MEMORANDUM

Date: 28 May 2025

From: Andrew Kadak
Mark Kirk

To: Diablo Canyon Independent Safety Committee (DCISC)

Subj: Responses to Six Additional Questions from Mr. Severance

On 23 February 2025 Mr. Bruce Severance sent a document to the DCISC titled as follows:

> **Critical Variables Regarding Diablo Canyon Unit 1 Embrittlement Are Not Factored by the Current 2011 PTS Evaluation**
> An Addendum to a recent 40-page Comment to the DCISC by Bruce Severance, February 23, 2025

This document highlighted six questions of importance to Mr. Severance. We have prepared this document for the DCISC to provide our perspectives on each of Mr. Severance's questions. A summary of each question is followed by our reply, which appears in blue text. Mr. Severance's complete question is not copied here, it can be found in his original document.

None of these questions influence or alter the conclusions of the reports provided in 2024 by these consultants to the DCISC concerning the embrittlement status of the Unit 1 reactor at Diablo Canyon.

Should you have any additional questions or require clarifications please do not hesitation to contact us.

Question 1

Mr. Severance asked the following question:

> **Why are there very different outcomes when comparing the Palisades and DCPP-1 Sum of Squares Procedure for Reducing Chemistry Factors and Predicted Embrittlement at end-of-license when both are based upon consideration of the same data sets?**

Reply

Mr. Severance identifies what he believes to be an inconsistency in the analysis of surveillance data from weld wire heat 27204 as performed by the Palisades plant[1] when compared with that performed by Diablo Canyon Unit 1[2]. For weld wire heat 27204, Mr. Severance asserts the following:

- In the Diablo Canyon Unit 1 analysis, the estimate of RT_{PTS} for this weld reduces by 37 °F (from 280 °F to 243 °F) when credible surveillance data are considered.
- However, in the Palisades analysis, the estimate of RT_{PTS} for this weld reduces by only 7.3 °F (from 281 °F to 273.7 °F) when credible surveillance data are considered.

Mr. Severance correctly observes that the surveillance data for weld wire heat 27204 considered by both plants is identical, so he questions how the reduction of RT_{PTS} for consideration of credible surveillance data could be so different between the two plants. As described below, his perception of an inconsistency results from an incorrect reading of the tables in the Palisades report.

For Diablo Canyon Unit 1, the 37 °F reduction in RT_{PTS}, from 280 °F to 243 °F, is correctly noted by Mr. Severance. These values trace to Table 6.1-2 from the citation in footnote 2, which appears on page 3 of this document. These values appear in the last column of the two rows highlighted by the red box. The only difference between these two rows is the use of the NRC's unaltered RG1.99 formula in the upper row versus the use of this formula as modified to account of credible data in the lower row. As noted in the title of the table, the fluence associated with all entries in the table is that associated with 54 EFPY (nominally 60-years) of operation. For the 27204 weld this corresponds to a fluence of 2.01×10^{19} n/cm^2 on the inner-diameter surface of the Diablo Canyon Unit 1 reactor.

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1 Revised Pressurized Thermal Shock Evaluation for the Palisades Reactor Pressure Vessel, Structural Integrity Associates, Nov. 12, 2010, SIA Report#10000915.401, NRC# ML110060692 (sic – should be ML110060692).

2 B.A. Rosier, "Diablo Canyon Units 1 and 2 Pressurized Thermal Shock and Upper-Shelf Energy Evaluations," WCAP-17315-NP Revision 0, July 2011, Westinghouse Electric Company.

Table 6.1-2 RT_{PTS} Calculations for the Diablo Canyon Unit 1 Reactor Vessel Beltline and Extended Beltline Materials at 54 EFPY[a]

Reactor Vessel Material	R.G. 1.99, Rev. 2 Position	CF[b] (°F)	Fluence[c] ($x10^{19}$ n/cm^2, E > 1.0 MeV)	FF[c]	IRT_{NDT}[d] (°F)	ΔRT_{NDT} (°F)	σ_U[d] (°F)	σ_Δ[f] (°F)	Margin (°F)	RT_{PTS} (°F)
Beltline Materials										
IS Plate B4106-1	1.1	85.3	2.02	1.1917	-10	101.7	0	17	34	126
IS Plate B4106-2	1.1	81.0	2.02	1.1917	-3	96.5	0	17	34	128
IS Plate B4106-3	1.1	55.2	2.02	1.1917	30	65.8	17[e]	17	48.1	144
→ Using non-credible surveillance data	2.1	37.4	2.02	1.1917	30	44.6	17[e]	17	48.1	123
LS Plate B4107-1	1.1	89.8	2.01	1.1904	15	106.9	0	17	34	156
LS Plate B4107-2	1.1	82.2	2.01	1.1904	20	97.9	0	17	34	152
LS Plate B4107-3	1.1	81.4	2.01	1.1904	-22	96.9	0	17	34	109
IS Longitudinal Weld 2-442A & B	1.1	226.8	1.49	1.1104	-56[e]	251.8	17[e]	28	65.5	261
→ Using credible surveillance data	2.1	214.1	1.49	1.1104	-56[e]	237.7	17[e]	14	44.0	226
IS Longitudinal Weld 2-442C	1.1	226.8	0.768	0.9259	-56[e]	210.0	17[e]	28	65.5	220
→ Using credible surveillance data	2.1	214.1	0.768	0.9259	-56[e]	198.2	17[e]	14	44.0	186
LS Longitudinal Weld 3-442A & B	1.1	226.8	1.19	1.0485	-56[e]	237.8	17[e]	28	65.5	247
→ Using credible surveillance data	2.1	214.1	1.19	1.0485	-56[e]	224.5	17[e]	14	44.0	213
LS Longitudinal Weld 3-442C	1.1	226.8	2.01	1.1904	-56[e]	270.0	17[e]	28	65.5	280
→ Using credible surveillance data	2.1	214.1	2.01	1.1904	-56[e]	254.9	17[e]	14	44.0	**243**
IS to LS Circ. Weld 9-442	1.1	172.2	2.01	1.1904	-56[e]	205.0	17[e]	28	65.5	215

For the Palisades Plant, the 7.3 °F reduction in RT_{PTS}, from 281 °F to 273.7 °F, traces to two tables from the citation in footnote 1. These two tables, designated Table 1 and Table 10, are copied on page 4 of this document.

- The 281 °F value appears in the last column of the row highlighted by the red box In Table 1. This value is estimated by use of NRC's unaltered RG1.99 formula, as evidenced by the Chemistry Factor (CF) of 226.8 °F that appears in the highlighted row. As noted in the title of Table 1, the values in this table correspond to the **original (40-year)** license expiration date for Palisades. For the 27204 weld this corresponds to a fluence of 2.061×10^{19} n/cm^2 on the inner-diameter surface of the Palisades reactor.
- The 273.8 °F value appears in the last column of the row highlighted by the red box In Table 10. This value is estimated by use of NRC's RG1.99 formula after adjustment to account for credible data, as evidenced by the Chemistry Factor (CF) of 216.13 °F that appears in the highlighted row. As noted in the title of Table 1, the values in this table correspond to the **extended (60-year)** license expiration date for Palisades. For the 27204 weld this corresponds to a fluence of 3.429×10^{19} n/cm^2 on the inner-diameter surface of the Palisades reactor.

In summary, the two RT_{PTS} values quoted by Mr. Severance for Diablo Canyon Unit 1 differ only in the use (or not) of credible surveillance data. Conversely, the two RT_{PTS} values quoted by Mr. Severance for Palisades differ in both the use (or not) of credible surveillance data **and also** in the fluence associated with the two RT_{PTS} estimates. The

Table 1: Palisades Vessel Beltline Material Properties on Operating License Expiration Date (3/24/11) (from Ref. 2)

RPV Material	Heat No.	Cu%	Ni%	CF (°F)	Surface Fluence (E19)	FF	$RT_{NDT}(U)$ (°F)	RT_{NDT} Shift (°F)	Margin (°F)	RT_{PTS} (°F)
Axial Weld 2-112A/B/C	W5214	0.213	1.01	231.08	1.492	1.111	-56	256.7	65.5	266
Axial Weld 3-112A/B/C	W5214	0.213	1.01	231.08	1.492	1.111	-56	256.7	65.5	266
	34B009	0.192	0.98	217.7	1.492	1.111	-56	241.8	65.5	251
Circ Weld 9-112	27204	0.203	1.018	226.8	2.061	1.197	-56	271.5	65.5	281
Plate D-3803-1	C-1279	0.24	0.50	153.3	2.061	1.197	-5	183.5	17	195
Plate D-3803-2	A-0313	0.24	0.52	160.4	2.061	1.197	-30	192.0	34	196
Plate D-3803-3	C-1279	0.24	0.50	153.3	2.061	1.197	-5	183.5	17	195
Plate D-3804-1	C-1308A	0.19	0.48	128.8	2.061	1.197	0	154.2	34	188
Plate D-3804-2	C-1308B	0.19	0.50	131	2.061	1.197	-30	156.8	34	161
Plate D-3804-3	B-5294	0.12	0.55	82	2.061	1.197	-25	98.2	34	107

Table 10: Palisades Vessel Beltline Material Properties on Extended Operating License Expiration Date (3/24/31)

RPV Material	Heat No.	Cu%	Ni%	CF (°F)	Surface Fluence (E19)	FF	$RT_{NDT}(U)$ (°F)	RT_{NDT} Shift (°F)	Margin (°F)	RT_{PTS} (°F)
Axial Weld 2-112A/B/C	W5214	0.213	1.007	227.74*	2.161	1.209	-56	275.4	65.5	284.9
Axial Weld 3-112A/B/C	W5214	0.213	1.007	227.74*	2.161	1.209	-56	275.4	65.5	284.9
	34B009	0.192	0.98	217.7	2.161	1.209	-56	263.2	65.5	272.7
Circ Weld 9-112	27204	0.203	1.018	216.13*	3.429	1.322	-56	285.7	44	273.7
Plate D-3803-1	C-1279	0.24	0.50	157.5	3.429	1.322	-5	209.5	34	238.5
Plate D-3803-2	A-0313	0.24	0.52	160.4	3.429	1.322	-30	212.0	34	216.0
Plate D-3803-3	C-1279	0.24	0.50	157.5	3.429	1.322	-5	209.5	34	238.5
Plate D-3804-1	C-1308A	0.19	0.48	128.8	3.429	1.322	0	170.3	34	204.3
Plate D-3804-2	C-1308B	0.19	0.50	131	3.429	1.322	-30	173.2	34	177.2
Plate D-3804-3	B-5294	0.12	0.55	82	3.429	1.322	-25	108.4	34	117.4

comparison made by Mr. Severance is therefore not an appropriate “apples to apples” comparison between the two plants.

To compare the assessment of surveillance data from weld wire heat 27204 performed by Diablo Canyon to that performed by Palisades it is appropriate to compare the chemistry factors both before and after adjustment for credible surveillance data. The table below provides this comparison, using values extracted from the tables on pages 3 and 4. The small difference in chemistry factor after surveillance data adjustment between the two plants can be attributed to differences in the plant operating temperatures, which as noted in the citation that appears in footnote 5 is a minor factor in the chemistry factor analysis.

Plant Name	NRC “Table” Chemistry Factor [°F]	Chemistry Factor Considering Credible Surveillance Data [°F]
Diablo Canyon Unit 1	226.8	214.1
Palisades	226.8	216.13

<u>Question 2</u>

Mr. Severance asked the following question:

> **Have the 2011 Westinghouse PTS calculations followed the NRC guidance that RT_{PTS} embrittlement values must be calculated from the "tip of flaws" detected using ultrasonic testing (UT)? How can the DCISC assert their confidence in the 2011 Westinghouse PTS Evaluation which provides the basis for the current 2023 License Renewal Application 2023LRA], when that report's RT_{PTS} value (embrittlement at EOL) has been calculated without consideration of RPV flaws as required by Section 50 Appendix H and the Generic Letters? Such cracks and voids that can be attributed to radiation damage are only detected through ultrasonic testing**

<u>Reply</u>

Mr. Severance's comment that RT_{PTS} values must be calculated at the tips of flaws is expressed in more detail in a previous communication between Mr. Severance and the DCISC.[3] In his document of 1 February 2025 Mr. Severance quoted a section of the 1987 NRC regulatory analysis for Revision 2 of Regulatory Guide 1.99[4]. Mr. Severance's contention (see page 26 of the 1 February 2025 document) that "*the NRC's 1987 analysis … suggests that RT_{PTS} values must be calculated from the tip of vessel wall flaws in order to accurately reflect actual RT_{PTS} values and their impact on integrity*" is an incorrect interpretation of the words in the 1987 NRC document and, more importantly, is at odds with the actual requirements of the PTS rule (10 CFR 50.61). Further details are provided below:

- <u>1987 NRC document</u>: This document, as quoted by Mr. Severance, states "*Another application of Regulatory Guide 1.99 is in the analysis of flaws found by inservice inspection of the reactor vessel beltline.*" This statement does not refer to a PTS evaluation but, instead, refers to an "Analytical Evaluation of Flaws" performed in accordance with Appendix A to Section XI of the ASME Code. Appendix A uses information on flaw size determined by UT inspection, information on vessel loading, and information on fracture toughness as affected by embrittlement to assess the acceptability for continued service of flaws found during plant operation. These evaluations are performed for normal operating loads and conditions; they are not an evaluation postulated accident conditions such as PTS. The only documents that

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3 See document dated 1 February 2025 from Mr. Severance entitled "*NRC Regulations Clearly Require Accurate Reporting of Embrittlement at End-of-License: The Clear Purpose of the Capsule Reports is to Provide Stress Test Data to Confirm Calculated Predictions Are Not Factored by the Current 2011 PTS Evaluation.*"

4 Regulatory Analysis, Revision 2 to Regulatory Guide 1.99, Radiation Embrittlement of Reactor Vessel Materials, 20 November 1987, United States Nuclear Regulatory Commission, ADAMS ML102310298

describe PTS evaluations in the United States are those of the NRC: 10 CFR 50.61 and 10 CFR 50.61a.

- 10 CFR 50.61 Requirements: The NRC's PTS requirement that compares a RT_{PTS} value for a plant to a screening criteria given in 10 CFR 50.61 was established based in part on a probabilistic analysis of a population of flaws assumed to exist within the vessel. This analysis was performed by the NRC during the development of 10 CFR 50.61[5]. This approach implicitly incorporated flaw size into the value of the PTS screening criteria making vessel-specific information on flaw size unnecessary. No part of the 10 CFR 50.61 requirements for the estimation of a RT_{PTS} value involves information concerning flaw location or flaw size.

In summary, Mr. Severance has misinterpreted the NRC's PTS requirements, which are expressed in 10 CFR 50.61. That law requires comparison of a RT_{PTS} value for a plant to a screening criterion established by that law. PG&E has made this calculation and performed this comparison as it is required to do. No part of the 10 CFR 50.61 requirements for the estimation of a RT_{PTS} value involves information concerning flaw location or flaw size.

[5] SECY-82-465, Letter from W.J. Dircks to the NRC Commissioners concerning Pressurized Thermal Shock, 23 November 1982, ADAMS ML16232A574.

Question 3

Mr. Severance asked the following question:

> **How does the current DCPP-1 PTS report assess the NRC concern about weld seam chemistry variability which increases the uncertainties regarding the structural integrity of a weld from one end of the weld seam to the other? The "Generic Letters" *GL92-01 Rev.1,Supplement , Wichman 1995, text provided to DCISC] raises the concern that due to shortcomings in the dual wire feed weld process and substandard quality process controls in the 1965-1970 vintage RPV manufacturing processes, the dual-wire feed weld process produced high variability in the nickel and copper impurities in critical welds. Has PG&E ever performed chemical analysis of weld chemistry along the full length of the limiting axial weld (weld#3-442C)? If you have not factored the uncertainties regarding this weld variability and tested for chemical composition through every foot of the critical weld seams, how can you claim to have any certainty in the results of the 2011 Westinghouse PTS Evaluation? This report doesn't appear to account for this critical variable identified by the NRC as early as 1995.**

Reply

The current Diablo Canyon Unit 1 PTS analysis complies with all applicable NRC requirements regarding chemistry analysis and chemistry variability. Mr. Severance is correct to note that the structural welds in many early construction plants exhibited larger than expected variability in the percentages of elements primarily responsible for embrittlement sensitivity (i.e., copper and nickel), this being caused by the welding processes in common and accepted use at the time. As Mr. Severance notes, the NRC recognized this situation in the early 1990s and communicated this information to the industry via Generic Letter 92-01 (GL92-01). This initiated a multi-year effort on the part of both the NRC and the industry to collect and analyze data on weld chemistry. This effort produced NRC recommended practices for weld chemistry analysis that were summarized in a 1998 NRC presentation[6]. The NRC developed these practices considering knowledge of the larger than expected chemistry variability in early plants. The NRC deemed the practices expressed in their 1998 presentation, which have since become *de facto* requirements following by all licensees (including PG&E), adequate to address this variability in consideration of (1) the "Margin" term that accounts for material property variability already required for use in a regulatory guide 1.99 analysis and in consideration of (2) the conservatisms inherent to the screening criteria for PTS. Neither these practices nor any other NRC regulation requires the "*chemical analysis … along the full length of the limiting axial weld*" suggested by Mr. Severance.

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[6] K.R. Wichman, M.A. Mitchell, and A.L. Hiser, "Generic Letter 92-01 and RPV Integrity Assessment: Status, Schedule, and Issues," Presentation to NRC/Industry Workshop on RPV Integrity Issues, February 12, 1998, ADAMS ML 110070570, https://www.nrc.gov/docs/ML1100/ML110070570.pdf.

Question 4

Mr. Severance asked the following question:

> **The 2006 37-month extension of the DCPP-1 license from September of 2021 to November of 2024, as documented in an NRC letter dated July 17, 2006 [NRC Tac#MC8206 & 8207, ML0160660220[7], pdf pg.11], clearly states that the 2021 end-of-license (EOL) date was based on a fluence for 32EFPY (40-years), and that the extension to November 2024 was based on 35.2 EFPY. Therefore, a 20-year extension based on 54EFPY of accumulated fluence (60-year) should extend 20 years from the prior 32EFPY date: the Sept. 2021 32 EFPY date, not Nov. 2024 35.2 EFPY date. This appears to be an obvious error in the 2011 Westinghouse PTS Evaluation that grants an additional three years of compliance regardless of other assumptions because it changes the fundamental fluence calculations based upon an incorrect correlation of EFPY to calendar dates. If the 2011 Westinghouse report does allow a 20-year extension, it should be a 20-year extension to Sept. 2041 and not Nov. 2024 as stated. This appears to be an oversight which the DCISC has missed despite my previous attempts to flag the error.**

Reply

This question reflects a misunderstanding of the correspondence between the date on which a plant's license expires ("end of license," or EOL) and estimates of EFPY (effective full-power years). The plant's EOL is a calendar date that appears in the license. Once established this date can be changed only by license amendment (see, for example the 2006 document referenced by Mr. Severance[8] in which the NRC granted an extension of slightly more than 3-years to recover the amount of time spent in low-power testing of the Unit 1 reactor prior to initial start-up. In the NRC's Safety Evaluation of the license extension appearing in the 2006 document the NRC stated the following (see pdf page 8):

> *"Extend the expiration date for DCPP-1 Operating License No. DPR-80 from September 22, 2021, to November 2, 2024. This revised expiration date equates to 35.2 effective full power years (EFPY) of power operations."*

It should be noted that only the calendar date, 2 November 2024, is changed in the Unit 1 license. The Safety Evaluation does not become part of the license, so the rationale by which the new EOL date was established is not part of the license. Nevertheless, it remains true that for a plant to legally operate to the EOL date it must continue to

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[7] There is a typographical error in Mr. Severance's citation. The correct accession number for this document is ML061660220.

[8] Letter from NRC to Diablo Canyon (Mr. Wang to Mr. Keenan) concerning issuance of amendments for recovery of low power testing time, 17 July 2006, ADAMS ML061660220

demonstrate that it remains in compliance with all applicable regulatory requirements. In the context of the current discussion of Unit 1, this means that PG&E must show compliance with the PTS rule (10 CFR 50.61) and with upper shelf energy requirements (10 CFR 50 Appendix G). Unit 1 complies with these requirements as demonstrated by the many documents reviewed and confirmed by the report concerning Unit 1 submitted by these consultants to the DCISC[9].

It should also be noted that there is no absolute correspondence between the initial 40-year period of licensed operations and 32 EFPY, as Mr. Severance's statements appear to assume. 32 EFPY corresponds to a capacity factor of 80% (32÷40=0.8). An 80% capacity factor is often assumed as a nominal value for early construction plants. The statement that Unit 1 is projected reach 35.2 EFPY by its amended EOL date (2 November 2024) reflects PG&E's estimate, made in 2006, that a higher overall capacity factor would be achievable.

[9] https://www.dcisc.org/download/library/topical-library/report-part-2-evaluation-of-diablo-canyon-unit-1-embrittlement-76.pdf

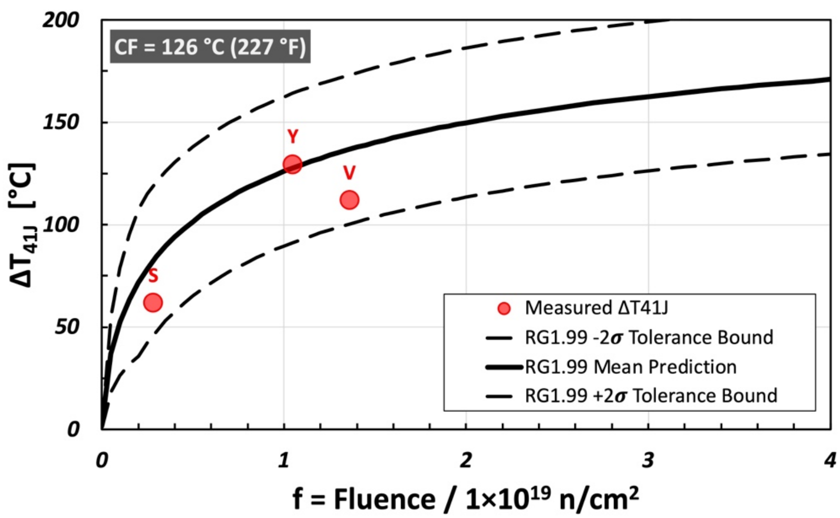
Question 5

Mr. Severance asked the following question:

> **The Capsule V data was stated to be in “good agreement” with RG1.99 predictions when this did not appear to be the case based on prior publications and reporting: While the Capsule V Report states that surveillance data was in “good agreement” with RG1.99 R.2 predictions, the measured RT_{NDT} value was 135°F, with a ΔRT_{NDT} shift of 201 °F. It is important to note that shift would be 191.5°F if same $RT_{NDT(u)}$ was used as in other reports, and that the $RT_{NDT(u)}$ appears to have been changed from -56F to -67F to bring it the report RT_{NDT} values more into line with RG1.99 predictions.**

Reply

Assessment of the agreement, or lack thereof, between plant-specific surveillance data and the NRC’s predictive model for ΔT_{41J} that appears in Revision 2 of Regulatory Guide 1.99 is performed by comparing the quantity measured in surveillance (ΔT_{41J}) to the NRC’s predictions of that quantity. As explained in the Part 1 report to the DCISC[10], delta (Δ) values of T_{41J} are the difference between the Charpy transition temperature at 41 Joules absorbed energy (T_{41J}) measured after irradiation and that same quantity measured before irradiation. The red circles on the graph below show the values of ΔT_{41J} measured as part of the Unit 1 surveillance program (labels indicate the capsule ID). These are compared to the NRC’s prediction using RG1.99 R2; both the mean prediction (**solid black curve**) and the ±2σ tolerance bounds (**dashed black curves**) are shown. All three measurements, including the measurement from Capsule V, lie well within the ±2σ tolerance bounds from the NRC’s mean prediction. The characterization of these data as being in “good agreement” with the model is therefore appropriate.



10 https://www.dcisc.org/download/library/topical-library/report-part-1-addressing-public-concerns-75.pdf

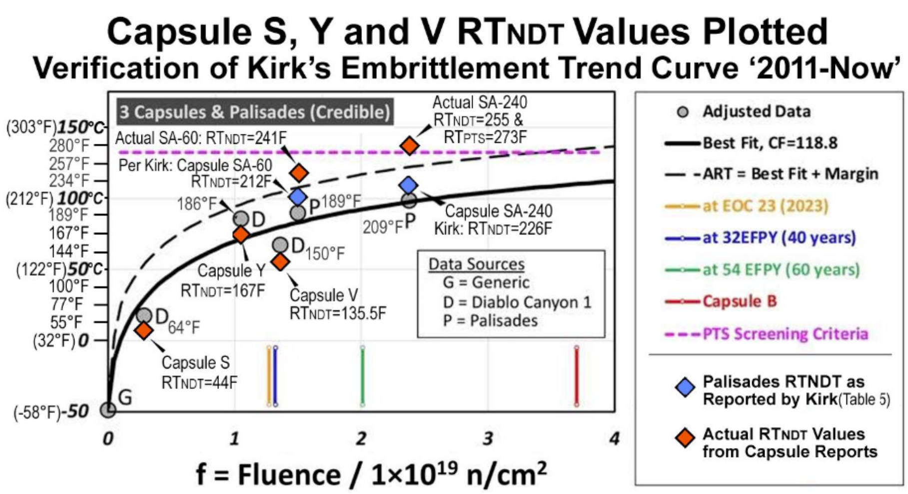
Question 6

Mr. Severance asked the following question:

> **Has the DCISC noted that Dr. Kirk's Embrittlement Trend Curves (ETCs) as indicated in the Kirk Report Part 2, Figure 2 (pg. 28) appears to misrepresent actual RT_{NDT} values by as much as 46°F to 52°F?**

Reply

This question is expressed in more detail in a previous communication between Mr. Severance and the DCISC.[11] On page 30 of his document of 1 February 2025 Mr. Severance provides the figure that appears below, which he uses to call into question the data used by Kirk in the Part 2 report (see Figure 2, bottom graphic panel of that report) in performing his evaluation of PG&E's estimation of RT_{PTS}. Mr. Severance observes that the blue and red diamonds, which he added to Kirk's figure, do not agree with Kirk's data (grey circles) for Unit 1 Capsules S, Y, and V and with data from Palisades Capsules SA-60 and SA-240. We have been able to ascertain the reason for most of the "discrepancies" noted by Mr. Serverance. These reasons are described in the text following the figure.



(1) The blue diamonds are measured values of T_{41J} that Mr. Severance took from Table 5 of the Part 2 report. He has labeled these values as "RT_{NDT}," which is incorrect; T_{41J} and RT_{NDT} are not the same. T_{41J} is a measured value of Charpy transition temperature. Conversely, RT_{NDT} is determined by adding the unirradiated value of RT_{NDT} (as reported in the Part 2 report, Diablo Canyon uses a generic value of

[11] See document dated 1 February 2025 from Mr. Severance entitled "*NRC Regulations Clearly Require Accurate Reporting of Embrittlement at End-of-License: The Clear Purpose of the Capsule Reports is to Provide Stress Test Data to Confirm Calculated Predictions Are Not Factored by the Current 2011 PTS Evaluation.*"

unirradiated RT_{NDT} of -48.9 °C) to the ΔT_{41J} measurement from capsule testing after adjustment for temperature and chemistry differences[12]. An estimation of RT_{NDT} following the NRC's procedures result in the grey circles that appear on Figure 2 of the Part 2 report.

(2) Likewise, the red diamonds plotted for Unit 1 Capsules S, Y, and V are measured values of T_{41J} taken from the respective surveillance capsule reports. As just explained in (1), values of T_{41J} are not the same as values of RT_{NDT}.

(3) We have been unable to determine the source for the red diamond values of RT_{NDT} on this figure that Mr. Severance claims are associated with the Palisades capsules SA-60 and SA-240 (241 °F and 255°F, respectively). We have reviewed both capsule reports[13,14] as well as a report on a PTS evaluation of the Palisades[15] plant quoted by Mr. Severance as his Citation #50 and have been unable to determine the source of these values.

(4) The RT_{PTS} value of 273 °F that Mr. Severance associates with the red diamond for Palisades Capsule SA-240 appears in a PTS evaluation for Palisades (footnote 19) and in a table included in Mr. Severance's document (see Table 10 on page 31) that is copied from the Palisades PTS report (see again footnote 15). Placing a calculated RT_{PTS} value for Palisades on a data plot for Diablo Canyon Unit 1 is inappropriate. While the plants share sister data for weld wire heat 27204, the RT_{PTS} values themselves are plant-specific and cannot be compared between plants. Additionally, Table 10 on Mr. Severance's page 31 indicates that the RT_{PTS} value of 273 °F corresponds to a fluence factor of 1.322 or, equivalently, a fluence of 3.45×10^{19} n/cm^2. However, on the plot copied on the previous page the red diamond for Capsule SA-240 appears at a fluence of 2.39×10^{19} n/cm^2.

In summary, there is no misrepresentation of the RT_{NDT} value for Diablo Canyon Unit 1 in any of Kirk's reports to the DCISC. The discrepancies arise mostly because Mr. Severance is comparing T_{41J} (Charpy transition temperature) values to RT_{NDT}. As explained here, RT_{NDT} and T_{41J} are not the same. Other sources of discrepancy are Mr. Severances' comparison of RT_{PTS} estimates for the Palisades plant to T_{41J} measurements, which is also inappropriate.

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[12] Chemistry and temperature adjustments are typically small. Section 2.4 of the Part 1 report to the DCISC describes the NRC's procedure to determine these adjustments.

[13] DeVan, M.J., "Test Results of Capsule SA-240-1 Consumers Energy Palisades Nuclear Plant," Babcock and Wilcox, Report BAW-2398, May 2001.

[14] DeVan, M.J., "Test Results of Capsule SA-60-1 Consumers Energy Palisades Nuclear Plant," Babcock and Wilcox, Report BAW-2341 Revision 2, May 2001.

[15] "Revised Pressurized Thermal Shock Evaluation for the Palisades Reactor Pressure Vessel," Structural Integrity Associates Report No. 1000915.410, 12 November 2010, NRC ADAMS ML110060694.