," J.M. Chicots, July 1993. The copper and nickel values presented in this WCAP for the materials in question were averaged from material test certifications for the original fabrication.

RAI 4.2.2-2

*When using 10 CFR 50.61 to calculate $RT_{PTS}$, the margin term is defined as*

*Margin* $=2\sqrt{(\sigma_u^2+\sigma_\Delta^2)}$

*10 CFR 50.61, defines au as the standard deviation for the initial $RT_{NDT}$. If a measured value of initial $RT_{NDT}$ for the material in question is available, $\sigma_I$ is to be estimated from the precision of the test method. If not, and generic mean values for that class of material are used, $\sigma_I$ is the standard deviation obtained from the set of data used to establish the mean.*

*The staff performed preliminary confirmatory calculations of $RT_{PTS}$ using the $\sigma_\Delta$ term as defined in 10 CFR 50.61. To obtain the margin term for some of the materials listed in LRA Tables 4.24 and 4.2-5, it appears that a value of 17°F was used for $\sigma_I$, while a $\sigma_I$ value of 0°F was used for other materials. In some cases, the values used for a, appear to be inconsistent with values in RVID.*

*For the materials listed in LRA Tables 4.2-4 and 4.2-5:*

1. *Identify whether the initial (unirradiated) $RT_{NDT(u)}$ value cited for the material is based on plant-specific data or on a generic value.*

2. *For those materials using a generic initial $RT_{NDT(u)}$ value, describe the method of determination and basis for the initial (unirradiated) $RT_{NDT(u)}$ values for the materials listed in LRA Tables 4.2-4 and 4.2-5, and provide a reference for each value.*

3. *For those materials in LRA Tables 4.2-4 and 4.2-5 using a generic value for initial $RT_{NDT}$, provide the $\sigma_I$, value used and the basis for the $\sigma_I$ value.*

4. *Confirm that the appropriate value of $\sigma_I$, (0°F vs. 17°F) has been used for all materials listed in LRA Table 4.2-4 and 4.2-5.*

PG&E Response to RAI 4.2.2-2

**Item 1** - Given that $\sigma_u$ is the standard deviation for $RT_{NDT(U)}$, the following values were used:

$\sigma_u = 0^\circ F$ when $RT_{NDT(U)}$ is a measured value
$\sigma_u = 17^\circ F$ when $RT_{NDT(U)}$ is a generic value

The following two tables demonstrate how the two values were used. The use of these two standard deviation values is standard practice in the industry. For example, see WCAP-15571 Supplement 1, "Analysis of Capsule Y from First Energy Company Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program," Revision 1, April 2008.

**Items 2-4** - $RT_{NDT(U)}$ is determined from drop weight and transverse Charpy tests. The following tables indicate what was used for the Unit 1 and 2 materials and how the $RT_{NDT(U)}$ values were determined. Combustion engineering provided drop weight test results for all the shell plates, and Westinghouse performed transverse Charpy tests for the intermediate and lower shell plates. Also, combustion engineering performed drop weight and Charpy tests on weldments made from Heats 27204/12008 for Unit 1 and Heats 21935/12008 for Unit 2. These are the only two weldments for which drop weight data are available.

When transverse Charpy data were not available for shell plates from testing, an estimate was made from the combustion engineering longitudinal Charpy data according to the method of Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements", Section 1.1(3)(b). This method yields an estimate for the 50 ft-lb and 35 mil lateral expansion temperature for the transverse case (i.e., $T_{CV}$), determined by adding 20°F to the measured 50 ft-lb and 35 mil lateral expansion temperature for the longitudinal case. Then per ASME 1998 Code Section III, Division 1 NB-2331 (a)(3), the $RT_{NDT(U)}$ is the higher of the nil-ductility transition temperature, determined by drop weight tests, and ($T_{CV}$ – 60°F). In all cases, ($T_{CV}$ - 60°F) was higher than the nil-ductility transition temperature.

Per 10 CFR 50.61 (c)(1)(ii), when a measured value for $RT_{NDT(U)}$ is not available, then a generic value of -56°F is used for welds made with Linde 0091, 1092, and 124 and ARCOS B-5 weld fluxes.

The following two tables confirm that the appropriate value of $\sigma_u$ (also known as $\sigma_i$) was used for all materials listed in License Renewal Application Tables 4.2-4 and 4.2-5.

DCPP Unit 1

| Location | $RT_{NDT(U)}$ | Meas/Gen | Source |
|---|---|---|---|
| Upper Shell Plate B4105-1 | 28 | Generic, $\sigma_u = 17^\circ F$ | Estimated per Branch Technical Position MTEB 5-2 1.1(3)(b) using Combustion Engineering longitudinal Charpy data |
| Upper Shell Plate B4105-2 | 9 | Generic, $\sigma_u = 17^\circ F$ | Estimated per Branch Technical Position MTEB 5-2 1.1(3)(b) using Combustion Engineering longitudinal Charpy data |
| Upper Shell Plate B4105-3 | 14 | Generic, $\sigma_u = 17^\circ F$ | Estimated per Branch Technical Position MTEB 5-2 1.1(3)(b) using Combustion Engineering longitudinal Charpy data |
| Intermediate Shell Plate B4106-1 | -10 | Measured, $\sigma_u = 0^\circ F$ | Combustion Engineering drop weight tests and Westinghouse transverse Charpy tests |
| Intermediate Shell Plate B4106-2 | -3 | Measured, $\sigma_u = 0^\circ F$ | Combustion Engineering drop weight tests and Westinghouse transverse Charpy tests |
| Intermediate Shell Plate B4106-3 | 30 | Generic, $\sigma_u = 17^\circ F$ | Estimated per Branch Technical Position MTEB 5-2 1.1(3)(b) using Combustion Engineering longitudinal Charpy data |
| Lower Shell Plate B4107-1 | 15 | Measured, $\sigma_u = 0^\circ F$ | Combustion Engineering drop weight tests and Westinghouse transverse Charpy tests |
| Lower Shell Plate B4107-2 | 20 | Measured, $\sigma_u = 0^\circ F$ | Combustion Engineering drop weight tests and Westinghouse transverse Charpy tests |
| Lower Shell Plate B4107-3 | -22 | Measured, $\sigma_u = 0^\circ F$ | Combustion Engineering drop weight tests and Westinghouse transverse Charpy tests |
| Upper Shell Long. Welds 1-442 A, B, C | -20 | Measured, $\sigma_u = 0^\circ F$ | Combustion Engineering drop weight and Charpy tests |
| Upper Shell to Interm. Shell Circumferential Weld 8-442 | -56 | Generic, $\sigma_u = 17^\circ F$ | Per 10 CFR 50.61 (c)(1)(ii) |
| Intermediate Shell Long. Welds 2-442 A, B, C | -56 | Generic, $\sigma_u = 17^\circ F$ | Per 10 CFR 50.61 (c)(1)(ii) |
| Lower Shell Long. Welds 3-442A, B, C | -56 | Generic, $\sigma_u = 17^\circ F$ | Per 10 CFR 50.61 (c)(1)(ii) |
| Intermediate to Lower Shell Circumferential Weld 9-442 | -56 | Generic, $\sigma_u = 17^\circ F$ | Per 10 CFR 50.61 (c)(1)(ii) |

DCPP Unit 2

| Location | $RT_{NDT(U)}$ | Meas/Gen | Source |
|---|---|---|---|
| Upper Shell Plate B5453-1 | 28 | Measured, $\sigma_u = 0^\circ F$ | Combustion Engineering drop weight tests and Westinghouse transverse Charpy tests |
| Upper Shell Plate B5453-3 | 5 | Generic, $\sigma_u = 17^\circ F$ | Estimated per Branch Technical Position MTEB 5-2 1.1(3)(b) using Combustion Engineering longitudinal Charpy data |
| Upper Shell Plate B5011-1R | 0 | Generic, $\sigma_u = 17^\circ F$ | Estimated per Branch Technical Position MTEB 5-2 1.1(3)(b) using Combustion Engineering longitudinal Charpy data |
| Intermediate Shell Plate B5454-1 | 52 | Measured, $\sigma_u = 0^\circ F$ | Combustion Engineering drop weight tests and Westinghouse transverse Charpy tests |
| Intermediate Shell Plate B5454-2 | 67 | Measured, $\sigma_u = 0^\circ F$ | Combustion Engineering drop weight tests and Westinghouse transverse Charpy tests |
| Intermediate Shell Plate B5454-3 | 33 | Measured, $\sigma_u = 0^\circ F$ | Combustion Engineering drop weight tests and Westinghouse transverse Charpy tests |
| Lower Shell Plate B5455-1 | -15 | Measured, $\sigma_u = 0^\circ F$ | Combustion Engineering drop weight tests and Westinghouse transverse Charpy tests |
| Lower Shell Plate B5455-2 | 0 | Measured, $\sigma_u = 0^\circ F$ | Combustion Engineering drop weight tests and Westinghouse transverse Charpy tests |
| Lower Shell Plate B5455-3 | 15 | Measured, $\sigma_u = 0^\circ F$ | Combustion Engineering drop weight tests and Westinghouse transverse Charpy tests |
| Upper Shell Long. Welds 1-201 A, B, C | -50 | Measured, $\sigma_u = 0^\circ F$ | Combustion Engineering drop weight and Charpy tests |
| Upper Shell to Interm. Shell Circumferential Weld 8-201 | -56 | Generic, $\sigma_u = 17^\circ F$ | Per 10 CFR 50.61 (c)(1)(ii) |
| Intermediate Shell Long. Welds 2-201 A, B, C | -50 | Measured, $\sigma_u = 0^\circ F$ | Combustion Engineering drop weight and Charpy tests |
| Lower Shell Long. Welds 3-201A, B, C | -56 | Generic, $\sigma_u = 17^\circ F$ | Per 10 CFR 50.61 (c)(1)(ii) |
| Intermediate to Lower Shell Circumferential Weld 9-201 | -56 | Generic, $\sigma_u = 17^\circ F$ | Per 10 CFR 50.61 (c)(1)(ii) |

<u>RAI 4.2.2-3</u>

*For DCPP, Unit 2, the applicant stated that in accordance with Regulatory Guide 1.99 Revision 2, the Charpy V-Notch Upper Shelf Energy ($C_v$ USE) data from Unit 2 Surveillance Capsule V were deemed credible for intermediate shell plate B5454-1 (Heat No. C5161-1). The applicant stated that the $C_v$ USE values were projected to 54 effective full power years (EFPY) of operation using Regulatory Guide 1.99 Position 1.2. If credible surveillance data are available, Regulatory Guide 1.99 Revision 2 Position 2.2 recommends that the decrease in upper-shelf energy may be obtained by plotting the reduced plant surveillance data on Figure 2 of the regulatory guide and fitting the data with a line drawn parallel to the existing lines as the upper bound of all the data, and that this line should be used in preference to the existing graph. The staff notes that the most recent surveillance capsule report for DCPP, Unit 2 (Reference 1), indicated that the surveillance data for Plate B5454-1 were credible for both the $\Delta RT_{NDT}$ and USE, and that the surveillance data were used in the end of license extension (EOLE) projection of $RT_{PTS}$ for this material. The EOLE $C_v$ USE values for the Unit 2 beltline and extended beltline materials are provided in LRA Table 4.2-7.*

*Although surveillance data for intermediate shell plate B5454-1 are available, it was not clear whether the surveillance $C_v$ USE data were used in the projection for that plate.*

*Clarify if surveillance program data used in the projection of USE for DCPP Unit 2 intermediate shell plate 85454-1 (Heat No. C5161-1). If surveillance data was not used, provide justification.*

<u>PG&E Response to RAI 4.2.2-3</u>

For the entire Unit 2 surveillance capsule materials (i.e., intermediate shell plate B5454-1 and weld metal 12008/21935), the measured decrease in upper shelf energy (USE) was taken into account when making USE projections. Figure 2 of Regulatory Guide 1.99, Revision 2, was used to determine Position 1.2 predicted USE decreases for the surveillance materials, which were compared to the measured USE decreases. For intermediate shell plate B5454-1, the measured USE decreases were smaller (less limiting) than the Position 1.2 predictions, and therefore Position 1.2 was used instead of Position 2.2. For the surveillance weld metal (corresponding to intermediate shell axial welds 2-201A, B, and C), the Position 1.2 predictions were less limiting, and therefore Position 2.2 was used. The different values are presented in the following three tables.

Position 1.2 Predicted USE Decreases (%)

| Material | Caps. U | Caps. X | Caps. Y | Caps. V |
|---|---|---|---|---|
| Shell Plate (longitudinal) | 18 | 22 | 25 | 28 |
| Shell Plate (transverse) | 18 | 22 | 25 | 28 |
| Weld Metal | 28 | 35 | 39 | 44 |

Measured USE Decreases (%)

| Material | Caps. U | Caps. X | Caps. Y | Caps. V |
|---|---|---|---|---|
| Shell Plate (longitudinal) | 11 | 20 | 18 | 24 |
| Shell Plate (transverse) | 0 | 12 | 7 | 6 |
| Weld Metal | 31 | 38 | 40 | 40 |

Bounding USE Decreases Used For Projections (%)

| Material | Caps. U | Caps. X | Caps. Y | Caps. V |
|---|---|---|---|---|
| Shell Plate (longitudinal) | 18 | 22 | 25 | 28 |
| Shell Plate (transverse) | 18 | 22 | 25 | 28 |
| Weld Metal | 31 | 38 | 43 | 48 |

<u>*RAI B2.1.15-1*</u>

*In its description of the Reactor Vessel Materials Surveillance Program in LRA Section B2.1.15, the applicant stated that for DCPP Unit 1, the last capsule is expected to be withdrawn during the current operating term after it has accumulated a fluence equivalent to 60 years of operation. The applicant further stated that the remaining five standby capsules have low lead factors, will remain inside the vessel throughout the vessel lifetime, and will be available for future testing.*

*The latest surveillance capsule withdrawal schedule was submitted by the applicant by letter dated March 12, 2008 (Ref. 4 of the LRA), and approved by a staff safety evaluation dated September 24, 2008 (Ref. 5 of the LRA). This schedule proposed that capsule B, with a lead factor of 3.46, will be withdrawn at 21.9 EFPY. The fluence will be equivalent to 75.8 EFPY which is between one and two times the vessel EOLE fluence (54 EFPY fluence). This meets the ASTM E 185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," criterion. The proposed schedule shows four capsules to be left in the vessel, all with lead factors around 1.3.*

*Clarify whether four or five surveillance capsules will remain installed in the vessel during the period of extended operation.*

<u>PG&E Response to RAI B2.1.15-1</u>

Capsule B was a supplementary capsule inserted during the Unit 1 fifth refueling outage (2R5) in 1992 at 5.86 effective full power years (EFPY), as shown by note b in the Diablo Canyon Power Plant Final Safety Analysis Report Table 5.2-22. Current best estimate for the Unit 1 sixteenth refueling outage (1R16) in 2010, is 21.71 EFPY. Therefore the calculation would be (21.71-5.86) x 3.46 = 54.84 EFPY.

There will be four surveillance capsules left in the Unit 1 Reactor Vessel after 1R16. These capsules are U, X, and Z, which were installed before initial startup, and A, which was a supplementary capsule installed during 1R5.