APPENDICES OF DOCUMENT EXCERPTS

Excerpts from PG&E & NRC License Renewal Updates and Requests for Additional Information (RAIs)

NOTE: All Excerpts listed in chronological order rather than in sequence noted in report:

APPENDIX A

Following excerpt from 2009 License Renewal Application (LRA) identifies axial weld#3-442C as the critical "limiting" element, although there are many welds which contain the same weld material with copper and nickel impurities that are more prone to radiological embrittlement as they are exposed to neutron bombardment (radiation) over time. "Neutron flux" is a measure of instantaneous radiation level and "neutron fluence" refers to the accumulative radiation exposure over time. Fluence calculations estimate cumulative exposure over time and can directly effect the estimated life, and aging effects on specific components that are subjected to material stress tests. NOTE: The estimated life of limiting weld 3-422C was listed at 32 effective full-power years in the original 2003 "coupon analysis" which is given the longer title: "Analysis of Coupon V from Pacific Gas and Electric Company Diablo Canyon Unit 1 Reactor Vessel Radiation Surveillance Program Technical Report", Westinghouse, Rawluski, Conermann & Hagler WCAP-15958, Rev 0, January 2003:

Section 4
TIME-LIMITED AGING ANALYSES

Unit 1

The data from the most recently withdrawn surveillance capsule, Capsule V, were not deemed credible [Reference 2, Appendix D]. Using Regulatory Guide 1.99 Position 1.1 methods, RT_{PTS} values were generated for beltline and extended beltline region materials of the Unit 1 reactor vessel for EOLE fluence values. The RT_{PTS} values for the Unit 1 materials are provided in Table 4.2-4. The projected RT_{PTS} values for EOLE did not meet the 10 CFR 50.61 screening criteria in all cases. The calculation [Reference 14] indicates the limiting weld material for Unit 1 is lower shell longitudinal (axial) weld 3-442C with a projected EOLE RT_{PTS} value of 280.4°F. Lower shell longitudinal weld 3-442C will satisfy the PTS screening criteria until approximately 43 EFPY. All other materials meet the 10 CFR 50.61 screening criteria. The limiting plate material on Unit 1 is the lower shell plate B4107-1 with a projected EOLE RT_{PTS} value of 156.2°F.

Using new fluence calc?: Limiting weld 3-442C RT-PTS value of 280.4F DIFFERS from 2003 coupon analysis which has 32EFPY. (was 250.4F)

The Unit 1 reactor vessel fluence will continue to be monitored in accordance with 10 CFR 50.61 as part of the DCPP Reactor Vessel Surveillance program (B2.1.15) to ensure that the reactor vessel material does not violate the PTS criteria.

PGE promises to continue to use 50.61 rule (not 61a) to check PTS safety

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Page 4.2-8 & 4.2-9

It is important to note that the named material test sample used to project the RT_{PTS} value of 280.4°F at End-Of-License Extension (EOLE). The RT_{PTS} is the temperature at which the material sample fails under stress at end-of-life, and refers to "Reference Temperature-Pressurized Thermal Shock". <u>All material stress test samples are supposed to fall below the <270°F temperature, so PG&E is referencing a sample that failed the test in order to lengthen the original projected life of 32EFPY to 43EFPY. This reference in the original 2009 LRA appears to be unorthodox and should be verified by the DCISC.</u>

APPENDIX CONTINUES ON FOLLOWING PAGE

APPENDIX B

Significance of LRA update letter from PG&E to NRC dated 12-21-11: PG&E changes fluence calculation (end-of-life radiation exposure estimate) of 2003 coupon analysis to significantly shift RT_{PTS} failure temperature of weld and plate materials using undisclosed Westinghouse Report WCAP-17299 and WCAP-17315. Failure temperature of weld# 3-442C shifts from 280.4°F (which is over <270°F allowed limit) to 243°F (15% change) based on alternative method under Reg. Guide1.99. Mid-page highlight modifies their strategy from deeming 2003 stress tests were "not credible" to claiming some results credible and applying RG1.99 Pos. 1.1 and Position 2.1, inconsistently. Fluence values are changed from specimen to specimen –not logical. Only limiting welds and plates RT-PTS values move 15% to 17%, all other values move 1% to 2% (as needed). Since there were no other sample tests since 2003, they had to adopt samples from another reactor to use shift USE and RT_{PTS} values, but this is neither described in text nor footnoted. It is highly irregular to shift material test data in this manner –without justification for the methodology.

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PG&E Letter DCL-11-136
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Section 4
TIME-LIMITED AGING ANALYSES

Unit 1 IMPORTANT: If data from Capsule V deemed "not credible", RG1.99 requires use of Position 1.1 & 1.2. Limiting weld RT-PTS value shifted from 280.4F to 243F using RG1.99 Position 2.1(only if data deemed credible?),

The data from the most recently withdrawn surveillance capsule, Capsule V, were not deemed credible [Reference 2~~,~~ Appendix D]. Using Regulatory Guide 1.99 ~~Position 1.1 methods~~, RT_{PTS} values were generated for beltline and extended beltline region materials of the Unit 1 reactor vessel for EOLE fluence values. The RT_{PTS} values for the Unit 1 materials are provided in Table 4.2-4. The projected RT_{PTS} values for EOLE ~~did not~~ meet the 10 CFR 50.61 screening criteria in all cases. The calculation [Reference ~~14~~*39*] indicates the limiting weld material for Unit 1 is lower shell longitudinal (axial) weld 3-442C with a projected EOLE RT_{PTS} value of ~~280.4~~*243*°F, *using Position 2.1*. ~~Lower shell longitudinal weld 3-442C will satisfy the PTS screening criteria until approximately 43 EFPY. All other materials meet the 10 CFR 50.61 screening criteria.~~ The limiting plate material on Unit 1 is the lower shell plate B4107-1 with a projected EOLE RT_{PTS} value of 156~~.2~~°F, *using Position 1.1 [Reference 39]*.References omitted at end, this #39 is likely WCAP-17315-NP footnoted on pg.82

Mixing RG1.99 methods on sample by sample basis, Allowed?

~~The Unit 1 reactor vessel fluence will continue to be monitored in accordance with 10 CFR 50.61 as part of the DCPP Reactor Vessel Surveillance program (B2.1.15) to ensure that the reactor vessel material does not violate the PTS criteria.~~

~~In 2009, the NRC approved the Final Rule 10 CFR 50.61a, *Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events* [Reference 15]. The revised rule amended its regulations to provide updated fracture toughness requirements based on updated analysis methods for protection against PTS events for PWR pressure vessels. It is anticipated that DCPP Unit 1 will meet these revised requirements through EOLE.~~

~~PG&E will implement the revised PTS (10 CFR 50.61a) rule at least three years prior to exceeding the PTS screening criterion of 10 CFR 50.61. In the event that the provisions of 10 CFR 50.61a cannot be met, PG&E will implement alternate options, such as flux reduction, as provided in 10 CFR 50.61.~~

The extended beltline materials were also evaluated. The results (Table 4.2-4) confirm the materials will not become limiting.

<u>IMPORTANT NOTE:</u> RG1.99, Criterion 3 specifically states that "scatter" or deviation between sets of RTPTS weld temp values may be greater than 28°F, but should not be greater than 56°F and here the difference is 37°F, -but at nearly double (68%) the projected life (32EFPY increased to 54EFPY. Taken together, the shift seems far too great to be credible. . Any fluence calculation that revises the effective full-power years (operating life) of the plant from 32EFPY to 54EFPY at the same time that RT_{PTS} shifts 15% would seem to falls outside of the uncertainty calculation range (margins of error). This is a fairly obvious anomaly that merits investigation. See tables in Appendix N showing Unit 1 and Unit 2 stress test data revisions from DCL-11-136 from pages 81-85.

APPENDIX C

Page 95 of DCL-11-136
Is one of many references in this document showing that PG&E has abandoned the strategy of complying through newer test rules in 10CFR 50.61a. It appears that they did not meet the criteria under this rule. They probably did not meet ASTM metallurgical standards, due to known flaws, e.g. weld material heat no. 27204 used on reactor vessel Unit 1.

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Final

Table A4-1 License Renewal Commitments

Item #	Commitment
23	~~DCPP will replace the current carbon steel with stainless steel clad CCP 2-2 pump casing in the CVCS with a completely stainless steel pump casing.~~ *Completed*
24	~~PG&E will implement the revised PTS rule (10 CFR 50.61a). In the event that the provisions of 10 CFR 50.61(a) cannot be met, PG&E will implement alternate options, such as flux reduction, as provided in 10 CFR 50.61.~~ *Deleted*

APPENDIX D

DCL-11-136 (letter with LRA updates dated 12-21-11) Page 98 states that “Reactor Vessel Surveillance Program experience at DCPP is evaluated and monitored to maintain an effective program”. This fails to mention that they were unable to extract Coupon B from Unit 1 which they had wanted to withdraw in 2010 to meet EPRI Material Reliability Program MRP-326, but were unable to do so because of a stuck access plug. The statement that there is no “unique plant-specific operating experience” ignores known flaws in reactor welds and plates. NUREG 1801 is also referred to the Generic Aging Lessons Learned (GALL). PG&E should be studying ALL of the lessons learned from other plants that happened to have the same metallurgical flaws.

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Appendix B
AGING MANAGEMENT PROGRAMS

Exceptions to NUREG-1801

None

Enhancements

None

Operating Experience

Reactor Vessel Surveillance program experience at DCPP is evaluated and monitored to maintain an effective program. This is accomplished by promptly identifying and documenting (using the Corrective Action Program) any conditions or events that could compromise the program. In addition, industry operating experience provides input to ensure that the program is maintained. The DCPP operating experience findings for this program identified no unique plant specific operating experience; therefore DCPP operating experience is consistent with NUREG-1801. Statement ignores known alloy flaws

APPENDIX CONTINUES ON FOLLOWING PAGE

APPENDIX E

PGE Letter to NRC, dated 12-20-12, Acc. No. ML12356A179 explains trouble withdrawing Capsule B in 2010 to support EPRI MRP-326 Program. As of May 2023, Capsule B has still not been withdrawn and in later correspondence (ML23076A210 dated March 2023) PG&E is asking to withdraw as late as Spring 2025, so stress test results will not be available until late 2026. In a contradictory narrative, a more recent March 2023 letter from PG&E to NRC (DCL-23-038) states: "Because the Unit 1 Capsule B removal was to support the DCPP license renewal, it was not withdrawn in 1R23" (23rd scheduled refueling outage)…. "By Reference 3, PG&E notified the NRC of the intent to submit a new DCPP license renewal application no later than December 2023. Consequently, Unit 1 reactor pressure vessel fluence data is now needed for license renewal and PG&E requests revision to the Unit 1 reactor vessel material surveillance program withdrawal schedule to allow withdrawal of Capsule B during the Unit 1 24th refueling outage (Fall 2023) or Unit 1 25th refueling outage (Spring 2025)."

NOTE:
Excerpt Indicates PG&E was NOT able to remove Capsule as early as 2010 in order to comply with EPRI Material Reliability Program (MRP-326)

4.2.1 Neutron Fluence Values

Summary Description

Loss of fracture toughness is an aging effect caused by the neutron embrittlement aging mechanism that results from prolonged exposure to neutron radiation. This process results in increased tensile strength and hardness of the material with reduced toughness. The rate of neutron exposure is defined as neutron flux, and the cumulative degree of exposure over time is defined as neutron fluence. As neutron embrittlement progresses, the toughness/temperature curve shifts down (lower fracture toughness as indicated by Charpy upper-shelf energy or C_V USE), and the curve shifts to the right (brittle/ductile transition temperature increases). Neutron fluence projections are made in order to estimate the effect on these reactor vessel material properties (Section 4.2.2 and Section 4.2.3), and to determine if additional reactor vessel materials will be exposed to fluence greater than 1×10^{17} n/cm^2 (E>1.0 MeV) as a result of license renewal (extended beltline).

Analysis

Unit 1

The last capsule withdrawn and tested from Unit 1 was Capsule V at the end-of-cycle (EOC) 11. At that point, Unit 1 had operated for 14.27 EFPY. This capsule had a lead factor of 2.26 resulting in an exposure equivalent to 32.25 EFPY of operation. The results were documented in WCAP-15958 [Reference 2].

This exposure is less than that expected at EOLE. In PG&E Letter DCL-08-021, PG&E requested a change to the withdrawal date of Unit 1 Capsule B from 20.7 EFPY to 21.9 EFPY in order to capture enough fluence data for EOLE. The change was approved by the NRC in a Safety Evaluation dated September 24, 2008, *Diablo Canyon Nuclear Power Plant, Unit No. 1 – Approval of Proposed Reactor Vessel Material Surveillance Capsule Withdrawal Schedule (TAC No. MD8371)* [Reference 13].

Trouble removing Capsule B as early as 2010

During the scheduled Unit 1 Sixteenth Refueling Outage (1R16), refueling personnel were not able to remove the Capsule B access plug on the reactor core barrel flange. In PG&E Letter DCL-10-141, dated October 25, 2010, PG&E requested a change to the withdrawal date of Unit 1 Capsule B from 21.9 EFPY to 23.2 EFPY. The change was approved by the NRC in a Safety Evaluation dated October 29, 2010, *Diablo Canyon Nuclear Power Plant, Unit No. 1 – Approval of Proposed Reactor Vessel Material Surveillance Program Withdrawal Schedule (TAC No. ME4924)* [Reference 38].

In PG&E Letter DCL-11-122, dated November 21, 2011, PG&E requested a change to the withdrawal date of Unit 1 Capsule B from 23.2 EFPY to 33 EFPY to support data acquisition for the EPRI MRP-326, Draft E, "Materials Reliability Program: Coordinated PWR Reactor Vessel Surveillance Program (CRVSP)." The withdrawal date corresponds to the Unit 1 23rd refueling outage (1R23), which is scheduled for May 2022. The change was approved by the NRC in a Safety Evaluation dated March 2, 2012, Diablo Canyon Power Plant, Unit No. 1: Safety Evaluation for the Request to Revise the Reactor Vessel Material Surveillance Program Withdrawal Schedule (TAC ME7615) [Reference 41].

APPENDIX F

Enclosure 2
PG&E Letter DCL-12-124
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Section 4
IMPORTANT: **TIME-LIMITED AGING ANALYSES**

Excerpt from DCL12-124 from PG&E to NRC
Suggests that some welds in Unit 1 did not meet RT-PTS limits at 54 EFPY (60 year life)

Both Units

Neutron fluence calcs from WCAP17299, (see Table 4.2-1)

Based on the guidance specified in Regulatory Guide 1.190, a neutron fluence assessment of the beltline and extended beltline regions was performed by Westinghouse in WCAP-17299-NP [Reference 40], for Units 1 and 2, through EOLE. The peak calculated fast neutron fluence values at the pressure vessel clad/base metal interface are shown in Table 4.2-1 and Table 4.2-2 for Units 1 and 2, respectively. These fluence data tabulations include fuel cycle specific power distributions through the end of Cycle 16 for Units 1 and 2, as well as fluence projections at several intervals out to 54 EFPY.

Wasn't flux reduction (fuel configuration) used during this timeframe starting about 2010-2011? Later docs (DCL15-121?) eliminate flux reduction requirement.

The calculations account for a Unit 1 core power uprate from 3338 MWt to 3411 MWt at the onset of Cycle 11. Fluence projections beyond the end of Cycle 16 on Units 1 and 2 are based on the assumption that the spatial core power distributions are defined by the average of Cycles 13-15 for Units 1 and 2

Are average of these cycles the right benchmark?

IMPORTANT

In WCAP 17299 some nozzle welds exceeded RT-PTS limits at 54EFPY.

They don't give ID number for welds that exceed limit. This statement is contradicted in conclusions on page 50 where it says all OK to 54EFPY.

For license renewal, Westinghouse performed additional calculations to define which materials in the DCPP pressure vessels, other than beltline materials, are projected to exceed the threshold neutron fluence of $1x10^{17}$ n/cm^2 at 54 EFPY (extended beltline materials). The results of these calculations are documented in WCAP-17299-NP [Reference 40], for Units 1 and 2, through EOLE For both units, although the nozzle shell course and the associated nozzle shell to intermediate shell weld are projected to exceed the $1x10^{17}$ n/cm^2 threshold, the nozzles themselves as well as the nozzle to nozzle shell welds remain below the $1x10^{17}$ n/cm^2 threshold through 54 EFPY. Likewise, the lower shell to lower head weld remains below $1x10^{17}$ n/cm^2 through 54 EFPY for both units.

Table 4.2-3 shows the EOLE fluence values for all beltline and extended beltline materials for both Units 1 and 2.

IMPORTANT:
Excerpt from DCL-12-124 correspondence to NRC including updates to the License Renewal Application:
Based on WCAP-17299 (Westinghouse) fluence calculations, weld and plate materials failure temperature ratings (RT_{PTS} values) decrease significantly below previous Westinghouse coupon analysis (WCAP-15958). But, mid-page above, this LRA update from PG&E to the NRC states that "nozzle shell… weld are projected to exceed the … [limit] threshold… through 54EFPY. With this disclosure, how is Unit 1 later found to meet requirements. Please note also, this contradicts conclusions on page 50 of the same report. If any of the welds doesn't meet limits, then conclusions should not state that all the welds meet thresholds.

APPENDIX CONTINUES ON FOLLOWING PAGE

APPENDIX G

RAI: 4.2.3-1: NRC response on 9-24-15 to DCL-11-136 (Dec. 21, 2011): This NRC Request for Additional Information RAI Set 38, ML15217A481, dated 9-24-15, Pg.7-8 refers to RG1.99, Critrion 3, and raises concern for deviation in sample data in regard to Position 2.1 when two or more data sets available. Also raises concern that PG&E's new fluence calculation referred to in DCL-11-136 is not applied to all material samples of Heat No. 27204 (limiting, flawed weld material) to 54 EFPY Upper Shelf Energy (USE) analysis for all welds of the same flawed material. Note: USE_{CV} calculations are generic end-of-life (EOL) fracture toughness calculations that are usually extrapolated mathematically. NRC specifically asks for a justification "why all capsule data in Unit 1 2003 coupon analysis (WCAP-15958) have not been used as the basis for calculating the 54EFPY USE values. This is not a blanket approval of PG&E's proposed methodology, and it suggests they want to see the calculations that PG&E has yet to reveal at this point.

RAI 4.2.3-1

Excerpt from 9-24-15 NRC Letter
RAI Set 38, ML15217A481, Pg7&8

Background:

In Pacific Gas and Electric Company (PG&E) Letter DCL-11-136 dated December 21, 2011, the applicant provided an update of the upper shelf energy (USE) analysis for ferritic components in the reactor pressure vessels (RPVs) of Diablo Canyon, Units 1 and 2. The applicant stated that, in accordance with Regulatory Guide (RG) 1.99, Revision 2, the USE data from Unit 1 surveillance Capsule V were determined not to be credible and were, therefore, not included in the USE projections for Unit 1 RPV components represented in the Diablo Canyon RPV surveillance program for Unit 1. Instead, the applicant stated that the USE values were projected to 54 effective full power years (EFPY) of operation using USE analysis methods and criteria that are given in Position 1.2 of RG 1.99, Revision 2.

IMPORTANT

Issue:

Although this refers to Position 2.1 and PG&E is using Pos.1.2 to project USE values at 54EFPY, the scatter of data should not be more than 28F for welds and even at high fluence deviation, not more than 56F.

Page No. 1.99-2 in RG 1.99, Revision 2, establishes the following regulatory discussion regarding the application of Charpy-impact data for neutron fluence-dependent RPV adjusted reference temperature calculations and USE analyses:

This is Reg Guide1.99 Rev2, Criterion 3

PG&E's new fluence calcs result in extrapolation of weld 3-442C fracture temp at 43F, revised down from 250.9F-completely out of range.

> When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Regulatory Position 2.1 normally should be less than 28 °F for welds and 17 °F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fail this criterion for use in . . . [ΔRT_{NDT}] . . . shift calculations, they may be credible for determining decrease in upper-shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E185-82.

The staff seeks further justification why all capsule data (i.e., those from the Capsule S, Y, and V Charpy-impact tests of materials representing Weld Heat 27204 in the Unit 1 RPV material surveillance program) have not been applied to the 54 EFPY USE analyses for RPV weld components in Unit 1 fabricated from the same weld heat.

Request:

IMPORTANT

Justify why all capsule data (i.e., those from the Capsule S, Y, and V Charpy-impact test specimens for Weld Heat 27204 in the Unit 1 reactor vessel material surveillance program as reported and analyzed in WCAP-15958, Rev. 0) have not been used as the basis for calculating the 54 EFPY USE values for Unit 1 RPV weld components fabricated from the same weld heat (i.e., for the USE calculations of intermediate shell axial welds 2-442 A, B and C, and lower shell axial welds 3-442, A, B, and C).

It appears NRC is reluctant to approve new DCL-11-136 interpretation of 2003 data because PG&E did NOT ASSESS LIMITING WELD 3-442C when it changed the fluence calc basis to 54EFPY

APPENDICES CONTINUE ON FOLLOWING PAGE

APPENDIX H

RAI 4.2.2-4

Excerpt from 9-24-15 NRC Letter
RAI- Set 38 ML15217A481

Background:

In PG&E Letter DCL-11-136 (Dec. 21, 2011), the applicant provided an update of the pressurized thermal shock (PTS) analysis for ferritic components in the RPVs of Diablo Canyon, Units 1 and 2.

Issue 1:

The staff performed independent PTS calculations for the Unit 1 RPV beltline and extended beltline components (54 EFPY) and has verified that all ferritic components in the beltline and extended beltline regions of the Unit 1 RPV will satisfy the PTS screening criteria for the components through 60 years of licensed operations (i.e., through 54 EFPY). However, some of the analysis parameter values independently calculated by the staff differ from those reported

NRC staff verifies PTS calc through 54EFPY, BUT parameter values differ. Is this a cover for parameters being out of range? NEED TO VERIFY THE CF VALUES USED BY PG&E UNDER RG1.99 POSITION1.2 which requires chemical analysis of samples. Were these chemical values within expected ranges?

for RT_{PTS} assessment parameters in license renewal application (LRA) Table 4.2-4 for Unit 1 or in LRA Table 4.2-5 for Unit 2.

Request 1:

a) Margin term values for Unit 1 RPV upper shell plates B4105-1 (Heat No. C2824-1) and B4105-2 (Heat No. C2824-2): Provide the σ_U and σ_Δ values used to calculate the margin term value for the RT_{PTS} calculation and the basis for reporting a margin term value of 39.2 ºF for these components.

b) Margin term values for Unit 1 RPV upper shell plate B4105-3 (Heat No. C2608-2B): Provide the σ_U and σ_Δ values used to calculate the margin term value for the RT_{PTS} calculation and the basis for reporting a margin term value of 41.2 ºF for these components.

All welds with heat no.27204 should be regarded as limiting welds. is a CF of 214.1F within range for this material spec?

c) Margin term values for Unit 1 RPV intermediate shell axial welds 2-442 A, B, and C, and lower shell axial welds 3-442 A, B, and C (all made from Heat No. 27204): Provide the σ_U and σ_Δ values used to calculate the margin term value for the RT_{PTS} calculation and the basis for reporting the margin term value of 44.0 ºF for these components.

d) Chemistry factor values for Unit 1 RPV intermediate shell axial welds 2-442 A, B, and C, and lower shell axial welds 3-442 A, B, and C (all made from Heat No. 27204): Provide the basis for reporting a chemistry factor of 214.1 ºF for these components.

e) Chemistry factor values for Unit 2 RPV upper shell axial welds 1-201 A, B, and C, and intermediate shell axial welds 2-201 A, B, and C (all made from Tandem Heat 21935/12008): Provide the basis for reporting a chemistry factor of 204.6 ºF for these components.

f) Provide the methodology basis (i.e., plant-specific, generic, NRC-generic, MTEB 5-2, etc.) of the $RT_{NDT(U)}$ value that was reported for each RPV beltline or extended beltline component referenced in LRA Table 4.2-4 and in LRA Table 4.2-5.

RAI 4.2.2-4
NRC 9-24-15 response to PG&E letter DCL11-136, (ML15217A481)
Based on PG&E's assumptions, NRC found RV components compliant through 54EFPY, but had different results and wants to know why. They specifically ask for confirmation of a number of variables including tested chemistry factor of limiting welds required under this method. (RG1.99 Position 1.2)
IMPORTANT:
It doesn't appear that NRC is questioning the shift in RT_{PTS} value of limiting weld 3-442C from 280.4°F at 43EFPY to 243°F at 54EFPY or that normally as EFPY goes up, the RT_{PTS} should go up as well rather than down. The NRC doesn't seem to question the fact that under the new fluence calculation, the projected EFPY has nearly doubled at the same time that the RT_{PTS} values have dropped, seemingly beyond range of uncertainty calcs.

APPENDIX I

PG&E Response: DCL-15-121 (10-21-2015) ML15294A437, PG&E confirms no longer trying to use new alternative compliance rule 50.61a which requires In-Service Inspection (ISI) referred to here as "special methods" such as ultrasonic testing.

They reverse their suggestion that there were non-compliant nozzle welds in the reactor vessel saying now they "will not be limiting". They suggest managing through RV surveillance program TLAA (Time Limiting Aging Analysis).

Enclosure 3
PG&E Letter DCL-15-121
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Section 3.3
AGING MANAGEMENT OF REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

3.1.2.2.3.1 Loss of Fracture Toughness due to Neutron Irradiation Embrittlement - TLAA

Evaluation of loss of fracture toughness is a TLAA as defined in 10 CFR 54.3. TLAAs are evaluated in accordance with 10 CFR 54.21(c)(1).

Paragraph deleted in PG&E Letter DCL-13-119

~~For the Unit 1 reactor vessel, PG&E will implement the revised PTS rule, 10 CFR 50.61a, at least three years prior to exceeding the PTS screening criterion of 10 CFR 50.61. In the event that the provisions of 10 CFR 50.61 a cannot be met, PG&E will implement alternate options, such as flux reduction, as provided in 10 CFR 50.61.~~

For the Unit *1 and Unit* 2 reactor vessels, recent coupon examinations demonstrated that beltline materials will remain limiting, and that adequate adjusted reference temperature, upper shelf energy, and pressurized thermal shock screening temperature margin will remain at the end of the period of extended operation; and therefore that subsequent revisions to pressure-temperature limits will provide adequate operating margin, without the use of special methods.

An evaluation of the axial fluence distribution for the reactor vessel nozzles found that the projected embrittlement parameters for these materials will not be limiting. Loss of fracture toughness for the reactor vessel shell and nozzles is managed with the Reactor Vessel Surveillance program (B2.1.15). Section 4.2 describes the disposition of these neutron embrittlement TLAAs.

APPENDIX J

From same letter DCL-15-121: Reference explains to NRC that 2003 Capsule V coupon analysis was "not credible" for RTNDT calc (Ref.Temp. Nill-Ductility Transient) for each weld using stress test, but was deemed credible for Cv USE projections at 54EFPY. This seems to directly contradict a reference on Pg. 86 of DCL-11-136 (dated 12-21-11) that: "In accordance with Reg. Guide 1.99, the CV USE data from Unit 1 surveillance Capsule V were determined NOT to be credible and were, therefore, not included in the EOLE C_V USE projections". So how did that get reversed? The answer probably lies in the application of the new fluence calculations in WCAP-17299 which moved all of the 2003 coupon analysis RT_{PTS} values 6-fold.

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PG&E Letter DCL-15-121
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Section 4
TIME-LIMITED AGING ANALYSES

4.2.3 Charpy Upper-Shelf Energy

Unit 1

In accordance with Regulatory Guide 1.99, the C_V USE data from Unit 1 surveillance Capsule V were determined not to be credible *for determination of ΔRT_{NDT}, but*~~and~~ were *credible for determining* ~~, therefore, not included in~~ the EOLE C_V USE projections.

APPENDICES CONTINUE ON FOLLOWING PAGE

APPENDIX K

From same letter DCL-15-121: PG&E responds to NRC's 9-24-15 RAI4.2.1-1 request for clarification whether neutron fluence calculation in WCAP-17299 (Westinghouse never given to NRC) is consistent with other regs, RG1.190 etc. PG&E offers a 2-sentence reply in effect simply saying "yes", without further detail.

RAI 4.2.1-1

Background:

Attachment 2 of the applicant's 2011 annual update (December 21, 2011) indicates that a neutron fluence assessment of the beltline and extended beltline regions through the period of extended operation was performed by Westinghouse in WCAP-17299-NP, "Fast Neutron Fluence Update for Diablo Canyon Unit 1 and Unit 2 Pressure Vessels," Revision 0, February 2011.

In the following reference, the applicant indicated that its methods used to develop the calculated reactor vessel fluence are consistent with the NRC-approved methodology described in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," Revision 2, January 1996.

Incorrect Reference: Actually WCAP 15958

- *WCAP-15985, Revision 0, "Analysis of Capsule V from Pacific Gas and Electric Company Diablo Canyon Unit 1 Reactor Vessel Radiation Surveillance Program," January 2003 (ADAMS Accession No. ML031400342)*

Issue:

The applicant did not clearly address whether the neutron fluence methodology used in WCAP-17299-NP, Revision 0 and the 2011 annual update is consistent with the methodology described in WCAP-14040-NP-A, Revision 2.

Request:

Clarify whether the neutron fluence calculational methodology used in WCAP-17299-NP, Revision 0 and the applicant's 2011 annual update is consistent with the methodology described in WCAP-14040-NP-A, Revision 2. If not, provide additional information to demonstrate that the applicant's fluence methodology adheres to Regulatory Guide 1.190.

PG&E Response to RAI 4.2.1-1

The neutron fluence calculation methodology used in WCAP-17299-NP, Revision 0 and PG&E's 2011 annual update is consistent with the methodology described in WCAP-14040-NP-A, Revision 4. WCAP-14040-NP-A, Revision 4 fluence methodology adheres to NRC-approved Regulatory Guide (RG) 1.190.

NOTE:
PG&E does not demonstrate compliance of WCAP-17299 fluence calculation , doesn't submit WCAP-17299, they simply state it complies.

RAI 4.2.3-1

Background:

In Pacific Gas and Electric Company (PG&E) Letter DCL-11-136 (Dec. 21, 2011), the applicant provided an update of the upper shelf energy (USE) analysis for ferritic components in the reactor pressure vessels (RPVs) of Diablo Canyon, Units 1 and 2. The applicant stated that, in accordance with Regulatory Guide (RG) 1.99, Revision 2, the USE data from Unit 1 surveillance Capsule V were determined not to be credible and were, therefore, not included in the USE projections for Unit 1 RPV components represented in the Diablo Canyon RPV surveillance program for Unit 1. Instead, the applicant stated that the USE values were projected to 54 effective full power years (EFPY) of operation using USE analysis methods and criteria that are given in Position 1.2 of RG 1.99, Revision 2.

Upper Shelf Energy (USE) values deemed "not credible" under RG1.99 criterion 3, but neverthless used to project 54EFPY USE values under Position 1.2, using data from Palisades reactor

APPENDIX L

Excerpt from Reg Guide 1.99 Position 1.2 clearly states: “Charpy upper-shelf energy should be assumed to decrease as a function of fluence and copper content as indicated in Fig. 2 (equation).

Based on PG&Es WCAP-17299 fluence calcs, RT_{PTS} values seem to violate the principle demonstrated here.

Limitations section suggests calculation method may not apply to Heat No.27204 limiting welds.

Application of procedure requires use of CF factors consistent with material specifications and “compositions beyond the range found in the data bases used for this guide should be justified by submittal of data.”

1.2 Charpy Upper-Shelf Energy

Charpy upper-shelf energy should be assumed to decrease as a function of fluence and copper content as indicated in Figure 2. Linear interpolation is permitted.

$$\Delta RT_{NDT} = (CF)\, f^{(0.28\, -\, 0.10 \log f)} \qquad (2)$$

1.3 Limitations

Application of the foregoing procedures should be subject to the following limitations:

1. The procedures apply to those grades of SA-302, 336, 533, and 508 steels having minimum specified yield strengths of 50,000 psi and under and to their welds and heat-affected zones.

2. The procedures are valid for a nominal irradiation temperature of 550°F. Irradiation below 525°F should be considered to produce greater embrittlement, and irradiation above 590°F may be considered to produce less embrittlement. The correction factor used should be justified by reference to actual data.

3. Application of these procedures to fluence levels or to copper or nickel content beyond the ranges given in Figure 1 and Tables 1 and 2 or to materials having chemical compositions beyond the range found in the data bases used for this guide should be justified by submittal of data.

APPENDICES CONTINUE ON THE FOLLOWING PAGE

Appendix M

Supplemental Information from PG&E correspondence regarding substitution of coupon stress test data from another Westinghouse reactor as a means to bring the DCPP Unit 1 reactor into compliance with 10 CFR 50.61 and ASTM E185-82:

Letter from Tom Jones, PG&E Director of Government Relations
April 13, 2023

Dear Bruce,

We are sorry to read about your mom's health challenges and wish you and her the best….

When are PTS evaluations conducted?
Evaluations are periodically updated when required by regulation and during the license renewal process.
o PG&E provided the DCPP PTS evaluation for the license renewal period in the original license renewal application submitted to the NRC in 2009 (here, Section 4.2.2 starting in the last paragraph on page 948 of the PDF). Based on data available at that time, the PTS evaluation results showed the Unit 1 reactor vessel would not meet the NRC's embrittlement limits for the entire license renewal period of 60 years.
o In 2011, based on new material coupon data available, PG&E revised the Unit 1 PTS evaluation and license renewal application (here, Section 4.2.2 starting in the third paragraph on page 84 of the PDF). The PTS evaluation results showed the Unit 1 reactor vessel would meet the NRC's embrittlement limits for the entire license renewal period of 60 years.

Have the DCPP PTS evaluations been independently reviewed? Both the NRC and DCISC have conducted independent reviews of DCPP's PTS evaluations and agree the DCPP reactor vessels are within the NRC's limits for embrittlement. See the DCISC's evaluation here, Section 4.23.2 (page 323 of the PDF).

The DCISC has previously reviewed the proprietary information. The NRC regulation was revised while the initial license renewal application was under review to allow for the use of representative coupons from other plants to be used for reactor vessel integrity evaluations. This revised methodology is an acceptable methodology for the entire nuclear industry and was applied for Diablo Canyon Power Plant and was reviewed and accepted by both the NRC and the DCISC. As has been stated in public meetings, PG&E is planning to remove and test the remaining Unit 1 reactor vessel coupon as part of the new license renewal (LR) application. During the LR process, the NRC will further evaluate this issue.

Regards,
Tom
Cc: Trevor Rebel, Philippe Soenen

APPENDIX N

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Section 4
TIME-LIMITED AGING ANALYSES

Table 4.2-4 DCPP Unit 1 Vessel RT_{PTS} at 54 EFPY(i)

Material Description			~~Sourc~~e R.G 1.99, Rev.2 Position	Chemical Composition		Chemistry Factors °F	Initial RT_{NDT} °F	EOLE Fluence 10^{19} n/cm² E>1.0 MeV	Fluence Factor	Δ~~RT_{PTS}~~ RT_{NDT} °F	Margin °F	RT_{PTS} °F	Screening Criteria °F	Extended Beltline Region
Location	Heat No.	Type		Cu Wt%	Ni Wt%									
Upper Shell Plate B4105-1	C2624	A 533B	1.1~~Ref-14~~	0.120	0.56	82.2	28	~~0.0286~~0.0341	~~0.2136~~0.2365	~~17.6~~19.4	~~38.3~~39.2	~~83.8~~87	≤270	Yes
Upper Shell Plate B4105-2	C2624-2	A 533B	1.1~~Ref-14~~	0.120	0.57	82.4	9	~~0.0286~~0.0341	~~0.2136~~0.2365	~~17.6~~19.5	~~38.3~~39.2	~~64.9~~68	≤270	Yes
Upper Shell Plate B4105-3	C2608-2B	A 533B	1.1~~Ref-14~~	0.140	0.56	98.2	14	~~0.0286~~0.0341	~~0.2136~~0.2365	~~21.0~~23.2	~~39.9~~41.2	~~74.9~~78	≤270	Yes
Intermediate Shell Plate B4106-1	C2884-1	A 533B	1.1~~Ref-14~~	0.125	0.53	85.3	-10	~~2.06~~2.02	~~1.1968~~1.1917	~~102.1~~101.7	34	126~~.1~~	≤270	No
Intermediate Shell Plate B4106-2	C2854-2	A 533B	1.1~~Ref-14~~	0.12	0.50	81	-3	~~2.06~~2.02	~~1.1968~~1.1917	~~96.9~~96.5	34	~~127.9~~128	≤270	No
Intermediate Shell Plate B4106-3	C2793-1	A 533B	1.1~~Ref-14~~	0.086	0.476	55.2	30	~~2.06~~2.02	~~1.1968~~1.1917	~~66.1~~65.8	48.1	144~~.1~~	≤270	No
→Using non-credible surveillance data	C2793-1	A 533B	2.1	0.086	0.476	37.4	30	2.02	1.1917	44.6	48.1	123	≤270	No
Lower Shell Plate B4107-1	C3121-1	A 533B	1.1~~Ref-14~~	0.13	0.56	89.8	15	~~2.04~~2.01	~~1.1943~~1.1904	~~107.2~~106.9	34	156~~.2~~	≤270	No
Lower Shell Plate B4107-2	C3131-2	A 533B	1.1~~Ref-14~~	0.12	0.56	82.2	20	~~2.04~~2.01	~~1.1943~~1.1904	~~98.2~~97.9	34	152~~.2~~	≤270	No
Lower Shell Plate B4107-3	C3131-1	A 533B	1.1~~Ref-14~~	0.12	0.52	81.4	-22	~~2.04~~2.01	~~1.1943~~1.1904	~~97.2~~96.9	34	109~~.2~~	≤270	No
Upper Shell Long. Weld 1-442 A	27204 / 12008	Linde 1092	1.1~~Ref-14~~	0.190	0.970	215.7	-20	~~0.0216~~0.0245	~~0.1804~~0.1948	~~38.9~~42	~~38.9~~42.0	~~57.8~~64	≤270	Yes
Upper Shell Long. Weld 1-442 B	27204 / 12008	Linde 1092	1.1~~Ref-14~~	0.190	0.970	215.7	-20	~~0.0127~~0.0149	~~0.1287~~0.1428	~~27.8~~30.8	~~27.8~~30.8	~~35.5~~42	≤270	Yes
Upper Shell Long. Weld 1-442 C	27204 / 12008	Linde 1092	1.1~~Ref-14~~	0.190	0.970	215.7	-20	~~0.0257~~0.0306	~~0.2004~~0.2222	~~43.2~~47.9	~~43.2~~47.9	~~66.5~~76	≤270	Yes
Upper Shell to Intermediate Shell Circumferential Weld 8-442	13253	Linde 1092	1.1~~Ref-14~~	0.25	0.730	197.5	-56	~~0.0286~~0.0341	~~0.2136~~0.2365	~~42.2~~46.7	~~54.2~~57.8	~~40.4~~48	≤300	Yes

Practice of altering physical stress test data "using credible surveillance data" without explaining where the data came from is highly irregular. Note most values shift 2%, except limiting welds shifts 15% to 17%, a clear indication that PG&E is selectively altering only the data that poses a compliance issue.

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TIME-LIMITED AGING ANALYSES

Table 4.2-4 DCPP Unit 1 Vessel RT_{PTS} at 54 EFPY(i)

Material Description			~~Sourc~~e R.G 1.99, Rev.2 Position	Chemical Composition		Chemistry Factors °F	Initial RT_{NDT} °F	EOLE Fluence 10^{19} n/cm² E>1.0 MeV	Fluence Factor	Δ~~RT_{PTS}~~ RT_{NDT} °F	Margin °F	RT_{PTS} °F	Screening Criteria °F	Extended Beltline Region
Location	Heat No.	Type		Cu Wt%	Ni Wt%									
Intermediate Shell Long. Welds 2-442A, B	27204	Linde 1092	1.1~~Ref-14~~	0.203	1.018	226.8	-56	~~1.56~~1.49	~~1.1229~~1.1104	~~254.7~~251.8	65.5	~~264.2~~261	≤270	No
→Using credible surveillance data	27204	Linde 1092	2.1	0.203	1.018	214.1	-56	1.49	1.1104	237.7	44.0	226	≤270	No
Intermediate Shell Long. Weld 2-442C	27204	Linde 1092	1.1~~Ref-14~~	0.203	1.018	226.8	-56	~~0.795~~0.768	~~0.9356~~0.9259	~~212.2~~210.0	65.5	~~221.7~~220	≤270	No
→Using credible surveillance data	27204	Linde 1092	2.1	0.203	1.018	214.1	-56	0.768	0.9259	198.2	44.0	186	≤270	No
Lower Shell Long. Welds 3-442A, B	27204	Linde 1092	1.1~~Ref-14~~	0.203	1.018	226.8	-56	~~1.28~~1.19	~~1.0687~~1.0485	~~242.4~~237.8	65.5	~~251.9~~247	≤270	No
→Using credible surveillance data	27204	Linde 1092	2.1	0.203	1.018	214.1	-56	1.19	1.0485	224.5	44.0	213	≤270	No
Lower Shell Long. Weld 3-442C	27204	Linde 1092	1.1~~Ref-14~~	0.203	1.018	226.8	-56	~~2.04~~2.01	~~1.1943~~1.1904	~~270.9~~270.0	65.5	~~280.4~~280	≤270	No
→Using credible surveillance data	27204	Linde 1092	2.1	0.203	1.018	214.1	-56	2.01	1.1904	254.9	44.0	243	≤270	No
Intermediate to Lower Shell Circumferential Weld 9-442	21935	Linde 1092	1.1~~Ref-14~~	0.183	0.704	172.2	-56	~~2.04~~2.01	~~1.1943~~1.1904	~~205.7~~205.0	65.5	215~~.2~~	≤300	No

Notes:

(i) Reference 39, WCAP-17315-NP

APPENDIX N (continued)

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Section 4
TIME-LIMITED AGING ANALYSES

Table 4.2-5 DCPP Unit 2 Vessel RT_{PTS} at 54 EFPY[(i)]

Material Description			R.G 1.99, Rev.2 Position ~~Source~~	Chemical Composition		Chemistry Factors °F	Initial RT_{NDT} °F	Fluence 10^{19} n/cm² E >1.0MeV	Fluence Factor	Δ~~RT_{PTS}~~ RT_{NDT} °F	Margin °F	RT_{PTS} °F	Screening Criteria °F	Extended Beltline Region
Location	Heat No.	Type		Cu Wt%	Ni Wt%									
→Using credible surveillance data	21935 / 12008	Linde 1092	2.1	0.220	0.870	204.6	-50	0.0629	0.3306	67.6	28	46	≤270	Yes
Upper Shell Long. Weld 1-201 C	21935 / 12008	Linde 1092	1.1~~Ref. 14~~	0.220	0.870	211.2	-50	~~0.0393~~0.0514	~~0.2564~~0.2971	~~54.1~~62.7	~~54.1~~56	~~58.3~~69	≤270	Yes
→Using credible surveillance data	21935 / 12008	Linde 1092	2.1	0.220	0.870	204.6	-50	0.0514	0.2971	60.8	28	39	≤270	Yes
Upper Shell to Intermediate Shell Circumferential Weld 8-201	21935	Linde 1092	1.1~~Ref. 14~~	0.183	0.704	172.2	-56	~~0.0520~~0.0677	~~0.2990~~0.3434	~~51.5~~59.1	~~61.7~~65.5	~~57.2~~69	≤300	Yes
Intermediate Shell Long. Weld 2-201A	21935 / 12008	Linde 1092	1.1~~Ref. 14~~	0.22	0.87	211.2	-50	~~1.32~~1.24	~~1.0772~~1.0599	~~227.5~~223.9	56	~~233.5~~230	≤270	No
→Using credible surveillance data	21935 / 12008	Linde 1092	2.1	0.22	0.87	204.6	-50	1.24	1.0599	216.9	28	195	≤270	No
Intermediate Shell Long. Weld 2-201B	21935 / 12008	Linde 1092	1.1~~Ref. 14~~	0.22	0.87	211.2	-50	~~1.59~~1.53	~~1.1281~~1.1176	~~238.2~~236.0	56	~~244.2~~242	≤270	No
→Using credible surveillance data	21935 / 12008	Linde 1092	2.1	0.22	0.87	204.6	-50	1.53	1.1176	228.7	28	207	≤270	No
Intermediate Shell Long. Weld 2-201C	21935 / 12008	Linde 1092	1.1~~Ref. 14~~	0.22	0.87	211.2	-50	~~1.35~~1.30	~~1.0834~~1.0730	~~228.8~~226.6	56	~~234.8~~233	≤270	No
→Using credible surveillance data	21935 / 12008	Linde 1092	2.1	0.22	0.87	204.6	-50	1.30	1.0730	219.5	28	198	≤270	No
Lower Shell Long. Weld 3-201A	33A277	Linde 124	1.1~~Ref. 14~~	0.258	0.165	126.3	-56	~~1.33~~1.28	~~1.0793~~1.0687	~~136.3~~135.0	65.5	~~145.8~~144	≤270	No
→Using credible surveillance data	33A277	Linde 124	2.1	0.258	0.165	115.9	-56	1.28	1.0687	123.9	44.0	112	≤270	No

The 2011 Annual License Renewal Application (LRA) Update used unknown sources to revise not only projected life of reactors (increased 68%) based on fluence (projected radiation damage) but also extrapolated from the lower projected radiation damage to change the physical stress test results laid out in these tables. The actual calculation methods are not shown, and whether data from similar nuclear reactors was substituted at this point is not explained (only later confirmed verbally). All referenced studies and footnotes do not appear at the end of the docketed version of the DCL-11-136 document as posted on ADAMS. The footnote below (WCAP-17315-NP) is the only mention of how such data revisions may be justified.

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Section 4
TIME-LIMITED AGING ANALYSES

Table 4.2-5 DCPP Unit 2 Vessel RT_{PTS} at 54 EFPY[(i)]

Material Description			R.G 1.99, Rev.2 Position ~~Source~~	Chemical Composition		Chemistry Factors °F	Initial RT_{NDT} °F	Fluence 10^{19} n/cm² E >1.0MeV	Fluence Factor	Δ~~RT_{PTS}~~ RT_{NDT} °F	Margin °F	RT_{PTS} °F	Screening Criteria °F	Extended Beltline Region
Location	Heat No.	Type		Cu Wt%	Ni Wt%									
Lower Shell Long. Weld 3-201B	33A277	Linde 124	1.1~~Ref. 14~~	0.258	0.165	126.3	-56	~~1.31~~1.23	~~1.0751~~1.0577	~~135.8~~133.6	65.5	~~145.3~~143	≤270	No
→Using credible surveillance data	33A277	Linde 124	2.1	0.258	0.165	115.9	-56	1.23	1.0577	122.6	44.0	111	≤270	No
Lower Shell Long. Weld 3-201C	33A277	Linde 124	1.1~~Ref. 14~~	0.258	0.165	126.3	-56	~~1.57~~1.51	~~1.1246~~1.1141	~~142.0~~140.7	65.5	~~151.6~~150	≤270	No
→Using credible surveillance data	33A277	Linde 124	2.1	0.258	0.165	115.9	-56	1.51	1.1141	129.1	44.0	117	≤270	No
Intermediate to Lower Shell Circumferential Weld 9-201	10120	Linde 0091	1.1~~Ref. 14~~	0.046	0.082	34	-56	~~2.30~~2.22	~~1.2252~~1.2161	~~41.7~~41.3	~~53.8~~53.5	39~~.4~~	≤300	No

Notes:

(i) Reference 39, WCAP-17315-NP

APPENDIX O:

2009 LER – not on NRC docket-still searching for LER that refers to a coolant system leak wherein all 5 leak detection systems were not operable at once.

APPENDIX P

(adjacent text)

PG&E response to RAI 4.2.3-1 from DCL-15-121 (Oct.2015) states that they agree that surveillance for most limiting components must be used regardless of credibility, (near bottom).

RAI 4.2.3-1

Background:

In Pacific Gas and Electric Company (PG&E) Letter DCL-11-136 (Dec. 21, 2011), the applicant provided an update of the upper shelf energy (USE) analysis for ferritic components in the reactor pressure vessels (RPVs) of Diablo Canyon, Units 1 and 2. The applicant stated that, in accordance with Regulatory Guide (RG) 1.99, Revision 2, the USE data from Unit 1 surveillance Capsule V were determined not to be credible and were, therefore, not included in the USE projections for Unit 1 RPV components represented in the Diablo Canyon RPV surveillance program for Unit 1. Instead, the applicant stated that the USE values were projected to 54 effective full power years (EFPY) of operation using USE analysis methods and criteria that are given in Position 1.2 of RG 1.99, Revision 2.

Issue:

Page No. 1.99-2 in RG 1.99, Revision 2, establishes the following regulatory discussion regarding the application of Charpy-impact data for neutron fluence-dependent RPV adjusted reference temperature calculations and USE analyses:

> *When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Regulatory Position 2.1 normally should be less than 28 °F for welds and 17 °F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fail this criterion for use in . . . [ΔRT_{NDT}] . . . shift calculations, they may be credible for determining decrease in upper-shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E185-82.*

The staff seeks further justification why all capsule data (i.e., those from the Capsule S, Y, and V Charpy-impact tests of materials representing Weld Heat 27204 in the Unit 1 RPV material surveillance program) have not been applied to the 54 EFPY USE analyses for RPV weld components in Unit 1 fabricated from the same weld heat.

Request:

Justify why all capsule data (i.e., those from the Capsule S, Y, and V Charpy-impact test specimens for Weld Heat 27204 in the Unit 1 reactor vessel material surveillance program as reported and analyzed in WCAP-15958, Rev. 0) have not been used as the basis for calculating the 54 EFPY USE values for Unit 1 RPV weld components fabricated from the same weld heat (i.e., for the USE calculations of intermediate shell axial welds 2-442 A, B and C, and lower shell axial welds 3-442, A, B, and C).

PG&E Response to RAI 4.2.3-1

PG&E amends LRA, Section 4.2.3, as shown in Enclosure 3, to state that in accordance with RG 1.99, Revision 2, the C_V USE data from Unit 1 surveillance Capsule V were determined not to be credible for determination of ΔRT_{NDT}, but were credible for determining the USE projections for Unit 1 RPV components represented in the DCPP RPV surveillance program for Unit 1.

RG 1.99, Revision 2 defines two methods that can be used to predict the decrease in USE due to irradiation. The method to be used depends on the availability of credible surveillance capsule data. For vessel beltline materials that are not in the surveillance program or are not credible, the Charpy USE (Position 1.2) is assumed to decrease as a function of fluence and copper content, as indicated in RG 1.99, Revision 2.

IMPORTANT: PG&E indicates their read of RG1.99 to mean Position 1.2 can be used only if stress test data can be deemed "not credible".

When two or more credible surveillance data sets become available from the reactor vessel, they may be used to determine the Charpy USE of the surveillance materials.

The surveillance data are then used in conjunction with Figure 2 of the RG to predict the decrease in USE (Position 2.2) of the reactor vessel materials due to irradiation. If the end-of-license and/or end-of-license extended USE values calculated using Position 2.2 are most limiting, then they must be used regardless of the credibility of the surveillance data.

Unit 1 USE values were projected to 54 EFPY of operation using Position 1.2 results because they were more limiting than the Position 2.2 results.