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PG&E Letter DCL-16-043

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

Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
Supplement to Reactor Vessel Internals Aging Management Program, Unit 1 Pressurizer Spray Line Pipe Weld Flaw Growth Analysis, and Neutron Embrittlement for the Review of the Diablo Canyon Power Plant, Units 1 and 2, License Renewal Application

Dear Commissioners and Staff:

By Pacific Gas and Electric Company (PG&E) Letter DCL-09-079, "License Renewal Application," dated November 23, 2009, PG&E submitted an application to the U.S. Nuclear Regulatory Commission (NRC) for the renewal of Facility Operating Licenses DPR-80 and DPR-82, for Diablo Canyon Power Plant (DCPP) Units 1 and 2, respectively. The application included the license renewal application (LRA) and appendices.

By PG&E Letter DCL-15-150, "10 CFR 54.21(b) Annual Update to the Diablo Canyon Power Plant, Units 1 and 2, License Renewal Application, Amendment 51," dated December 21, 2015, PG&E submitted updated information related to the Reactor Vessel Internals Aging Management Program at DCPP and updated the LRA 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. By PG&E Letter DCL-16-023, "Response to NRC Letter dated February 2, 2016, 'Requests for Additional Information for the Review of the Diablo Canyon Power Plant, Units 1 and 2, License Renewal Application – Set 39 (TAC Nos. ME2896 and ME2897),'" dated February 25, 2016, PG&E submitted responses to requests for additional information (RAI) 4.2.1-2 related to neutron embrittlement.

In response to questions asked by the NRC staff during telephone conversations with PG&E on March 23, 2016, and April 7, 2016, PG&E is submitting supplemental information to PG&E Letters DCL-15-150 and DCL-16-023, RAI 4.2.1-2 in Enclosure 1. Enclosure 2 contains the affected LRA pages resulting from the supplemental information with the changes shown as electronic markups (deletions crossed out and insertions italicized).

PG&E does not revise any regulatory commitments (as defined by NEI 99-04) in this letter. However, PG&E makes a new commitment which is 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 declare under penalty of perjury that the foregoing is true and correct.

Executed on April 15, 2016.

Sincerely,

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

Enclosures

cc: Marc L. Dapas, NRC Region IV Administrator
Richard A. Plasse, NRC Project Manager, License Renewal
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Michael J. Wentzel, NRC Project Manager, License Renewal (Environmental)
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Pacific Gas and Electric (PG&E) Clarifications to PG&E Letter DCL-15-150, "10 CFR 54.21(b) Annual Update to the Diablo Canyon Power Plant, Units 1 and 2, License Renewal Application, Amendment 51" and PG&E Letter DCL-16-023, "Response to NRC Letter dated February 2, 2016, 'Requests for Additional Information for the Review of the Diablo Canyon Power Plant, Units 1 and 2, License Renewal Application – Set 39 (TAC Nos. ME2896 and ME2897)'"

Pacific Gas and Electric (PG&E) Clarifications to PG&E Letter DCL-15-150, "10 CFR 54.21(b) Annual Update to the Diablo Canyon Power Plant, Units 1 and 2, License Renewal Application, Amendment 51" and PG&E Letter DCL-16-023, "Response to NRC Letter dated February 2, 2016, 'Requests for Additional Information for the Review of the Diablo Canyon Power Plant, Units 1 and 2, License Renewal Application – Set 39 (TAC Nos. ME2896 and ME2897)'"

NRC Question 1:

Background:

By letter dated December 31, 2015, Pacific Gas and Electric Company (the applicant) amended the LRA and submitted the aging management programs and inspection plans for the reactor vessel internal (RVI) components at Diablo Canyon Power Plant (DCPP) Units 1 and 2. The programs are described and discussed in WCAP-17462-NP, Revision 1, for Unit 1 and WCAP-17463-NP, Revision 1, for Unit 2, which were enclosed in the December 31, 2015, submittal. The applicant provides the AMR line items for the RVI components in these reports.

Issue Part 1:

NRC LR-ISG-2011-04 includes GALL AMR line item IV.B2.RP-388, which may be used as the AMR basis for managing loss of fracture toughness in Westinghouse-design upper and lower core barrel assembly girth welds. The WCAP reports do not include any AMR line items for managing loss of fracture toughness in the RVI upper and lower core barrel assembly girth welds.

Request Part 1:

Explain why the AMR line item tables in the applicable WCAP reports do not include any AMR line items for managing loss of fracture toughness in the upper and lower core barrel assembly girth welds.

Issue Part 2:

NRC LR-ISG-2011-04 includes GALL Report AMR line items IV.B2.RP-356 and IV.B2.RP-355, which may be used as the AMR bases for managing cracking and loss of material in control rod guide tube (CRGT) support pins. Although the referenced WCAP reports have AMR line items for managing loss of material and cracking in specific CRGT bolting components, the reports do not include any AMR line items for managing cracking and loss of material in CRGT support pins. The staff seeks clarification whether the referenced bolting components are CRGT support pins.

<u>*Request Part 2:*</u>

Clarify whether the AMR line items for CRGT bolting components in the referenced WCAP reports correlate to AMR line items for CRGT support pins in AMR Items IV.B2.RP-355 and IV.B2.RP-356. If the CRGT bolts are not the same as CRGT split pins, explain why the referenced WCAPs do not include any AMR line items for managing cracking and loss of material of CRGT support pins.

<u>PG&E Response to NRC Question 1:</u>

Part 1:

License Renewal Application (LRA) Table 3.1.2-1 is revised to add LR-ISG-2011-04 aging management review (AMR) line item IV.B2.RP-388 for managing loss of fracture toughness in the upper and lower core barrel assembly girth welds.

Part 2:

CRGT Bolts
LRA Table 3.1.2-1 inadvertently included the CRGT bolts and did not include the CRGT split pins. LRA Table 3.1.2-1 is revised to delete LR-ISG-2011-04 AMR line items for IV.B2.BP-382 associated with CRGT bolts.

The following documents the process used for determining if components should be within scope of the RVI aging management program (AMP), and provides basis for conclusion that the CRGT bolts were screened out.

Initial screening of RVI components was based on a consideration of material properties (e.g., chemical composition) and operating conditions (e.g., neutron fluence exposure, temperature history, and representative stress levels) in order to determine the susceptibility of pressurized water reactor (PWR) RVI components to the identified aging mechanisms. This resulted in the binning of these RVI components as either susceptible or not susceptible to each of the eight degradation mechanisms, based on the degradation screening criteria.

Next, the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) performed a failure modes, effects, and criticality analysis (FMECA) of the RVI components. The FMECA process was discussed in detail in MRP-191, "Materials Reliability Program: Screening, Categorization and Ranking of Reactor Internals of Westinghouse and Combustion Engineering PWR Designs." The FMECA was a qualitative process that included expert elicitation by technical experts. Expert elicitation was used for developing the technical basis for categorization of various RVI components under different categories based on the combination of the likelihood of component degradation due to one or more of the eight degradation mechanisms, and the severity of safety consequences. Each component was assigned to one of three categories (for each degradation mechanism) ranging from insignificant effects (Category A) to potentially moderately significant effects (Category B) to potentially significant effects (Category C).

As detailed in MRP-191, the Westinghouse control rod guide tube assembly bolts were identified as not being susceptible to any of the eight identified degradation mechanisms. Additionally, the control rod guide tube assembly bolts were determined to have no identified consequences of failure. Based on not being susceptible to any of the identified degradation mechanisms and having no identified consequence of failure, the FMECA process concluded the control rod guide tube assembly bolts are a Category A component. Thus, the bolts are items for which aging effects are below the screening criteria, and the components were excluded from further consideration for inclusion within the population of components required within the RVI AMP. As noted in MRP-191, a final expert review of the result endorsed the conclusion that the control rod guide tube assembly bolts screened out as Category A components.

CRGT Split Pins
LRA Table 3.1.2-1 currently contains an AMR line item crediting the DCPP Water Chemistry Program (B2.1.2) for management of cracking in all RVI components (NUREG-1801, Volume 2, Revision 1 line item IV.B2-32). This addresses the LR-ISG-2011-04 AMR line item IV.B2.RP-355 for cracking in the CRGT split pins.

LRA Table 3.1.2-1 is revised to add LR-ISG-2011-04 AMR line item IV.B2.RP-356 for managing loss of material of the CRGT split pins by using the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program (B2.1.1) instead of the PWR Vessel Internals Program (B2.1.41), as discussed below. DCPP performs VT-3 inspections of the accessible portions of the Units 1 and 2 upper core plates, defined as B-N-3 components, during 10-year ISI inspections.DCPP performs VT-3 inspections of the accessible portions of the Units 1 and 2 upper core plates, defined as B-N-3 components, during 10-year ISI inspections. Although the split pins and guide tube assemblies do not provide a core support function, and are therefore not categorized as B-N-3 components, the 10-year ISI inspections provide a partial view of the accessible split pins associated with the peripheral guide tube assemblies from the top of the upper core plate. Additionally, a foreign object search and retrieval inspection of the fuel assemblies is performed every refueling outage during core offload and of the lower core plate prior to core reload. These inspections are expected to detect the presence of split pin fragments (loss of material) in the unlikely event that failure does occur.

See revised LRA Table 3.1.2-1 in Enclosure 2.

<u>*NRC Question 2:*</u>

<u>*Background*</u>*:*

In Applicant/Licensee Action Item (A/LAI) #5 of the NRC's December 16, 2011, safety evaluation for EPRI MRP Report No. MRP-227-A, the staff asked applicants owning Westinghouse-designed reactors to define and provide the acceptance criteria that would be applied to physical measurements of their reactor vessel internal (RVI) hold-down springs, as addressed in Table 4-3 of the MRP-227-A report. The action item applies only to Westinghouse-designed RVI hold-down springs that are made from austenitic stainless steel materials. The criteria do not apply to hold-down springs made from martensitic stainless steel materials, which are less prone to stress-relaxation aging mechanisms.

<u>*Issue*</u>*:*

The applicant's A/LAI response confirms that physical measurements will be performed on the Unit 1 hold-down spring consistent with the augmented inspection criteria in the MRP-227-A report. However, the applicant's A/LAI response for the Unit 1 hold-down spring does not specifically identify the acceptance criteria that will be used as the basis for evaluating the results of the physical measurements that will be applied to the RVI hold-down spring in Unit 1.

<u>*Request*</u>*:*

Provide and justify the acceptance criteria that will be used as the acceptance standard for evaluating the results of physical measurements performed on Unit 1 hold-down spring.

<u>PG&E Response to NRC Question 2:</u>

PG&E will establish acceptance criteria using the methodology in the NRC-approved version of WCAP-17096-NP, "Reactor Internals Acceptance Criteria Methodology and Data Requirements."

<u>*NRC Question 3:*</u>

<u>*Background*</u>*:*

Applicant/Licensee Action Item (A/LAI) #8, Subitem 5, of the NRC's December 16, 2011, safety evaluation for EPRI Technical Report (TR) No. MRP-227-A, addresses plant analyses that are applicable to RVI components and may need to be identified as TLAAs in the LRA. For RVI TLAAs involving metal fatigue analyses (i.e., cumulative usage factor [CUF] analyses), the action item addresses *how the impacts of environmentally-assisted fatigue will be factored into the bases for dispositioning the TLAAs under 10 CFR 54.21(c)(1)(i), (ii), or (iii), or else into the bases for managing fatigue-induced cracking of the components using the applicant's RVI management program.*

Issue:

In its response to A/LAI No. 8, Subitem 5, the applicant stated that the CLB does not include any analyses that are TLAAs for the RVI components at Units 1 and 2. LRA Section 4.3.3, as supplemented in the response to RAI 4.1-7, identifies that the following RVI components have CUF analyses qualifying as TLAAs: (a) upper core plates, (b) lower core plates, (c) lower support plates, (d) lower support columns, (e) core barrel nozzles, (f) lower supports, and (g) baffle-to-former bolts. The LRA uses the criterion in 10 CFR 54.21(c)(1)(iii) and the Metal Fatigue of Reactor Coolant Pressure Boundary Program (LRA AMP B3.1) as the basis for accepting these TLAAs and for demonstrating that the impacts of cumulative fatigue damage or fatigue-induced cracking will be adequately managed during the period of extended operation.

Request:

Explain how exposure of these components to the reactor coolant environment will be assessed for its impact on the CUF analyses for these components, or else on the bases for inspecting, monitoring and managing fatigue-induced cracking of the components under implementation of the PWR Vessel Internals Programs for the units (i.e., explain how the components will be assessed for environmentally-assisted fatigue).

PG&E Response to NRC Question 3:

For DCPP, as shown in LRA Table 3.2.1-1, all time-limited aging analysis components will be managed under the PWR Vessel Internals Program (B2.1.41), except for the core barrel nozzles. As shown in LRA Table 3.2.1-1, the core barrel nozzles will be managed by the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program (B2.1.1).

NRC Question 4:

Background:

In 2010, a licensee owning another Westinghouse-designed pressurized water reactor (PWR) reported the occurrence of cracked bolts in their clevis insert assemblies. The operating experience (OpE) was not assessed in WCAP-17462-NP, Revision 1, for Unit 1 or WCAP-17463-NP, Revision 1, for Unit 2.

<u>Issue</u>:

It is unclear whether the OpE with clevis insert bolt cracking is applicable to the design of the clevis insert assemblies at Diablo Canyon Units 1 and 2, and if so, whether the OpE has been evaluated for its impact on the "Existing Program" criteria (i.e., ASME Section XI Examination Category B-N-2 inspection requirement criteria) for the assemblies.

<u>Request</u>:

Explain whether the OpE associated with cracking of the Westinghouse-design clevis insert bolts is applicable to design of clevis insert assemblies at Diablo Canyon Units 1 and 2. If so, provide a technical discussion and justification basis that will establish whether the ASME Section XI Examination Category B-N-2 inspection criteria for the assemblies will need to be further augmented, such that the intended function of the clevis insert assemblies will be adequately maintained during the period of extended operation.

<u>PG&E Response to NRC Question 4:</u>

Applicability of Operating Experience:

The main function of the Lower Radial Support System (LRSS) is to limit or restrict lateral and rotational motion of the lower internals assembly while permitting axial and radial differential expansion. Although labeled as radial supports, the supports actually support the core barrel only in the tangential, or lateral, direction because the tangential gaps between the core barrel lower radial keys and clevis inserts are much smaller than the radial gaps.

Due to the small tangential clearance between the core barrel lower radial keys and clevis inserts, these components are potentially subjected to flow induced vibration loads and wear at the interfaces. The supports are designed to prevent excessive lateral and rotational displacement of the lower internals assembly during seismic and loss-of-coolant accident (LOCA) conditions. The supports also limit displacements and misalignments in order to avoid overstressing the core barrel and to ensure that the control rods can be freely inserted. Therefore, as long as the clevis inserts remain in place due to being limited by adjacent components, the design function of the LRSS will be maintained during seismic and LOCA conditions.

Because the clevis insert bolts for DCPP Units 1 and 2 are of similar design, of the same material, are torqued to a similar preload stress, and are subject to T_{cold} inlet temperatures as compared to the reference plant where clevis insert bolt cracking was observed, it is possible that these bolts can eventually experience degradation in a manner similar to that of the reference plant. Therefore, the operating experience discussed in this NRC question is applicable to DCPP Units 1 and 2.

Structural evaluations performed to justify continued operation with the as-found condition at the reference plant demonstrated that the LRSS would function as required and that safe operation was acceptable for two additional fuel cycles. The only concern was possible long-term effects, such as the potential for vibratory loads to eventually cause loosening and wear of the clevis insert and the subsequent increase in gaps between the clevis insert, lower radial key, and vessel lugs.

Through the Pressurized Water Reactor Owner's Group, an evaluation of the structural adequacy of the clevis insert design applicable to DCPP Units 1 and 2 was conducted to determine if degraded clevis insert bolts present a structural concern for safe operation. This evaluation is similar to the review performed for the reference plant that justified continued operation. The conclusions of the structural evaluations and loose parts assessment are summarized in the following paragraphs.

Clevis Insert Primary Plus Secondary Stress
The bending stress of the insert is maximized if it is assumed that one entire column of cap screws is broken and the other column of screws is intact. This forces the loose side of the insert to expand and contract to a greater extent relative to the support lug. With the maximum resulting interference during heatup and maximum tangential and radial loadings during steady state and cooldown, when a small clearance can exist, the resulting stress range remains within the primary plus secondary stress range analyzed in the generic analysis of record for this clevis insert design. Therefore, the increase in insert stress due to broken cap screws remains acceptable.

Cap Screw Primary Plus Secondary Stress
This scenario uses the same cap screw arrangement as discussed above where one column of cap screws is entirely broken. In this case, during cooldown and potentially during steady-state, when the insert is not tangentially preloaded against the support lug, the entire applied radial load on the insert is reacted by the intact cap screws. The resulting cap screw stress produced by this prying load on the insert is acceptable with four intact screws. However, with three or less intact screws, the allowable stress intensity can be potentially exceeded. During heatup, the clevis insert remains preloaded against the support lug, and this type of loading on the intact screws will not occur.

Clevis Insert Restraining Force (No Cap Screws)
If all of the cap screws are broken, and no restraint by the dowel pins is assumed, it is possible that the clevis insert can lose preload during steady state operation but it will still maintain its ability to perform its intended function. The clevis insert is restrained tangentially by the support lugs and restrained radially by the limited clearance with the radial key. In addition, the insert has a thick upper flange that prevents it from falling downward, and the downward force from the downcomer flow

will prevent it from working upward. Operating experience with damaged bolts and one dowel pin, as described in Reference 2 (Westinghouse IG-10-1), showed no discernible change in the clevis insert wear surfaces after operation for two additional cycles.

During core barrel removal at cold conditions, the interference fit of the insert provides greater frictional force than the applied frictional force produced by the key sliding upward against the insert. The two dowel pins will also provide additional vertical constraint of the insert. Therefore, the clevis insert design prevents separation of the insert during core barrel removal operations if the cap screws (and dowel pins) are nonfunctional.

Loose Parts Assessment
As discussed above, loss of the insert itself will not occur. Although over time, it may slowly displace radially inward toward the core barrel key, it will not move any further. The remaining engagement of the insert in the support lug will maintain adequate support of the core barrel against any normal, upset, or faulted condition loads. The insert cap screws have the same head design and locking device design as the reference plant. A lock bar is installed in a groove in the cap screw head and the bar is welded to the insert counterbore where the cap screw is inserted. If a cap screw head should separate, the lock bar can, over time, wear and separate, causing the cap screw head to be loose in the counterbore recess. The as-built radial gaps measured between the core barrel radial keys and the inserts are all less than the height of the cap screw heads. Therefore, the cap screw heads remain captured, unless over a long period of time, wear of the heads reduces the height of the heads by this amount. The cap screw head wear is expected to be small because the cap screw material is much harder than the clevis insert and radial key material.

Justification for Not Augmenting Reactor Vessel Internals Inspection Programs:

Safe operation of the reactors and primary systems is assured based on the structural evaluations summarized previously and operation with potential loose parts of the type and quantities that are no different than have already been evaluated. The ability of the LRSS to perform the intended design function under seismic and LOCA condition loadings is unrelated to the integrity of the clevis insert bolts and dowel pins that are used to hold the clevis inserts in place. Even if all clevis insert bolts and dowel pins fail, complete disengagement of the clevis insert will not occur, because of the small gap size between engaged components. As long as the clevis inserts remain in place, the design function of the LRSS will be maintained. Even if it were postulated that one of the clevis inserts becomes non-functional, the other lower radial supports are capable of resisting all of the internal and external asymmetric loads. Wear or some degradation of the components might occur, but they would still be expected to maintain functionality. Taken as a whole, the core barrel and LRSS are expected to maintain their design function with degraded clevis insert bolts. Based on the evaluations performed to date, there are no safety or operability concerns.

Relative to augmentation of the PWR Vessel Internals Program (B2.1.41), crack detection prior to clevis insert bolt failure is not required due to the inherent design redundancy as discussed previously. The only aspect to consider is the possibility of wear and looseness if the clevis insert bolts should become degraded. The MRP-227 categorization for wear only is based on the primary concern for clevis insert looseness and wear of the clevis insert and lower radial key interfacing surfaces that could potentially lead to increased motion at the bottom end of the core barrel, rather than clevis insert bolt material cracking. Stress corrosion cracking (SCC) was considered and screened in MRP-191. Actions to address SCC are included in MRP-227, under the Existing Programs group. Manifestation of clevis insert bolt cracking is identified as a result of the observation of wear. Existing inspections are already in place to account for this concern. Qualified personnel performing video camera inspections at 10-year internals can also detect wear and displacement of the clevis insert. Inspection of the clevis insert and lower radial key contact surfaces can detect wear-in relative to adjacent non-contact surfaces. If clevis insert bolt heads are observed to be loose, any movement of the insert relative to the vessel support lug can be easily observed. Anomalous conditions of this sort will result in corrective actions before any LRSS loss of function can occur.

Based on these considerations and observations, it is not necessary to augment the PWR Vessel Internals Program (B2.1.41) for crack detection. Continued monitoring of industry operating experience in this area will be performed to adjust the program as necessary.

NRC Question 5:

Background:

As amended by letter dated December 21, 2015, LRA Section 4.7.5 describes a TLAA for a weld flaw indication located in "ASME Code Class 1 pressurizer spray line pipe weld WIB-378." The TLAA states that the flaw was evaluated and accepted and the evaluation submitted to the NRC by letter dated November 3, 2015. The letter also states that, as part of the crack analysis, a fatigue crack growth evaluation was performed to determine if the weld is adequate for continued operation, and that the evaluation was projected for a 60-year plant life to the end of the period of extended operation. The applicant states that the fatigue crack growth analysis shows that allowable flaw depth will not be reached for the remainder of the projected plant life. As a result, the LRA, as amended by letter dated December 21, 2015, concludes that the analysis has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

In the submitted fatigue crack growth evaluation for a flaw in "ASME Code Class 1 pressurizer spray line pipe weld WIB-378" and the attached crack growth evaluation, the applicant claims that the flaw was identified by ultrasonic examinations and is acceptable for continued service. The flaw evaluation provided with the letter analyzes to the end of a 60-year plant life and states that the measured flaw depth is 0.08 in. with a final expected depth of 0.3276 in. compared to a maximum allowable crack depth of 0.3278 in.

Issue:

Considering the relatively small margin between final projected crack depth and the code allowable crack depth (0.0002 in.), it is not clear to the staff that the applicant's basis for analyzing the crack as acceptable provides reasonable assurance that the component's intended functions will be maintained to the end of the period of extended operation. In particular, it is unclear to the staff how uncertainties in the crack size and flaw geometry were taken into account in the flaw evaluation and how they affect the result.

Request:

Address the staffs concern regarding the uncertainty in the flaw geometry and measurement of the initial crack size and the projected final crack size. In particular, provide more information on the following items to support the conclusion that the flaw identified in weld WIB-378 will not exceed code allowable depth to the end of PEO.

a) *What are the dimensions of the flaw, as determined by UT?*
b) *What is the measurement uncertainty, for each flaw dimension (length and depth), of the methodology used to determine the measured crack size?*
c) *How was the uncertainty in the measurement used for this evaluation (i.e. was the uncertainty subtracted from the acceptance criteria or added to the measured flaw size)?*
d) *Does the indication found during the UT inspection penetrate the inside diameter (ID) of the pipe? If not, since the analysis assumes ID penetration, how much was the crack depth indication increased in order make this assumption for the purposes of the analysis?*

PG&E Response to NRC Question 5:

a) As provided in PG&E Letter DCL-15-131, "Pressurizer Spray Line Pipe Weld Flaw Evaluation," dated November 3, 2015, the ultrasonic testing (UT) measured flaw depth is 0.08 inches as measured from the inner diameter (ID). The UT measured length is 0.625 inches in the circumferential direction.

b) Pressurizer spray line pipe weld WIB-378 is managed by the ASME Section XI program. The ASME Section XI, Appendix VIII Supplement 2 qualification acceptance criteria use a root mean squared (RMS) error of 0.125 inches for flaw depth and a RMS error of 0.75 inches for flaw length. An Appendix VIII qualified procedure and Appendix VII qualified examiners were used for this inspection.

Consistent with the ASME Code, paragraph IWB-2420 for Successive Inspections, weld WIB-378 will be re-examined for the next three inservice inspection (ISI) periods. If results of subsequent re-examinations show the flaw remains unchanged and there is no measureable growth, the exam frequency will revert to standard ASME Section XI ISI requirements of one examination in 10 years. If during the successive-inspection monitoring timeframe the flaw has reached 50 percent of through wall depth, PG&E will reanalyze the current acceptability and projected flaw growth rate for the remainder of the period of extended operation and continue monitoring it on a periodic basis.

See revised LRA Table A4.1-1, Item 76 in Enclosure 2 regarding the commitment to reanalyze the current acceptability and projected flaw growth rate if 50 percent of through wall depth is reached during the successive-inspection monitoring timeframe.

c) In the flaw growth analysis, the actual indication measurements of 0.625 inches long and 0.08 inches deep were used (i.e., no uncertainty was added to the measured flaw size). Conservatisms are included in the external piping loads and transient definitions and the flaw growth analysis is based on the nominal pipe thickness (0.437 inches) instead of the actual pipe thickness (greater than 0.452 inches).

d) It is not known if the flaw actually contacts the ID of the pipe. At this location, the ID of the piping is not accessible for inspection. The ultrasonic signal did not provide adequate resolution to determine if the flaw is or is not ID surface connected. Per ASME Code Article IWA-3300, paragraph IWA-3310(b) proximity rules, the overall measured size of the indication was considered a surface planar flaw.

<u>NRC Question 6:</u>

In RAI 4.2.1-2, the staff requested that the applicant identify the inside surface neutron fluence for the nozzles and nozzle-to-nozzle weld components. The applicant provided the inside surface neutron fluence for the nozzle shell plate material, but not the nozzle base metal. Also if the inside surface neutron fluence for the nozzle base metal is above the $1x10^{17}$ n/cm^2 threshold for 54 EFPY, then the *applicant needs to provide the associated pressurized thermal shock (PTS) and upper shelf energy (USE) calculations.*

<u>PG&E Response to NRC Question 6:</u>

The 54 effective full power year (EFPY) maximum fast neutron fluences for the Unit 1 and Unit 2 nozzle to nozzle shell welds and nozzle base metal are provided below.

Material	**Fluence (n/cm^2)**	
	Unit 1 54 EFPY	**Unit 2 54 EFPY**
Outlet Nozzle to Nozzle Shell Welds – Lowest Extent Nozzles 1, 2, 3, and 4	2.56×10^{16}	6.71×10^{16}
Inlet Nozzle to Nozzle Shell Welds – Lowest Extent	3.39×10^{16}	
Nozzle 1		8.41×10^{16}
Nozzle 2		8.92×10^{16}
Nozzle 3		8.41×10^{16}
Nozzle 4		8.92×10^{16}

As shown above, the nozzle to nozzle shell welds do not exceed the 1×10^{17} n/cm^2 threshold for 54 EFPY. Thus, the associated pressurized thermal shock and upper shelf energy calculations are not provided.

License Renewal Application (LRA) Amendment 54
Affected LRA Table

License Renewal Application (LRA) Amendment 54
Affected LRA Table

LRA Table	Reason for Change
Table 3.1.2-1	NRC Question 1
Table A4.-1, Item 76	NRC Question 5

Table 3.1.2-1 Reactor Vessel, Internals, and Reactor Coolant System – Summary of Aging Management Evaluation – Reactor Vessel and Internals

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
RVI Core Barrel Assembly (Upper and Lower Core Barrel Girth Welds)	*DF, SLD, SS*	*Stainless Steel*	*Reactor Coolant and neutron flux (Ext)*	*Loss of fracture toughness*	*PWR Vessel Internals (B2.1.41)*	*IV.B2.RP-388*	*3.1.1.27*	*A, 3*
~~RVI Control Rod Guide Tube Assembly (Guide Tube Bolts)~~	SS	~~Stainless Steel~~	~~Reactor Coolant and neutron flux (Ext)~~	~~Cracking~~	~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)~~	~~IV.B2.BP-382~~	~~3.1.1.63~~	~~A, 3~~
~~RVI Control Rod Guide Tube Assembly (Guide Tube Bolts)~~	SS	~~Stainless Steel~~	~~Reactor Coolant and neutron flux (Ext)~~	~~Loss of material~~	~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)~~	~~IV.B2.BP-382~~	~~3.1.1.63~~	~~A, 3~~
RVI Control Rod Guide Tube Assembly (Split Pins)	*SS*	*Stainless Steel*	*Reactor Coolant and neutron flux (Ext)*	*Loss of material*	*ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)*	*IV.B2.BP-356*	*3.1.1.63*	*A, 3*

Table A4-1 License Renewal Commitments

Item #	Commitment	LRA Section	Implementation Schedule
76	*Pressurizer Spray Line Pipe Weld WIB-378 will be re-examined for the next three inservice inspection (ISI) periods as described in PG&E Letter DCL-16-043. If the during the successive-inspection monitoring timeframe the flaw has reached 50 percent of through wall depth, PG&E will reanalyze the current acceptability and projected flaw growth rate for the remainder of the period of extended operation and continue monitoring it on a periodic basis.*	*B2.1.1*	*Prior to the period of extended operation*