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DIABLO CANYON INDEPENDENT SAFETY COMMITTEE

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VIA EMAIL ONLY
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August 6, 2024

Dear Dr. Macdonald:

Thank you for your prompt reply in your letter on July 19 to our letter of June 28, 2024 asking certain questions regarding your June 14, 2024 report entitled: "Technical Evaluation of Reports by Dr. Mark Kirk Regarding the Condition of Diablo Canyon Unit 1 Pressure Vessel" (see Appendix). Your responses help us by explaining your understanding of the degree of embrittlement of the reactor pressure vessel and other issues you raise. This letter responds to your answers to our questions and to clarify our position as explained in Dr. Kirk's two reports.

1. Corrected graph of RT_{NDT} vs Fluence

Your corrected graph showing calculated bands of uncertainty and clarifies your position. You stated in your reply that the fluence uncertainty corresponds to ±14-20% (a value you got from WCAP-15958) and the ART uncertainty corresponds to ±1σ where the σ values were reported in NUREG/CR-6551. However, you continue to show a data point that is not a measurement from Palisades but is a predicted value from an NRC document published in the 1980s. You stated that the ART value associated with this point of 161 °C can be found in ADAMS ML102310298. We cannot determine the fluence value for this point, but assume it corresponds to the Palisades fluence in 2007, as stated in Table 3 of ADAMS ML102310298. Finally, we have been unable to determine why you also attribute this point to a citation called "Kirk, RG 1.99, Rev. 2." We cannot locate this citation.

The NRC's procedures require that only measured values be used in the type of evaluation shown in Figure 1. The measurements (dots) are fitted with the curves to determine the fluence at which the vessel is forecast to exceed the PTS screening criteria. We should also point out that the Chemistry Factor is estimated as either a best-fit for credible data or as an over-estimate for not credible data. The Chemistry Factor is a fitting parameter, not a physical property, as applied in this procedure. The Margin values also depend on data credibility, larger for not credible data and smaller for credible data. As your curves show, if the data is not credible, the yellow highlighted line reflects the added margin applied when assessing the plant's proximity to the PTS screening criteria. When the data is "credible," lower margins are permitted by the NRC as shown by the two black curves. Thus, if only measured values are used in the analysis (1 datum before irradiation and 5 data after irradiation from surveillance capsules), the PTS screening criteria will be reached at about $3x10^{19}$ n/cm^2, which is well beyond fluence of $2x10^{19}$ n/cm^2 associated with 60 years of operation.

You also point out that one data point shows a lower value of ART at a higher fluence than the previous data point, but the difference between these values is within the uncertainty bounds that you calculated.

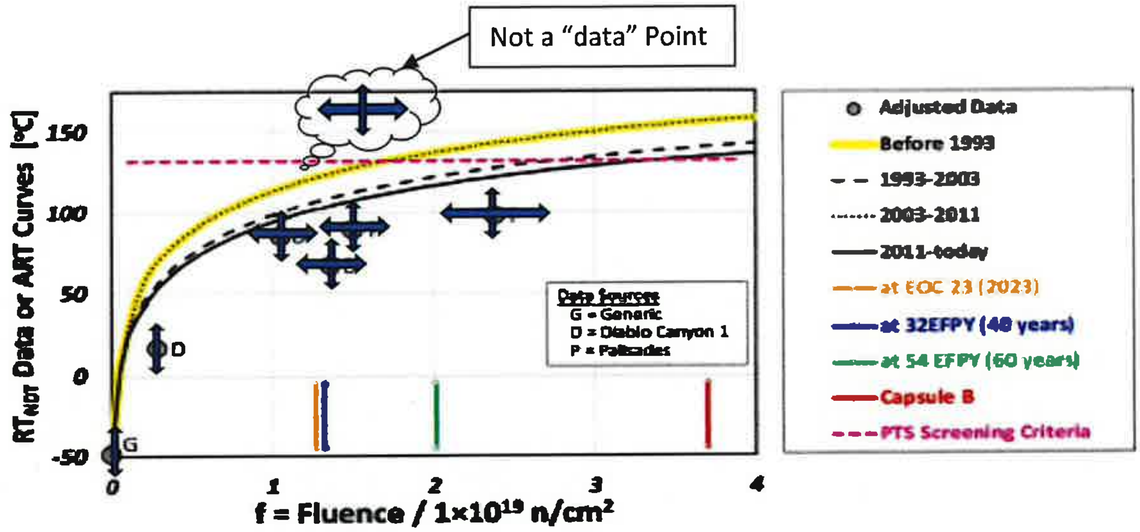

Figure 1. Corrected Figure 1 from June 14 Technical Report.

2. Extent of Embrittlement Methodology

In your letter you state that this methodology is yours alone and has not been published in any journal or peer reviewed publication. You also state that there are no "acceptance" criteria established for your methodology as it relates to Pressurized Thermal Shock. You rely on the "mathematical logic" in the relationship between Charpy absorbed fracture energy versus temperature curves to propose a "Reference Temperature Point of Inflection" $RT_{PoI.}$ You suggest that RT_{PoI} is a better metric than a reference temperature based on T_{41J}, which is both the industry standard and the NRC's requirement, for making embrittlement assessments.

You also suggested that Dr. Kirk perform his analysis using the $RT_{PoI..}$ We have performed two analyses to assess the RT_{PoI} approach. The upper pair of graphs shown below compare both the traditional (T_{41J}) and proposed (RT_{PoI}) Charpy embrittlement metrics to the true fracture toughness transition temperature (T_0), which any Charpy metric seeks to estimate (data used here are from RPV steels[1]). The predictive capability of both T_{41J} and RT_{PoI} are similar, with the latter proposal offering no advantage over the traditional (and required) approach.

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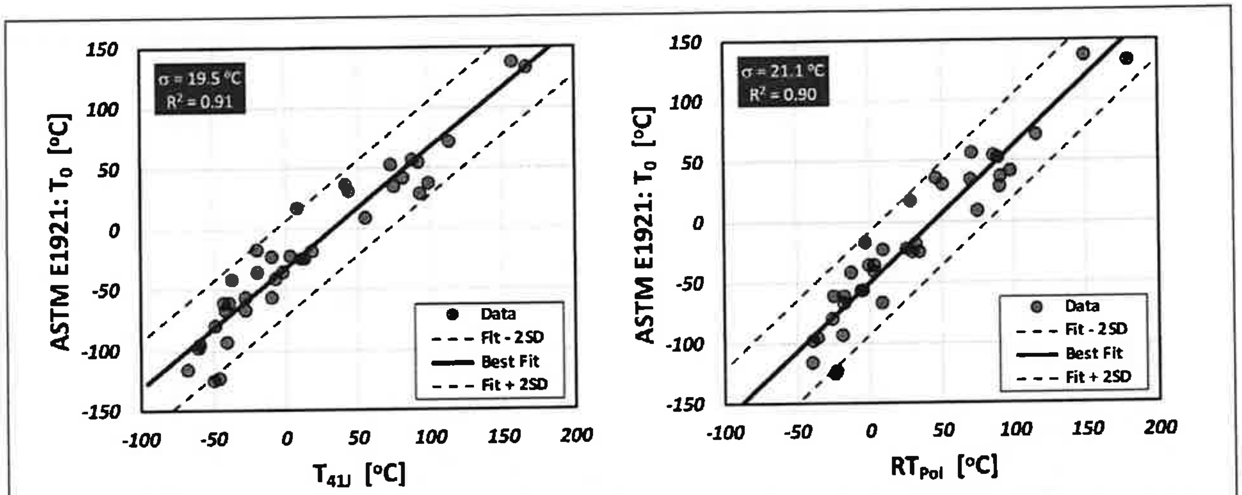
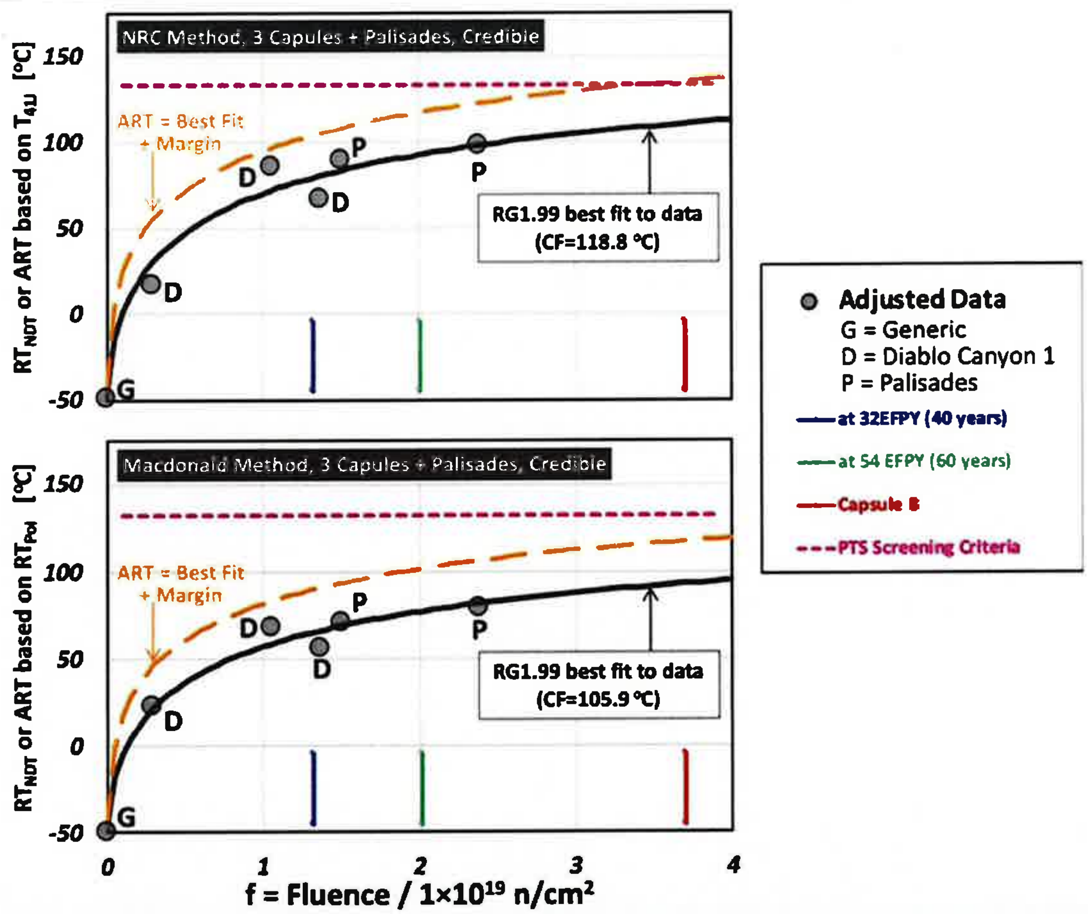
1 Data used here were reported in the following: M.A. EricksonKirk, M.T. EricksonKirk, S. Rosinski, and J. Spanner, "A Comparison of the tanh and Exponential Fitting Methods for Charpy V-Notch Energy Data," *Journal of Pressure Vessel Technology*, June 2009, Vol. 131, https://doi.orgferr/10.1115/1.3109987.



Graphs comparing the conventional Charpy embrittlement metric (T_{41J}, X-axis of left-hand graph) and Professor Macdonald's suggested metric (RT_{PoI}, X-axis of right hand graph) to the true fracture toughness transition temperature defined by ASTM E1921 (T_0, Y-axis of both graphs).

Secondly, we have used both T_{41J} and RT_{PoI} to perform a PTS assessment of Diablo Canyon Unit 1 using a data set comprised of a generic unirradiated value of RT_{NDT}, three surveillance data from Diablo Canyon Unit 1, and two surveillance data collected on the same weld wire heat as part of the Palisades surveillance program. This pair of graphs compare the traditional/required approach on the upper plot with Professor Macdonald's recommended approach on the lower plot. On the lower plot the transition temperature shift is calculated as the increase in RT_{PoI} between the unirradiated state and the various irradiated measurements.



PTS assessment of Diablo Canyon Unit 1 using the required NRC method (top graph) as compared to Professor Macdonald's proposed method (bottom graph). In the bottom graph ART is estimated using Professor Macdonald's metric RT_{Pol} instead of T_{41J}.

For this particular set of data for the weld wire heat 27204, the increase in RT_{Pol} produced by irradiation is somewhat lower (12 °C on average over the five irradiated measurements) than the increase in T_{41J}. As illustrated on the figure, this lower degree of embrittlement forecast by the RT_{Pol} approach reduces the estimated value of the chemistry factor and increases the fluence at which Diablo Canyon Unit 1 is predicted to exceed the NRC's PTS screening criteria (this fluence is determined by the intersection of the orange ART curve with the horizontal magenta line). Nevertheless, both the NRC's approach as well as Professor Macdonald's approach reach the same conclusion: based on all currently available data, Diablo Canyon Unit 1 is not expected to exceed the NRC's PTS screening criteria until it reaches a fluence far in excess of the fluence associated with 60 years of service.

<u>Hydrogen Induced Cracking (HIC)</u>

While you provided a lengthy discussion of HIC, all of the examples provided are on reactor vessel internals, not reactor vessel plates or welds. The materials cited are mostly Inconel alloys found in tubes and bolts. You mention the Davis Besse reactor vessel head leak as an example of hydrogen induced corrosion when it was due to intergranular stress induced corrosion of the Inconel tube. However, this is not an appropriate example since the original leak was from an Inconel control rod drive housing which led to boric acid corrosion of the reactor vessel which resulted from the leak. Fundamentally, the reactor vessel material subject to these embrittlement analyses is isolated from the reactor coolant by an internal layer of cladding. The vessel internal cladding layer is required to be regularly inspected using non-destructive examination techniques and cladding defects have never been discovered at Diablo Canyon and are extremely rare in the industry. In worldwide operational experience there is no record of detection of a through-cladding defect in a pressurized water reactor, which is what would be needed to expose the ferritic RPV material to the primary system coolant.

You did not provide any evidence of hydrogen induced cracking in any US or other reactor vessel or weld.

Additionally, the Diablo Canyon Reactor vessel beltline plate material is A533, CL 1 steel, not Alloy C-750 or A508. Thus, all assertions of HIC in Diablo Canyon's pressure vessel are not valid.

3. <u>Surveillance Requirements</u>

You continue to use Table 1 which you claim comes from ASTM E185-70. This table simply does not exist in the 1970 version of the standard to which Diablo Canyon was licensed. The first time this table appears is in the 1979 version, ASTM E185-79. Thus, the Table 1 requirement for 5 surveillance capsules does not apply to Diablo Canyon Unit 1. It should be noted that at present, with Palisades weld data, Diablo Canyon has five surveillance data, three of which are at a fluence exceeding the current fluence at the inner diameter of the reactor vessel. With the withdrawal of Capsule B in 2025, data will be available to a fluence far exceeding that expected after 60 years. In aggregate, these data provide a sufficient number of specimens to establish the embrittlement status of Diablo Canyon Unit 1.

As a practical matter, data obtained from surveillance capsules typically serve as benchmarks (checks) to confirm that the degree of embrittlement of a particular vessel and weld material conforms to industry data and NRC predictive models. When the data follow expected trends, the expected trends are used, with some reduction in the margin (uncertainty term) allowed depending on the quantity of data available. If the available surveillance data exceed the expected degree of embrittlement, the predictions are increased to ensure conservatism.

4. Use of Soviet Reactor Vessel Materials as Surrogates for Diablo

In your letter you state that Kuleshova paper confirms that the RPV of the VVER reactor pressure vessel comprises of SA508 Grade 3 Class 1 low alloy steel which is comparable to western PWRs. When comparisons are made of Soviet VVER reactor vessels, it is quite clear that they are radically different from what Diablo Canyon plate (A533 B Class 1) is made of as shown on the table[2] below. Thus, they provide no useful information on embrittlement of Diablo Canyon.

Table 1
Chemical composition of pressure vessel materials

Steel type	Weight%									
	Si	Mn	P	S	Cu	Ni	Cr	Mo	C	V
15Ch2MFA base metal VVER-440	0.27 0.37	0.39 0.48	0.009-0.0375	0.012 0.018	0.09 0.17	0.19-0.27	2.52 3.00	0.64 0.71	0.13-0.18	0.25 0.31
SV-10ChMFT weld metal VVER-440	0.15 0.35	0.97 1.03	0.018-0.039	0.012-0.013	0.15 0.21	0.09 0.29	1.37 1.58	0.43-0.50	0.05-0.07	0.19-0.23
25Ch3NM base metal experimental reactor	0.44	0.49	0.024	0.018	0.10	1.02	3.03	0.40	0.23	
15Ch2NMFA base metal VVER-1000	0.17 0.37	0.3 0.6	0.029-0.036	0.012 0.013	0.07 0.08	1.1 1.22	1.8 2.1	0.55 0.7	0.13-0.18	0.10 0.12
SV-10ChGNMAA weld metal VVER-1000	0.14 0.41	0.72 0.94	0.010-0.011	0.006-0.012	0.05-0.08	1.17 1.88	1.70 1.88	0.55 0.70	0.05 0.12	0.01 -0.03
Diablo Canyon[3]	0.24	1.34	0.012	0.013	0.12	0.52	--	0.46	0.22	---

The foregoing responses were prepared by DCISC technical consultants and circulated individually to the Committee Members in accordance with established DCISC protocol. In accordance with past practice a copy of this letter with its appendix will subsequently be posted on the Committee's website at www.dcisc.org. A discussion regarding integrity of the Unit 1 reactor pressure vessel will be on the agenda for the October 9-10, 2024, public meeting to be held in Avila Beach, California. Any questions may be addressed to the undersigned at info@dcisc.org.

On behalf of the Committee, thank you for your additional comments and input into our deliberations.

Sincerely,

Robert W. Rathie
DCISC Assistant Legal Counsel

Appendix

2

Kuleshova, E. A. (2002), B.A. Gurovich, Ya.I. Shtrombakh, D.Yu. Erak, and O.V. Lavrenchuk, "Comparison of microstructural features of radiation embrittlement of VVER-440 and VVER-1000 reactor pressure vessel steels". J. Nucl. Mats. 300(2-3), 127-140 (2002).

3 Averages of beltline materials of Unit 1 base metal from DCPP FSAR Update September 2033 Rev.21

cc:
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DCISC Members and Technical Consultants

APPENDIX

LETTER FROM DIGBY MACDONALD Ph.D. TO DCISC OF JULY 19, 2024

LETTER FROM DCISC TO DR. MACDONALD OF JUNE 28, 2024

TECHNICAL EVALUATION OF REPORTS BY DR. MARK KIRK REGARDING THE CONDITION OF DIABLO CANYON UNIT 1 PRESSURE VESSEL BY DR. DIGBY MACDONALD, JUNE 14, 2024

Digby Macdonald, Ph.D
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July 19, 2024

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SUBJECT: *Response to Your Questions of June 28, 2024 Regarding My June 14, 2024 Technical Evaluation of Reports by Dr. Mark Kirk Regarding Condition of Diablo Canyon Unit 1 Pressure Vessel*

Dear Mr. Rathie:

I am responding to the questions posed in your letter of June 28, 2024, concerning my June 14, 2024 Report entitled "Technical Evaluation of Reports by Dr. Mark Kirk Regarding Condition of Diablo Canyon Unit 1 Pressure Vessel" (hereinafter "June 14 Technical Report").

1. **Concerning your graph, an image of which appears on the next page:**

   a. **Please provide data sources for each datum plotted as well as the underlying properties (copper content, nickel content, unirradiated RT_{NDT}, measured ΔT_{41J}, plant of origin, fluence, irradiation temperature). Please also describe the basis for uncertainty bands for both fluence and ART.**

The data used in the plots are from Rosier, WCAP17315-NP (2011), Terek (1993), WCAP-13750 (1993), Kirk (2024), WCAP-15958, Revision 0 (2003). Eason, E D; Wright, J E; and Odette, G R (1998), and Kirk, ERRATA # 2 (2024). I did not perform independent Reference Temperature calculations in preparing my report except for recalculating RT_{NDT}, ΔRT_{NDT}, and hence ART_{NDT}, the *USE*, and the Extent of Embrittlement (*EoE*) directly from the symmetrical hyperbolic tangent function. Accordingly, I did not require the above data sources except to verify the data used by Dr. Kirk and establish the error bars. However, I would occasionally check some of the arithmetic. I did not find any errors. The only issue is that the "Chemistry Factor (*CF*)" and the Margin appear to be treated like adjustable parameters. However, a material attains its susceptibility to radiation embrittlement for good physical reasons, such as void condensation on intermetallic precipitates that act as barriers to the movement of dislocations, resulting in the loss of ductility. Accordingly, *CF*, at least, is a physical property like a melting temperature or a heat of fusion and has a fixed value within the confines of the Gibbs Phase Rule. In the case of the Margin (*M*), it is a function of the standard deviations of the initial value of RT_{NDT} and the shift in RT_{NDT} due to neutron irradiation. The magnitude of the standard deviations can often be traced to the poor definition of some physical property, and, in this regard, *M* is not unlike *CF*. There is no question, however, that minimization of *CF* and *M* can result in a substantial decrease in ART_{NDT} and, hence, in minimization of the probability that ART_{NDT} will exceed the PTS screening criterion. It is not surprising, therefore, that seeking an allowable decrease in *CF* has become a strategy for assessing the state of radiation embrittlement of beltline components in the RPVs of PWRs.

b. Does the point labeled "especially" come from the Palisades reactor capsule or some other place? We have reviewed your citation for this datum that appeared on page 6 of your June 14, 204 document sent to the DCISC (i.e., the NRC's regulatory analysis that led to RG1.99-Rev 2, ADAMS ML10210298). We believe we found the basis for this datum on page 45 of that document. In a table on that page a value of RT_{PTS} at EOL for Palisades estimated using RG1.99R2 is given as 322 °F (or 161 °C, which corresponds to your plot). Is this the source for the datum you plotted? If so, are you aware that this value is a predicted value; it is not a measurement (all the other data points that appear on your graph are measurements)? Please let us know what justifies the inclusion of one predicted value along with five surveillance measurements (three from Diablo Canyon Unit 1 and two from Palisades) as part of your analysis?

Yes, that datum came from Table 3 of NRC's Regulatory Analysis under the entry for Palisades. Neither PTS nor the values in Rev.2, as listed in Table 3, are strictly measured nor predicted measures of embrittlement. Thus, ART_{NDT} contains RT_{NDT} for unirradiated material (mostly measured but with an exception for a given class of material). The same is true for the Margin, *M*, which contains standard deviations of the initial (unirradiated) reference temperature and the irradiation-induced shift in RT_{NDT}. ART_{NDT} also contains the Chemistry Factor (*CF*), which is a function of the composition of the alloy (Ni, Cu, and P contents). ΔRT_{NDT} contains the fluence function that requires a measure of the neutron fluence (*f*). The fluence *f* may be calculated from first principles, as is done in the Westinghouse Fluence Report, with dosimetry being used to confirm the predictions. Regardless of its origin, I could find no good reason to exclude the datum from the plot.

Embarrassingly, the plot in my report contained an inadvertent error, when I plotted the error bars in °F on the °C scale employed by Dr. Kirk. Note that °F scale is normally used for expressing ART_{NDT}. The plot has been redone to correct the error. See Figure 1 below.

In addition, for clarity, the horizontal error bar has been omitted from Capsule S as its magnitude is barely more significant than the diameter of the data point. In this context, I would note that no physico-chemical measurement or theoretical prediction can be made with infinite accuracy. All experimental and theoretical calculations have associated uncertainties, commonly expressed as error bars in the three Cartesian coordinates (x, y, and z) or two Cartesian coordinates (x and y) for a planar plot. The error bars for the data from Capsule S have been superimposed on a plot taken from Kirk (Vol. 2) in Figure 1 below.

Finally, a third ART_{41J} at 161 °C at the *EoL* fluence of 1.4×10^{19} n/cm^2 has been inserted from Kirk, RG 1.99, Rev. 2. As noted above, no valid reason for excluding the point could be found.

Additionally, I note the following:

1. The uncertainty in fluence is estimated as ±14 to 20 % [WCAP 15958 (2003)].
2. I should reiterate my critique of Dr. Kirk for omitting error bars, as this practice appears prevalent in the field.
3. The three highest fluence points are from Palisades; the three lowest are from DCPP-1.
4. Intersection with PTS screening criterion occurs between 1.2×10^{19} and 3.0×10^{19} n/cm^2.
5. The plot predicts that the beltline welds will become unacceptably embrittled sometime within the first 20-year extension or early in the next.

Assuming that the data are credible, the measured and predicted ART_{NDT} values should not differ by more than one standard deviation, which does not materially change the plot.

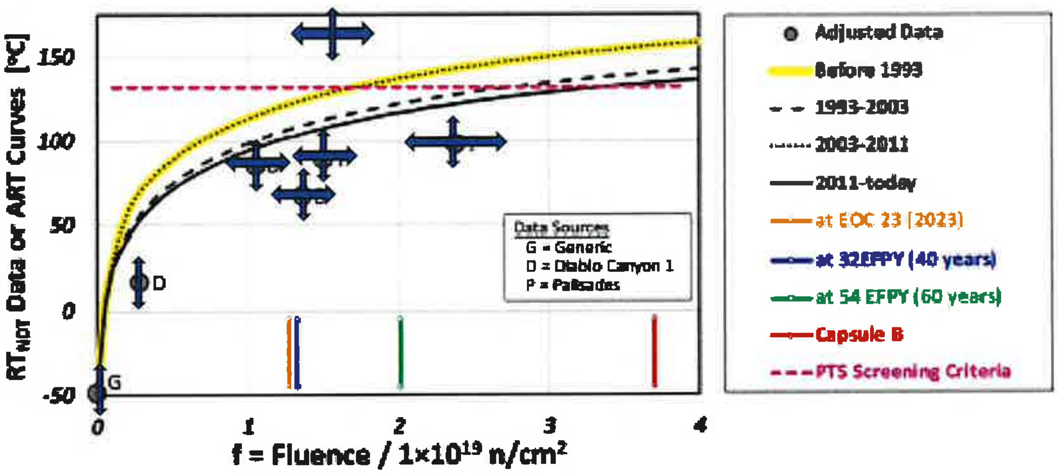

Figure 1. Corrected Figure 1 from June 14 Technical Report.

> **c. Please explain the basis for the uncertainty bands shown in your figure. Also, please clarify whether the plotted uncertainty bands represent one standard deviation or some other measure.**

The uncertainty in ART_{NDT} corresponds to ±1 standard deviation that can be found in Table 4.1 of Eason, E D; Wright, J E; and Odette, G R (1998) ("NUREG/CR-6551"). NUREG/CR-6551 considers phosphorous as a weld metal impurity in addition to Ni and Cu, which were included in RG 1.99, Rev. 2. By including phosphorus, the model of Eason et.al. is a more comprehensive treatment than that in RG 1.99, Rev. 2. This is shown by the fact that the standard deviation from RG 1.99, Rev. 2 is significantly larger than that from NUREG/CR-6551.

> **d. Embrittlement data typically show a leveling off of the increase of ART with increasing fluence. Your figure shows a straight line. Was the line in your plot fit using the method of least squares, or by some other method? Also, please describe the basis for assuming that the variation of ART with fluence is linear for the Diablo Canyon Unit 1 weld.**

In the corrected Figure 1 above, no attempt was made to perform a "least squares" fit of a linear function to the data simply because the uncertainty in the data does not warrant such an analysis. The non-linear function represented by the solid line is probably the best fit that can be made, but there are possibly many "embrittlement trend correlations (ETC)" that could fit the data equally well.

2. Concerning your "extent of embrittlement" analysis methodology:

a. Please provide evidence that the "Extent of Embrittlement" methodology proposed has been peer-reviewed as a suitable metric for embrittlement either in a peer-reviewed journal or by scientific bodies such as the American Society for Testing and Materials and the American Society of Mechanical Engineers.

The "Extent of Embrittlement" (*EoE*) was developed entirely by me. What I sought was a simple metric that could be calculated from the optimized parameters of the symmetrical hyperbolic tangent function, Equation (1),

$$FE = A + B\tanh\left(\frac{T - T_0}{C}\right) \tag{1}$$

where A, B, C, and T_0 are the optimized parameters obtained from fitting Equation (1) to the fracture energy (*FE*) vs. temperature (*T*). These fits are given in WCAP-15958, Revision 0 (2003). Because my *EoE* methodology has not been published, I have described it in great detail in Appendix IV of my June 14 Report. As discussed there, using the *EoE* method, I have been able to define a more rational reference temperature than the temperature of the arbitrarily chosen value of the Charpy Impact Fracture Energy of 30ft. lb or 41J. ASME chose this value arbitrarily because it was sufficiently low and had minimal risk of interfering with the USE. However, it has no theoretical basis and even indicates, absurdly, that irradiation may increase ductility (Eason, E D; Wright, J E; and Odette, G R (1998) (NUREG/CR-6551). As also discussed in Appendix IV of my June 14 Report, Dr. Kirk's attempt to denigrate and dismiss my work is fundamentally incorrect and displays a lack of understanding of what I did.

A publication has been prepared on this work and will be submitted to a suitable journal shortly. I am not aware of any similar treatment of Reference Temperature data published by others.

b. Please provide empirical evidence that either EoE or $RT_{NDT,PoI}$ correlates better with the true transition temperature of RPV steels, as defined by a T_0 value estimated from fracture toughness tests performed according to ASTM E1921, than does T_{41J}.

The field is in its infancy, and it is much too soon to expect the exhaustive response you appear to seek. To reproduce what is described in ASTM E1921 is well beyond my current resources. However, the concepts underlying my *EoE* methodology are backed by the force of mathematical logic. The first logical step is for Dr. Kirk to substitute RT_{PoI} for RT_{41J} in his analysis, as I suggested earlier (June 14 Report), because that is where our two treatments part ways. Let's see if we get the same result. Please note that RT_{PoI} is the acronym for "Reference Temperature, Point of Inflection" and corresponds to the temperature at which the point of inflection occurs in Equation (1) when optimized on the *FE* vs. *T* Charpy curve.

c. What are the acceptance criteria for "extent of embrittlement" and who developed and approved them?

As discussed above, the *EoE* methodology is my own development and therefore there are no "acceptance criteria." However, Figures IV.2 through IV.4 of my June 14 Report demonstrate that one obtains the expected result by applying the *EoE* methodology. With these analyses, I can offer the heavy hand of mathematical logic by way of approval. Hopefully, others will adopt the calculation of *EoE*, and with sufficient time, it will eventually become accepted as a useful metric.

3. Concerning hydrogen induced cracking:

a. Please provide evidence that US nuclear plants have exhibited Hydrogen Induced Cracking and that this cracking is of sufficient extent to affect the integrity of the reactor pressure vessel during either routine operations or under postulated accident conditions.

Hydrogen-induced brittle fracture occurs extensively on the primary coolant side of the RPV in components that are fabricated from highly cold-worked stainless-steel bolts (e.g., baffle bolts), milled-annealed Alloy 600 (steam generator tubing, pressurizer), and Alloy 182 weld metal [Chene (2016), Fekete et.al (2018), Kim et.al (2016), Kim et.al (2019), Koutsky et.al (1986), Mazel et.al (1980), Namboodhiri (1984), Platt et.al (2019), Shi et.al (2018), Symons (1999), Toribio et.al (2017), Westerman (1961), Yonezawa et.al (1983), and Young et.al (2012)]. As discussed in Appendix VI of my June 14 Report, the possibility of hydrogen segregation into the voids and fissures in the beltline region raises the ugly head of hydrogen embrittlement or hydrogen-induced cracking (HIC), which must be addressed. This is because hydrogen-induced cracking, a form of hydrogen embrittlement, is already recognized as the mechanism of the primary-side failure of cold-worked Alloy 600 steam generator tubes at bends, and expansion joints into the steam generator tube sheets (Figure VI.7 of my June 14 Report) and in penetrations through the carbon steel support plates in the event of denting corrosion. Ni-base alloys, including Alloy 600, Alloy 182, Alloy 690, and Inconel 800, are used extensively in PWRs.

The root cause of the cracking of the Davis Besse pressure vessel head was HIC of the weld holding the Control Rod Drive tube at the ID of the penetration of the tube through the reactor head [Crane and Cullen (2004)]. The corrosion that resulted from the cracking of the Davis Besse reactor head was sufficient to affect the integrity of the pressure vessel because the primary coolant was retained in the vessel only by the stainless-steel liner. If the liner had ruptured, a LOCA would have occurred, and the possibility of PTS would have to be considered. Accordingly, boric acid corrosion and the underlying cause of HIC in Alloy 82/182 weld metal between the RPV internal region and the annulus between the control rod tube and the reactor head, must be considered a threat to the integrity of the RPV. Importantly, Davis Besse was not an isolated incident as indicated in the following paragraph.

As discussed in Appendix VI of my June 14 Report, HIC and boric acid corrosion are related, because HIC in the Alloy 82/182 allows access of the boric acid/lithium hydroxide primary coolant to the hot annulus between the control rod guide tube and the RPV. Once in the annulus, the primary coolant boils and becomes very concentrated in boric acid due to the evaporation of water. Concentrated boric acid has proven to be very corrosive towards RPV steel, resulting in the loss of a massive amount of the steel (a roughly 18" diameter hole, in the case of Davis Besse [Crane and Cullen (2006)]. A useful survey of the "boric acid" head corrosion episodes has been published by NRC in Crane and Cullen (2004). The Davis Besse incident of corroded reactor pressure vessel head was the first reported US event of which many scientists were aware. Following the Davis Besse event, however, further research revealed multiple incidents of boric acid corrosion of threaded fasteners during the 1970s, 1980s, and 1990s. These incidents included events at Palisades, Zion Unit 1, Calvert Cliffs Units 1 and 2, St. Lucie Unit 1, Surry Unit 2, Arkansas Nuclear One Units 1 and 2, Maine Yankee, Fort Calhoun, HB Robinson, Oconee Units 2 and 3, Millstone Unit 2, Indian Point Unit 2, DC Cook Unit 2, Kewaunee, North Anna Unit 1, San Onofre Unit 2, Waterford, and Davis Besse Several PWRs in Switzerland and France also experienced the same problem [Crane and Cullen (2006)]. Then there is all of the HIC in mill-annealed Steam Generator Tubing, which does not involve the RPV per se but results in penetration from the primary loop to the secondary loop. See Appendix VI of my June 14 Report (Hydrogen Embrittlement in PWRs).

b. Please show evidence that the Unit 1 reactor pressure vessel contains cold worked 316L stainless steel, alloy 600, or alloy 182.

It is not just a question of the composition of the reactor vessel itself. The condition of the pressure vessel may be affected by corrosion of alloys in the pressure vessel internals. All Westinghouse-designed reactors contain various grades of stainless steels and other embrittlement-prone materials. More than 100 grades are employed, although not all simultaneously, including 316L, 304, and/or 347) [Feron, et al. (2012)]. The other embrittlement-prone alloys include Alloy 600, Alloy X-750, various precipitation-hardened (pre-embrittled) high-strength alloys such as PH 13-8Mo, 17-4 PH, 15-5 PH, and 17-7 PH, and Alloy 82/182. Diablo Canyon, Unit 1 is no different. Feron et. al. (2012) have reviewed the use of stainless steels in PWR internals, and they identify the two most important factors affecting the corrosion resistances of these materials: (1) neutron irradiation (embrittlement) and (2) cold working (hardening, via shot-peening or swaging). The mechanism for radiation embrittlement is like that proposed for ferritic, low-alloy RPV steels, while hardening imparts enhanced sensitivity to HIC. Other nickel-based materials, such as Alloy 82/182 weld metal and mill-annealed Alloy 600, also experience HIC in PWR primary coolant circuits.

Perhaps the most striking example of Alloy 82/182 weld metal HIC occurred in the Davis Besse PWR when a crack allowed primary coolant to leak into the annulus between the control rod tube and the reactor head. The water evaporated and left behind a concentrated boric acid environment that became very acidic and corroded a large hole in the reactor head. I investigated this system for EPRI and found that as the molar ratio of water to boric acid decreased due to water evaporation, a polymerization reaction occurred to release protons, resulting in a sharp decrease in the pH to form an exceedingly corrosive environment when in contact with ferritic RPV steel.

Davis Besse became known as the "reactor with a hole in its head." Subsequent inspection of many US PWRs found it to be a common problem [Crane and Cullen (2004)].

The feature distinguishing HIC from other stress corrosion cracking (SCC) forms is the potential dependence of crack initiation and propagation. HIC-prone alloys frequently exhibit a lower critical electrochemical potential, below which the potential must be for HIC to occur. In fact, under these low electrochemical potential conditions, the cathodic partial reaction of the corrosion process is the hydrogen evolution reaction. Importantly, Tomlinson (1981) demonstrated that most of the nascent hydrogen generated on a steel's surface in high-temperature water enters the steel to promote HIC further. An excellent example of this condition for the brittle fracture of mill-annealed Alloy 600 steam generator tubing is shown in Figure VI.10 of my June 14 Report. The (corrosion) potential is displaced below the critical potential of -0.85 V_{she} by a large amount of hydrogen in the primary coolant and by operating with a relatively high pH [typically pH = 6.6 to 6.9, Figure VI.11(a)]. Figure VI.11(b) of my June 14 Report illustrates how the problem may be avoided by tailoring the chemistry of the coolant by controlling the boron and lithium concentrations throughout a fuel cycle.

This HIC phenomenon is known as Low-Temperature Crack Propagation (LTCP) and is well-established scientifically [Daum et al. (1999), Young et.al (2012), Yonezawa et al. (1983)]. Hydrogen, which is added to the primary circuit of PWRs in amounts ranging from 25 – 50 cc (STP)/kg(H_2O) to mitigate corrosion and the radiolysis of water, is suspected of causing hydrogen embrittlement of welds and, in the literature, LTCP is considered a hydrogen-assisted cracking mechanism.

Another example of HIC is the failure of high-strength, baffle former bolts, as shown in Figure 2 below.

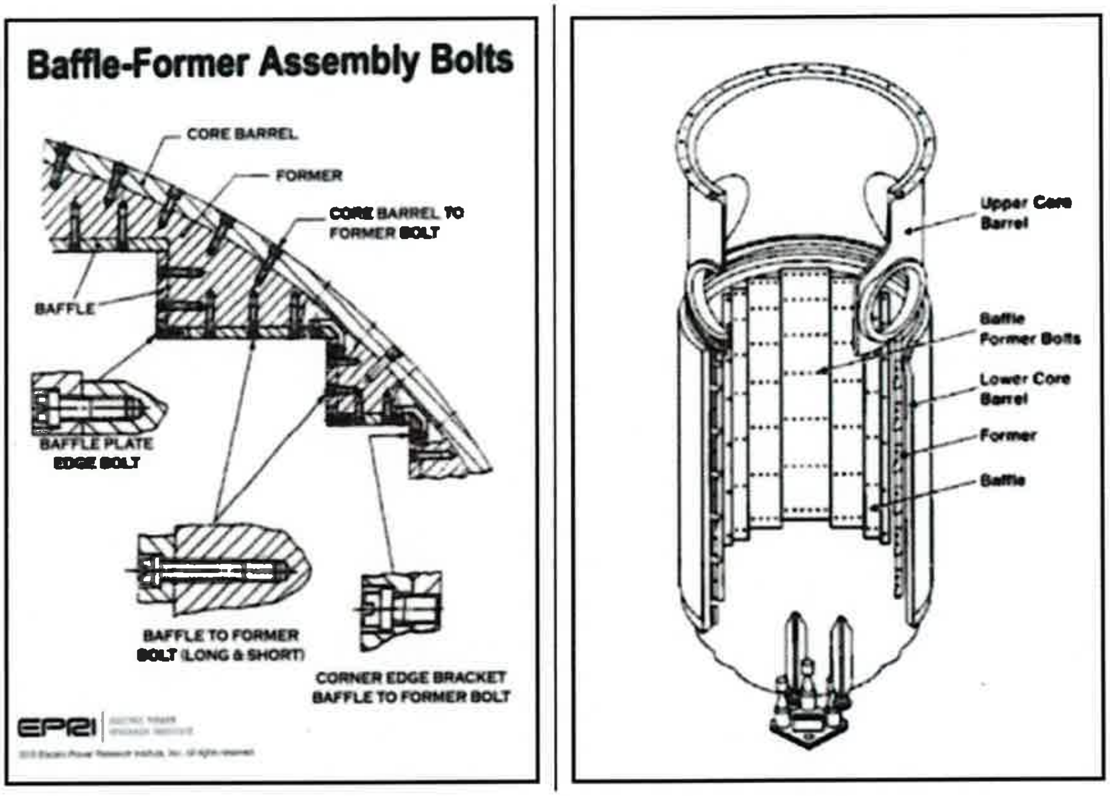

Figure 2. Configuration of baffle former bolts in a Westinghouse-designed PWR that frequently fail by HIC. [NRC (2012)]. Details of the bolts are given immediately following this caption, as follows:

> Typical baffle-former bolts are typically made of col-worked stainless steel (Type 347, Type 316, or Type 304), are approximately 5/8ths of an inch in diameter, and are typically 1.5 to 2 inches long. Units that have replaced baffle-former bolts have done so in accordance with a Westinghouse-approved design that uses bolts with less susceptible material properties and an improved head geometry to reduce stress concentrations. Bolts made from Type 316 stainless steel are less susceptible to degradation than those made from Type 347 stainless steel. Most baffle-former bolt designs secure the bolt heads with a welded lock tab . . . These lock tabs normally retain the bolt heads should they become detached.

Alloy X-750, which is also used extensively in PWR internals, is known to be susceptible to HIC, and many such events have been recorded in operating PWRs [Daum et.al(1999), Yonezawa et.al (1983), Young et.al (2012)]. Typically, pre-charging the material with hydrogen cathodically results in a precipitous loss in fracture toughness, which is commonly taken as a demonstration of HIC [Feron et.al (2012)]. Bulk hydrogen contents have been measured in nickel-base alloys after constant extension rate tests (CERT) in hydrogenated water with hydrogen overpressures of 0.005 to 0.1 MPa. [Young et.al (2012), Rubel et.al (1989)]. The measured hydrogen concentration was between 20 and 80 ppm by weight in the area near the fracture surface. It should be noted that this hydrogen concentration is much higher than indicated by Sievert's law for the solubility of hydrogen in the metal that would be in equilibrium with the 0.005 to 0.1 MPa hydrogen overpressures in water. In fact, for the 43 ppm hydrogen measured in the X-750 specimen by Yonezawa and coworkers (1983) and by Young et al. (2012), a hydrogen pressure of 20 MPa is required. To achieve this high local hydrogen pressure, the hydrogen must come from the local corrosion reaction on the specimen. Therefore, while suggesting that the cracking mechanism is hydrogen embrittlement, the local corrosion reaction is required to generate sufficient hydrogen to embrittle the material.

The analysis of the equilibrium pressure from the CERT testing assumes that the material in the fracture region is in equilibrium with the surface hydrogen concentration produced from the corrosion reaction. There are two paths for hydrogen transport through the specimen. The first is by bulk diffusion, and the second is by dislocation transport. The test time for the X-750 specimen tested at 360°C was 40 h. The diffusivity of hydrogen at 360 °C is 1.4×10^{-10} m²/s. The test time required to achieve a near-uniform hydrogen distribution through the thickness for these conditions is about 15 h. Therefore, while dislocation transport may reduce this time, accounting for the high measured bulk hydrogen concentration, it is unnecessary.

Fracture toughness values and tearing resistance of condition HTH alloy X-750 were evaluated in hydrogen gas with 38 ppm hydrogen in the metal and in air with no hydrogen at 260 °C and 338 °C. A concentration of 38 ppm hydrogen was used in this experiment to remain in the middle of the range of measurements of hydrogen in Ni–Cr–Fe alloys undergoing SCC in hydrogenated water. [Yonezawa et.al (1983)]. It was shown that at 260 °C and 338 °C, Alloy X-750 was severely embrittled in the hydrogen gas environment [Yonezawa et.al (1983)]. K_{IC} was decreased by over a factor of three. The tearing modulus decreased from over 80 to very low values in the hydrogen environment. The fracture morphology was also changed from a predominantly trans-granular ductile dimple morphology to a predominantly intergranular fracture with some trans-granular fracture facets. The hydrogen-induced intergranular fracture morphology is very similar to the fracture morphology observed on SCC specimens.

For an example of operational problems related to the use of cold-worked Alloy X-750 and Type 316 stainless steel, the DCISC is referred to the experience of Salem Nuclear Generating Station Units 1 and 2. [NRC (2015)]. Many EPRI (Electric Power Research Institute) reports summarize the essential aspects of HIC in the materials to which the reader is referred [*e.g.,* EPRI (2005-1), EPRI (2005-2), EPRI (2005-3), EPRI (2006)].

Rubel, et.al (1989) has reviewed the German (KWU) experience with Alloy (Inconel) X-750 in PWR internals. The excellent mechanical properties (high strength, high creep resistance) are the result of the precipitation of a semi-coherent γ'-phase (type Ni_3Al) during age-hardening and its distribution, as well as of the distribution and morphology of $M_{23}C_6$ carbide formation [Rubel et.al (1989)]. These precipitates act as barriers to the movement of dislocations, which results in the loss of ductility (i.e., hardening). Most significant is the damage in Japanese, French, and American PWRs to the centering pins of the control-rod guide assemblies [Rubel et.al (1989)]. The cause of damage was IGSCC because of high local stresses (in particular, notch effects arising from crosssection transitions), to-helical control rod drive springs, fuel guide pins, core barrel bolts, and other fasteners, all with unfavorable heat treatment.

Despite a great deal of information on the operational experience of these materials in PWRs worldwide, there were still concerns that the true failure mechanisms were still poorly understood. [Daum et.al (1999)] Initially, the nuclear power industry followed the treatment for Alloy X-750, which was developed by the aerospace industry for gas turbines. This heat treatment (AH heat treatment) comprised a stress-equalizing anneal (885 °C) for 24 hours and aging (704 °C) for 20 hours. This material showed great susceptibility to IGSCC and hydrogen embrittlement (HE). An alternate heat treatment, designated by HTH, was then developed to mitigate the cracking of the AH material. The HTH heat treatment required a solution anneal at 1093 °C for 1 to 2 hours followed by aging at 704 °C for 20 hours. The final HTH material had a larger grain size (100-120μm), finer γ'-precipitates, and semi-continuous intergranular carbide precipitates. The enhanced IGSCC resistance of the HTH material was attributed to the semi-continuous intergranular carbides rather than the discrete intergranular carbide precipitation found in the AH material.

Daum et al. (1999) studied the hydrogen embrittlement susceptibilities of Alloy X-750 in the AH and HTH heat-treat conditions and Alloy 625 in the direct aged (DA) condition at room temperature, 150 °C, and 288 °C in a simulated (out-of-flux) water reactor environment containing dissolved hydrogen gas (0 and 60 cc H_2/kg H_2O STP). The three materials tested showed a pronounced susceptibility to hydrogen embrittlement at ambient temperature. When evaluated in hydrogenated versus nitrogenated environments, the failure times and ductility decreased by about 35% and 50%, respectively, providing further evidence of these alloys' room-temperature hydrogen embrittlement susceptibility, including the transition from a ductile, trans-granular fracture to a predominantly intergranular fracture in the presence of dissolved hydrogen gas. The three materials tested exhibited an intergranular fracture contiguous to the specimen surface and represented about 15% of the fracture surface. Surface cracking changed from mixed mode propagation inclined to the stress axis under nitrogenated conditions to Mode I SCC (Stress Corrosion Cracking) at room temperature, hydrogenated tests, roughly normal to the stress axis. At 288 °C, none of the materials showed any hydrogen embrittlement. Ductile fracture due to microvoid coalescence occurred by cup-cone failure for the Alloy X-750 materials and Alloy 625 DA failed by ductile 45° shear. No intergranular fracture was observed. At 150 °C, Alloy 625 DA showed some susceptibility to hydrogen embrittlement, which is consistent with the low-temperature regime of susceptibility in these alloys reported by other researchers. The relative susceptibilities of these alloys to hydrogen embrittlement were indistinguishable. This observation, especially in Alloy X-750, implies that carbide precipitation does not affect HE under the testing conditions employed. Whether Alloy 625 DA is superior to Alloy X-750 under irradiated tensile conditions remains to be seen. The study by Daum et al. (1999) firmly established that the fundamental cause of fracture in Alloy X-750 is HIC.

4. Concerning surveillance

a. Please show where ASTM E-185-70 states that 5 capsules are required.

As per Table 1, ASTM E185-70 states that if the predicted shift in the Reference Temperature exceeds 200 °F, the recommended minimum number of capsules shall be 5. Once the deleterious effects of Ni and Cu were known on predisposing the welds to radiation embrittlement, and in the light of Westinghouse's warning to PG&E that the welds in Unit 1 had this compositional problem, a shift in RT_{NDT} of > 200 °F should have been immediately recognized and at least five capsules specified.

TABLE 1 Minimum Recommended Number of Surveillance Capsules and Their Withdrawal Schedule (Schedule in Terms of Effective Full-Power Years of the Reactor Vessel)

	Predicted Transition Temperature Shift at Vessel Inside Surface		
	≤ 56°C (≤ 100°F)	> 56°C (> 100°F) ≤ 111°C (≤ 200°F)	> 111°C (> 200°F)
Minimun Number of Capsules	3	4	5
Withdrawal Sequence:			
First	6^A	3^A	1.5^A
Second	15^B	6^C	3^D
Third	EOL^E	15^B	6^C
Fourth		EOL^E	15^B
Fifth			EOL^E

[A] Or at the time when the accumulated neutron fluence of the capsule exceeds 5×10^{22} n/m² (5×10^{18} n/cm²), or at the time when the highest predicted ΔRT_{NDT} of all encapsulated materials is approximately 28°C (50°F), whichever comes first.

[B] Or at the time when the accumulated neutron fluence of the capsule corresponds to the approximate EOL fluence at the reactor vessel inner wall location, whichever comes first.

[C] Or at the time when the accumulated neutron fluence of the capsule corresponds to the approximate EOL fluence at the reactor vessel ¼ T location, whichever comes first.

[D] Or at the time when the accumulated neutron fluence of the capsule corresponds to a value midway between that of the first and third capsules.

[E] Not less than once or greater than twice the peak EOL vessel fluence. This may be modified on the basis of previous tests. This capsule may be held without testing following withdrawal.

Indeed, before the limiting material had been identified (weld vs. base plate), the surveillance program specified 8 capsules, 3 containing weld metal and 5 containing base plate). We can all agree that a major problem with the Reference Temperature is the scarcity of data, which becomes acute for a 3-capsule surveillance program like that in place at Diablo Canyon, Unit 1.

b. Please provide documentation showing in the record where PG&E violated NRC's required capsule withdrawal schedule without NRC's approval of extensions.

Capsule B was inserted upon removal of Capsule Y at 5.86 surveillance EFPYs and was scheduled to be removed at 20.7 EFPYs as the fourth capsule in a four-capsule surveillance program as a condition for an NRC license amendment granting PG&E a 37-month life extension to recapture the initial time of Low Power Operation (LPOL). [NRC (2006)]. Please see the attached Opening Brief of San Luis Obispo Mothers for Peace and Friends of the Earth in *San Luis Obispo Mothers* *for Peace and Friends of the Earth v. NRC,* No. 23-3884 (now pending in U.S. Court of Appeals for the Ninth Circuit).

5. General questions

a. What materials are VVER Russian reactor pressure vessels made of? Are they similar to Diablo Canyon Unit 1? Do they experience neutron irradiation embrittlement by the same mechanisms as RPV steels made in the USA?

Toribio, et.al. (2016) describe the plate material of the RPV of a WWER 440 PWR as being comprised of "low carbon steel", but that could cover a wide range of ferritic grades not unlike Diablo Canyon, Unit 1. Like US PWRs, the inner surface of the RPV of the VVER 1000 and later models is clad with a few mm thick layer of stainless steel for general corrosion resistance. The inner surface of the WWER 440 PWR is not clad with stainless steel. Compared with Western PWRs, the WWER series of reactors differ markedly from their US counterparts in that the designs generally incorporate:

- Horizontal steam generators
- Hexagonal fuel assemblies
- No bottom penetrations in the pressure vessel
- High-capacity pressurizers providing a large reactor coolant inventory.
- Up to six independent primary coolant loops.

See Figure 3 below.

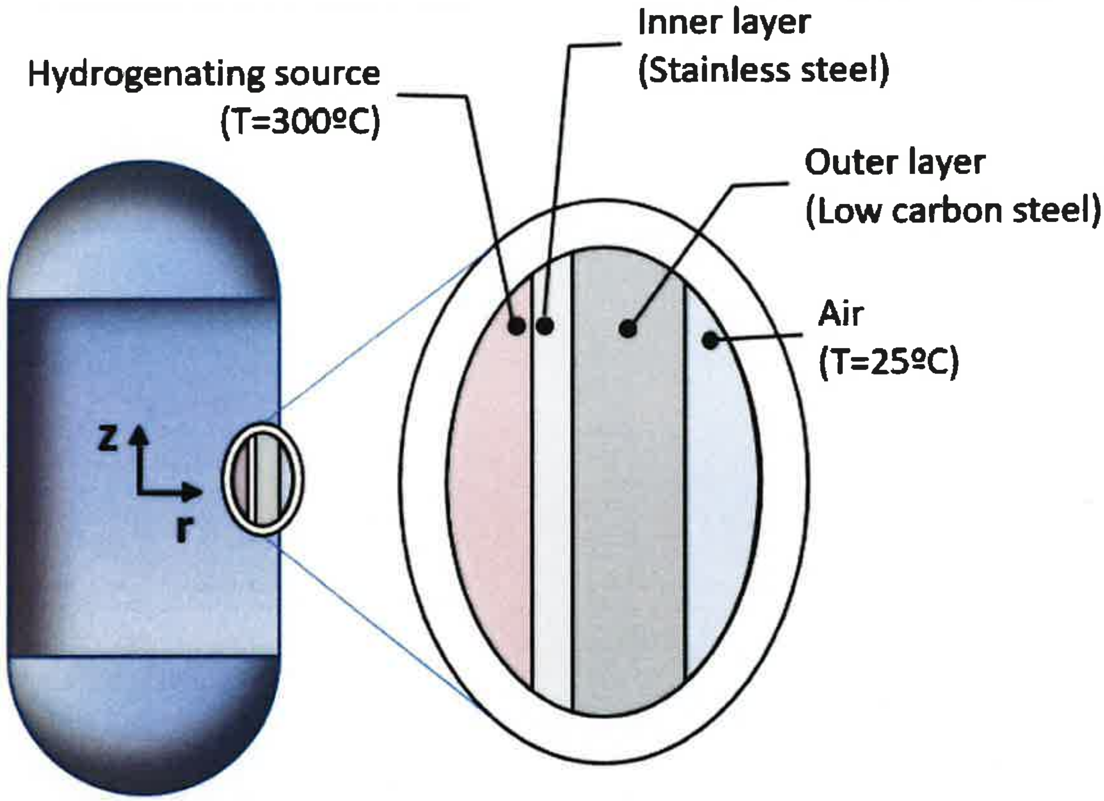

Figure 3. Schematic of the structure of the RPV of a WWER 440 Pressurized Water Reactor [Toribio et.al (2016)].

Kuleshova et.al. (2002) reported a study to directly compare PM-HIP (Powder Metallurgy-Hot Isostatic Pressing) to forged SA508 Grade 3 Class 1 low-alloy RPV steel at two neutron irradiation conditions: ~0.5–1.0 displacements per atom (dpa) at ~270 °C and ~370 °C. PM-HIP SA508 experiences greater irradiation hardening and embrittlement (total elongation) than forged SA508. This study confirms that the RPV of the WWER PWRs comprises SA508 Grade 3 Class 1 low-alloy steel, which makes them comparable to Western PWRs at least as far as the base plate of the RPV is concerned. It also confirms that radiation embrittlement is an ongoing operational problem in this reactor class.

b. What credit do you give industry-wide data from all US operating and worldwide reactors in terms of embrittlement to support US NRC and ASTM data supporting understanding of the effect of radiation damage to reactor pressure vessel steels in the development of USE and PTS limits?

I give the highest credit possible for the ongoing effort by US NRC and ASTM to understand the origins of neutron-irradiation damage, particularly from researchers such as E. Eason and G. Odette, who have made great strides in filling in the mechanistic details. Their work is to be commended for its quality of science and the thoroughness with which they have approached their tasks.

c. On slide 10 of your presentation you suggest that K1c tests should be performed, we assume following ASTM Standard E399. Do you suggest to perform these in the irradiated or un-irradiated condition? Please describe how many tests you suggest performing and how the specimen size requirements of Standard E399 will be satisfied.

In my opinion, any such study should begin with determining the fracture toughness of the unirradiated material as that provides a baseline from which to obtain ART_{NDT}. Once that is done, the fewer property correlations one has to make in arriving at a state of embrittlement metric, the better because the prediction quality will be improved. ASTM E399 covers the determination of the plane-strain fracture toughness (K_{Ic}) of metallic materials by tests using various fatigue-cracked specimens with a thickness of 0.063 in. (1.6 mm) or greater. The details of the various specimen and test configurations are shown in Annexes A1 through A7 and A9 to this standard. Plane-strain fracture toughness tests of thinner materials that are sufficiently brittle can be made with other types of specimens. There is no standard test method for testing such thin materials. This test method also covers the determination of the specimen strength ratio Rsx where x refers to the specific specimen configuration being tested. This strength ratio is a function of the maximum load that the specimen can sustain, its initial dimensions, and the yield strength of the material. Measured values of plane-strain fracture toughness stated in inch-pound units are to be regarded as standard.

Like Dr. Kirk, I prefer the direct measurement of K_{Ic} using mini-C(T) specimens. Much progress has been made in recent years in using mini-C(T) specimens to measure fracture toughness while maintaining the plane strain condition [Hosemann (2015)], and work has shown good agreement between the fracture toughness measured using mini-C(T) specimens and full-size specimens. The objective would be to machine mini-C(T) specimens from the remnants of the failed Charpy test specimens to obtain a minimum of ten mini-C(T) specimens for each of the materials in Capsules S, Y, and V. This would allow for the accurate determination of K_{Ic} and the associated standard deviation. Using the WOL specimens in each capsule may also be possible wherein, after determining the fracture arrest fracture toughness, the specimen is pulled to failure under tension to determine K_{Ic}.

Thank you for your consideration.

Sincerely,

Digby Macdonald
Digby Macdonald, Ph.D

Encl.: Brief of San Luis Obispo Mothers for Peace and Friends of the Earth in *San Luis Obispo Mothers for Peace and Friends of the Earth v. NRC,* No. 23-3884

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Sabrina Venskus, Esq., counsel to San Luis Obispo Mothers for Peace
Linda Seeley, San Luis Obispo Mothers for Peace
Hallie Templeton, Esq., counsel to Friends of the Earth
Caroline Leary, Esq., counsel to Environmental Working Group

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Case No. 23-3884

IN THE UNITED STATES COURT OF APPEALS
FOR THE NINTH CIRCUIT

SAN LUIS OBISPO MOTHERS FOR PEACE, INC.
AND FRIENDS OF THE EARTH, INC.
Petitioners,

v.

UNITED STATES NUCLEAR REGULATORY COMMISSION
and the UNITED STATES OF AMERICA,
Respondents,

PACIFIC GAS & ELECTRIC COMPANY,
Intervenor

Petition for Review of Final Administrative Action of the
United States Nuclear Regulatory Commission

PETITIONERS' OPENING BRIEF

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CORPORATE DISCLOSURE STATEMENT

In accordance with Federal Rule of Appellate Procedure 26.1, Petitioners certify that they are nonprofit organizations that have no parent or subsidiary entities. No Petitioners have stock, and therefore, no publicly held company owns 10 percent or more of its stock.

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GLOSSARY

APA	Administrative Procedure Act
ASTM	American Society for Testing of Materials
Commission	U.S. Nuclear Regulatory Commission
EFPY	Effective Full Power Years
EOL	End of Life
NRC	U.S. Nuclear Regulatory Commission
PG&E	Pacific Gas & Electric Co.

JURISDICTIONAL STATEMENT

A. Hobbs Act Jurisdiction

This case involves an appeal of a final Order entered on October 2, 2023 ("Denial Order"), by the United States Nuclear Regulatory Commission (the "NRC" or "Commission") regarding the operating license held by Pacific Gas and Electric Co. ("PG&E") for Unit 1 of the Diablo Canyon nuclear power plant. 1-ER-003.[1] The Commission's Order is reviewable by this Court under the Atomic Energy Act ("AEA"), 42 U.S.C. § 2239(b); the Hobbs Act, 28 U.S.C. § 2342(4); and the Administrative Procedure Act ("APA"), 5 U.S.C. § 702. The appeal was timely filed pursuant to 28 U.S.C. § 2344, because it was docketed on December 1, 2023, within 60 days of the date of the Commission's Order.

B. Standing of Petitioners

Section 189a of the Atomic Energy Act, 42 U.S.C. § 2239(a), requires the NRC to "grant a hearing upon the request of any person whose interest may be affected by the proceeding." Petitioners, with the support and authorization of their members, seek a hearing before the NRC to address their concerns that PG&E's ongoing lack of knowledge regarding the condition of the Unit 1 pressure vessel poses an unacceptable risk to their health and safety. *See* 2-ER-043 – 2-ER-045, 2-

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[1] This appeal concerns only the Unit 1 pressure vessel and does not include Unit 2.

ER-119 – 2-ER-121. And these concerns are germane to the purposes of the organizations. *Id.*

The NRC has found that petitioners who live within approximately fifty miles of a nuclear reactor meet the judicial test for an "affected person," *i.e.,* "injury in fact, causal connection, and redress by a favorable decision." *Calvert Cliffs 3 Nuclear Project, L.L.C. and Unistar Nuclear Operating Services, L.L.C.,* 70 N.R.C. 911, 917 (2009) (*citing Lujan v. Defenders of Wildlife,* 504 U.S. 555, 572 n.7 (1992)). Each of the Petitioners meets this test through members who live, work, and own property within 50 miles of the Diablo Canyon reactors. 2-ER-119 – 2-ER-121.

Petitioners also meet the judicial test for organizational standing because:

> [their] members would otherwise have standing to sue in their own right; (b) the interests [they] seek[] to protect are germane to the [organizations'] purpose; and (c) neither the claim asserted nor the relief requested requires the participation of individual members in the lawsuit.

Students for Fair Admissions, Inc. v. President and Fellows of Harvard University, 143 S.Ct. 2141, 2157 (2023) (quoting *Hunt v. Wash. State Apple Adver. Comm'n,* 432 U.S. 333, 343 (1977)).

STATEMENT OF ISSUES

1. Did the NRC violate the Atomic Energy Act and the APA by denying Petitioners' request for a hearing on the NRC's decision to extend the surveillance schedule for the Unit 1 pressure vessel and by failing to give any reason for its decision?

2. In extending the scheduled date for PG&E to remove Capsule B from the Unit 1 pressure vessel for embrittlement testing – from 2009 until 2024 and possibly beyond -- did the NRC amend a condition in PG&E's operating license, thereby requiring compliance with the procedural requirements of the Atomic Energy Act for license amendments?

3. In extending the schedule for removing Capsule B from the Diablo Canyon Unit 1 pressure vessel without acknowledging that it had altered the terms of PG&E's amended operating license or explaining the reasons for those changes, did the NRC violate the Atomic Energy Act and the APA?

STATUTORY ADDENDUM

In accordance with Ninth Circuit Rule 28-2.7, pertinent statutes and regulations are included in the Addendum to this Brief, beginning on Page A-1.

STATUTORY AND REGULATORY BACKGROUND

A. Atomic Energy Act

The NRC is responsible for ensuring that operation of nuclear reactors "provides adequate protection to the health and safety of the public." 42 U.S.C. § 2232. Operation of a nuclear reactor must be carried out under a license that the NRC has determined will meet this statutory standard. If the licensee wishes to modify the facility or take actions not specifically authorized by the license, the licensee must first seek an amendment to its license from the Commission. *Citizens*

Awareness Network v. NRC, 59 F.3d 284, 287 (1st Cir. 1995) (citing 42 U.S.C. §§ 2131-2133, 2237 (1988)).

Section 189a of the Atomic Energy Act requires the NRC to provide interested members of the public with a prior opportunity for a hearing on any proposed decision to amend, grant, or revoke an operating license for a nuclear facility. 42 U.S.C. § 2239(a)(1)(A). The NRC must also provide public notice in the Federal Register of its proposed licensing decisions. *Id.*

B. Implementing Regulations and Guidance for Reactor Vessel Surveillance Programs

NRC regulation 10 C.F.R. § 50.61 establishes requirements that nuclear reactor licensees must satisfy in order to demonstrate that reactor vessels in U.S. pressurized light-water reactor facilities will have adequate protection against the consequences of pressurized thermal shock events throughout their service lives. Requirements for reactor vessel surveillance programs are found in § (b)(2) and 10 C.F.R. Part 50, Appendix H. As summarized by the Commission, Appendix H:

> sets forth a surveillance program to monitor the fracture toughness of beltline materials in light-water reactor vessels. Appendix H directs licensees to attach a particular number of surveillance "capsules" to specified areas within the reactor vessel, typically near the inside vessel wall at the beltline. Each capsule contains a number of material specimens that remain exposed to radiation during plant operation. Under the Appendix H surveillance program, licensees must periodically withdraw capsules from the reactor vessel. Capsule removal permits the material specimens to be tested for changes in ductility and fracture toughness – effects of the neutron irradiation and elevated temperatures in a given reactor pressure vessel.

Cleveland Electric Illuminating Co., 44 N.R.C. 315, 317 (1996) ("*Cleveland Electric*").

In Section III.B, Appendix H also requires that a reactor's surveillance program must satisfy ASTM E 185, the established industry guidance of the American Society for Testing of Materials. ASTM E 185 "provides licensees with the criteria for determining both the minimum number of surveillance capsules that need to be installed within the reactor vessel at the start of the plant's life, and when in the plant's life -- measured in effective full-power years ["EFPY"] -- a capsule should be withdrawn for evaluation." *Cleveland Electric*, 44 N.R.C. at 317.[2]

ASTM E 185 E has been revised multiple times since the first edition in 1970. With respect to the edition of ASTM E 185 that is applicable to a particular reactor, Section III.B of Appendix H provides:

> The design of the surveillance program and the withdrawal schedule must meet the requirements of the edition of the ASTM E 185 that is current on

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2 EFPY is an irradiation or "fluence"-based measure of the life of a reactor. At the end of a reactor's design life of forty calendar years (or "end or life" ["EOL"]), a reactor typically has accumulated a fluence of 32 EFPY. At the end of a renewed operating license totaling 60 years of operation, a reactor typically will have accumulated a fluence of 48 EFPY. 2-ER-245 – 2-ER-247.

Because capsules are located closer to the reactor core than the wall of the reactor vessel itself, their fluence will be higher than the fluence of the reactor vessel itself. And some capsules are deliberately located closer to the core than others such that their fluence may be significantly higher. *See, e.g.*, 2-ER-247 (proposing relocation of Capsule V in order to "accumulate fluence at a faster rate.").

> the issue date of the ASME code to which the reactor vessel was purchased; for reactor vessels purchased after 1982, the design of the surveillance program and the withdrawal schedule must meet the requirements of ASTM E 185-82. For reactor vessels purchased in or before 1982, later editions of ASTM E 185 may be used, but including only those editions through 1982.

The Commission has held that if an operating license provides for compliance with the ASTM standard, changes to the surveillance schedule may be made without amending the reactor's operating license. *Cleveland Electric,* 44 N.R.C. at 328. However, the licensee must seek NRC review before changing the schedule, to allow the NRC to verify whether the changed schedule continues to conform to the applicable edition of ASTM E 185. *Id.,* 44 N.R.C. at 328.

STATEMENT OF THE CASE

I. INTRODUCTION

Petitioners seek their rightful opportunity to hear from NRC, and be heard, on a series of decisions that carry inordinate safety risks for communities and the environment surrounding the Diablo Canyon Nuclear Power Plant regarding a schedule for monitoring the integrity of the Unit 1 reactor vessel. As the receptacle that holds the highly radioactive core of a nuclear reactor, the pressure vessel is "perhaps the most important single component in the reactor coolant system." Final Rule, Fracture Toughness Requirements for Light Water Reactor Pressure Vessels, 60 Fed. Reg. 65,456, 65,457 (Dec. 19, 1995). *See also Yankee Atomic Electric Co.*, 34 N.R.C. 3, 12 (1991) ("*Yankee Rowe*") (pressure vessel is "one of the key components of a reactor.") Because it has no backup, the pressure vessel must be protected continuously against the risk of fracture and failure, which could lead to core melt and catastrophic consequences to public health and safety and the environment. 2-ER-079. *See also* 60 Fed. Reg. at 65,456 ("[m]aintaining the structural integrity of the reactor pressure vessel . . . is a critical concern related to the safe operation of nuclear power plants.").

Petitioners seek review of four consecutive unlawful decisions by the NRC, starting in 2008 and culminating in the 2023 decision upheld by the NRC in the Denial Order that is the subject of this Petition for Review. These decisions cumulatively extended, by a period of more than fourteen years and perhaps indefinitely, the schedule for withdrawing "Capsule B" from the Unit 1 pressure vessel and testing it for embrittlement.[3] In each of these decisions, and without explanation or rationale, the NRC abandoned a 2006 license amendment that had imposed a specific reactor vessel surveillance schedule on PG&E's Unit 1

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[3] *See* the 2008 Extension Decision, extending the withdrawal of Capsule B from 2007 to 2009, 1-ER-029 (discussed below in Section F.1); the 2010 Extension Decision, extending the withdrawal of Capsule B from 2010 to 2012, 1-ER-023 (discussed below in Section F.2); the 2012 Extension Decision, extending the withdrawal of Capsule B from 2012 to 2022, 1-ER-017 (discussed below in Section F.3); and the 2023 Extension Decision, extending the withdrawal of Capsule B from 2022 to 2024, 2025, or indefinitely, 1-ER-009 (discussed below in Section F.4).

Petitioners refer to these four decisions collectively as "Extension Decisions."

operating license. 2-ER-155. ("2006 License Amendment"). The NRC had imposed that specific surveillance schedule in exchange for granting PG&E's request to extend its operating license term by three years, changing the license expiration date from September 22, 2021 to November 2, 2024. 2-ER-161.

Despite the fact that these Extension Decisions qualified as license amendments, the NRC did not publish notice of any of the Decisions in the Federal Register. Nor did the NRC explain why -- or even acknowledge -- that it was abandoning the license condition that the agency had imposed in 2006 in exchange for extending Unit 1's operating license by three years. As a result, since 2021, the NRC has allowed PG&E to operate Diablo Canyon Unit 1 in violation of the 2006 license condition on which the extended operating license term for Unit 1 is founded. And despite the importance of the Unit 1 pressure vessel to the safety of the reactor's operation, the required 2009 inspection of the pressure vessel has been delayed by more than fourteen years. All of this has been done without a meaningful opportunity for public input.

In this appeal, Petitioner asks the Court to reverse and vacate the NRC's Denial Order and cumulative Extension Decisions based on multiple violations of federal law. First, the NRC contravened the Atomic Energy Act by failing to proactively give public notice and offer the opportunity for a public hearing before amending or revoking PG&E's license condition. Second, the NRC violated both the Atomic

Energy Act and the Administrative Procedure Act by summarily denying Petitioners' public hearing request. Finally, Petitioners seek review of NRC's related violation of the Atomic Energy Act by completely failing to justify or even acknowledge its abandonment of the terms of the 2006 License Amendment, including the lack of any safety rationale for abandoning the license condition imposed by the 2006 License Amendment.

II. STATEMENT OF THE FACTS

A. Construction Permit and Operating License for Diablo Canyon Unit 1

In 1968, the NRC issued PG&E a construction permit for Diablo Canyon Unit 1. 2-ER-226. In 1981, following completion of Unit 1 construction, the NRC issued a low-power license for the sole purpose of testing the reactor. After three years of low-power testing, the NRC issued PG&E a full-power operating license for Unit 1 on November 2, 1984. The license allowed PG&E to operate Unit 1 for forty years from the date of issuance of the construction permit, or until April 23, 2008. 2-ER-236.

B. Initial Program for Monitoring Unit 1 Reactor Pressure Vessel

In the 1970s, while construction was underway, PG&E established reactor vessel surveillance programs for the Diablo Canyon reactor pressure vessels. 2-ER-258. The purpose of these programs was to "monitor and ensure the structural integrity of reactor pressure vessels." *Cleveland Electric,* 44 N.R.C. at 317. This threat to reactor vessel integrity arises from [l]ong-term exposure to neutron radiation and elevated temperatures, causing the "ductility" of the reactor vessel materials to decrease and thereby "decreasing the vessel materials' 'fracture toughness,' or resistance to fracture." *Id.* A significant decrease in ductility renders the reactor vessel vulnerable to rupture if cold water were to be injected into the reactor vessel during a loss of coolant accident. 2-ER-085. Because the reactor vessel was purchased in the 1970s, the applicable ASTM E standard was ASTM E 185-70. 2-ER-244.

The Unit 1 surveillance program provided for placement inside the pressure vessel of "capsules" containing "specimens" or "coupons" of representative metal samples. Capsules S, Y, and V consisted of three "Type II" capsules containing "limiting" weld metal and base metal content.[4] In compliance with ASTM E 185-70, the surveillance program scheduled capsules S, Y, and V for removal at specific intervals over the forty-year operating life of the reactor and tested for embrittlement characteristics. 2-ER-247. The schedule was based on projections of when the capsules would reach certain levels of fluence -- exposure to neutron

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4 "Limiting" materials are envisioned to be the weakest components when embrittled and hence are those that will likely fail first. 2-ER-083.

irradiation -- based on their location in relation to the reactor core. Five other capsules that did not contain the "limiting" material were designated "standby," without a schedule for removal and testing. *Id.*

C. NRC promulgation of license renewal rule

In 1991, for the first time, the NRC promulgated safety regulations for the renewal of nuclear reactor licenses to allow operation for an additional twenty years after expiration of their initial forty-year licenses. 56 Fed. Reg. 64,943 (Dec. 13, 1991). The new regulations established standards for the management of aging safety equipment, including reactor pressure vessels, during a twenty-year renewal term.

D. PG&E's Supplemental Surveillance Program

1. PG&E application and NRC approval

In March 1992, PG&E applied to supplement the original Unit 1 surveillance program by adding Capsules A, B, C, and D. 2-ER-243. The supplemental surveillance program had two purposes: to provide "embrittlement data" for a possible additional twenty-year license renewal term and to "improve the overall surveillance program" by "incorporating, where possible," more recent industry and government guidance. 2-ER-244.

The proposed supplemental surveillance program "incorporate[d] both the existing surveillance capsules and the supplemental capsules," *i.e.,* Capsules S, Y, V, B, and A. 2-ER-245. PG&E described these "first five capsules" as "a modern ASTM E 185 surveillance program, within the limitations of the original program and available materials." 2-ER-247. But only four of these capsules -- Capsules S, Y, V, and B -- were given scheduled dates for withdrawal during the forty-year operating license term. Capsule A was designated "Standby," *i.e.,* reserved for future use with no specified withdrawal date. 2-ER-247.[5]

PG&E's schedule for removing and testing capsules showed that Capsules S and Y had already been removed and tested. 2-ER-247, 2-ER-252. Capsule V was scheduled for removal at 12.9 EFPY or approximately 2002. *Id.* PG&E estimated that at 12.9 EFPY, Capsule V would provide "the fluence equivalent to the vessel surface at 32 EFPY" or approximately forty years of operation. 2-ER-247. And Capsule B was scheduled for withdrawal at EFPY 19.2, or approximately 2007. *Id.* At that point, due to its location relatively close to the core, PG&E estimated that Capsule B would provide "embrittlement data through 48 effective full power years (EFPY) or approximately 60 years of operation." 2-ER-245. *See also* note 2, *supra.*

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[5] Capsules C and D were not listed as part of the "modern ASTM E 185 surveillance program" because they had the separate purpose of demonstrating "the response of the vessel material to thermal annealing and the rate of reembrittlement during reirradition after annealing." 2-ER-247.

The NRC Staff approved PG&E's supplemental surveillance program, concluding that PG&E's proposed changes to the program were "acceptable because they augment the current program, and will provide additional data on the limiting reactor vessel materials." 2-ER-234.

2. Withdrawal and testing of Capsules S, Y, and V

In 2002, PG&E withdrew and tested Capsule V from Diablo Canyon Unit 1. When the test showed that Unit 1 would be approaching a regulatory threshold for concern about embrittlement at the end of its operating life in 2021, PG&E discounted the data as "not . . . credible." 2-ER-188.[6] Instead, PG&E substituted generic data and data from other reactors. 2-ER-081. But PG&E stated that it did not intend to rely on generic data and data from other reactors indefinitely. Instead, PG&E asserted that "Capsule V is not the last planned capsule to be evaluated in the [Diablo Canyon Unit 1] surveillance program." 2-ER-189.

[6] Petitioners dispute whether PG&E complied with NRC guidance in rejecting this data and instead relying on generic data and data from other reactors. This dispute is one of the bases for Petitioners' hearing request and their concern that the NRC lacks sufficient information about the condition of the Unit 1 pressure vessel to support a conclusion that it is operating safely. *See* 2-ER-081, 2-ER-100 – 2-ER-101 and discussion below in Section II.H.

E. License Amendments for Recovery of Time for Construction and Low-Power Testing

1. 1995 license amendment to recover thirteen-year construction period

In 1995, the NRC Staff approved PG&E's application for a license amendment to "recover" or "recapture" the thirteen-year construction period for Unit 1 by changing the Unit 1 operating license expiration date from April 23, 2008 to September 22, 2021. 2-ER-161, 2-ER-211. In support of its decision, the Staff generally cited, *inter alia,* PG&E's "comprehensive vessel material surveillance program [that] is maintained in accordance with 10 CFR Part 50, Appendix H that ensures the fracture toughness requirements of Appendix G are met." 2-ER-226.

2. 2006 license amendment to allow recovery of low-power testing interval

a. NRC policy for recovery of low-power testing time

In 1999, the NRC Commissioners established a new policy of allowing reactor licensees to "recover" the initial time of low-power testing of a newly-constructed reactor under a low-power license, by adding the same amount of time to the term of a full-power license. 2-ER-191.[7] While the vast majority of reactors needed only

[7] In other words, the NRC would change the commencement date of a forty-year full-power operating license from the date the low-power license was issued to the later date when the full-power operating license was issued.

a few months for low-power testing, for some reactors -- like Diablo Canyon Unit 1 -- it took years to complete. *Id.,* 2-ER-195. For those cases, the NRC Staff devised an approach to consider the aging effects on the pressure vessel caused by allowing it to be irradiated for several more years beyond its forty-year design life. *Id.,* 2-ER-194.

In Commission-approved Policy Memorandum SECY-98-296, addressing a request from the Grand Gulf Nuclear Station to recover low-power testing time, the Staff set out its approach to the safety review as follows:

> Although there is no regulatory guidance for review of this type of recapture, the staff performed its review on the basis of the effects of aging of safety-related and other structures and components, relative to the licensing basis. The review specifically focused on the adverse effects of aging to ensure that important systems, structures, and components will continue to perform their intended functions during the requested period of recapture. The staff reviewed the effect of the recapture period on the reactor pressure vessel, structures, mechanical equipment, electrical equipment, and quality assurance and maintenance programs, and addressed outstanding safety issues. The staff concluded that no safety issues existed that would preclude an additional 28.5 months of operation.

2-ER-194. Consistent with this approach, the Staff reviewed the Grand Gulf reactor vessel surveillance program and found that it "will aid in adjusting the operational conditions in order to maintain sufficient safety margin for the prevention of brittle failure of the reactor vessel." 2-ER-204.

The Staff's review also covered specific details of the reactor vessel surveillance program:

> To date one material specimen capsule has been removed from the reactor vessel; however, by letters dated May 2 and 31, 1996, the licensee requested that it be placed back in the vessel because testing of the first capsule at 8 effective full power years (EFPY) may not be useful. The low neutron fluence and good material chemistry for the vessel will result in a minimal shift in the material properties of the specimen in the capsule. A revision to the capsule withdrawal schedule and placing the first capsule back in the vessel was approved by the staff in its letter of August 27, 1996.

2-ER-205. "Based on the above," the Staff concluded that "there is reasonable assurance that the [reactor pressure vessel] will, for the proposed license term extension requested by the licensee, be in conformity with the applicable provisions of the rules and regulations of the Commission, and the [Grand Gulf Nuclear Station] license." 2-ER-205.

b. 2006 license amendment for Diablo Canyon Unit 1

In 2005, citing the new NRC policy for recovery of time spent on low-power testing of nuclear reactors, PG&E applied to extend the Unit 1 initial operating license term by three years – the time that had been consumed by low-power testing of Unit 1. 2-ER-170.[8] In support of its application, PG&E cited both the "original" surveillance program and the "supplemental" surveillance program it had proposed and agreed to when it sought in 1995 to recapture the 13-year construction period. 2-ER-176 – 2-ER-177.

[8] PG&E clarified that the proposed extension "does not constitute license renewal." 2-ER-189.

In 2006, the NRC Staff granted the three-year license amendment. 2-ER-155, 2-ER-152. Consistent with the NRC policy set forth in Policy Memorandum SECY-98-269, the Staff performed a "Safety Evaluation" for the proposed license amendment that assessed whether or how the addition of three years to the license term for Unit 1 would affect the safety of the pressure vessel, including the "[i]mpact on the RVMSP [reactor vessel material surveillance program]." 2-ER-165 – 2-ER-166.

In reviewing PG&E's license amendment application, the Safety Evaluation cited the Staff's previous determination that:

> The supplemental RVMSP withdrawal schedule met the criteria of ASTM E185-70 and constituted an acceptable withdrawal schedule for implementation under 10 CFR Part 50, Appendix H.

2-ER-165. The Staff also clarified that "this supplemental program" consisted of "four capsules, Capsule S, Y, V, and B, [that] were designated for removal from the [Diablo Canyon Unit 1 reactor vessel]." *Id.*

Further, the Staff noted that "Capsules S, Y, and V have been removed and tested in accordance with the licensee's program." *Id.* This left only the fourth capsule, Capsule B.

As per Policy Memorandum SECY-98-296, the Safety Evaluation then assessed whether adding three years to the operating life of Unit 1 would affect the adequacy of the reactor vessel surveillance program. 2-ER-163. The Staff concluded that "the adjustments to the withdrawal time and projected neutron fluence for Capsule B will still be in compliance with 10 CFR Part 50, Appendix H" because:

> The request to recover the testing time for DCPP-1 *amends the projected withdrawal for Capsule B to approximately 20.7 EFPY* [i.e., around 2009], when the capsule is projected to achieve a neutron fluence of 2.9 x 10^{19} n/cm^2 (E > 1.0 MeV). Therefore, the capsule will achieve a neutron fluence approximately equal to twice the projected limiting inside RV fluence for DCPP-1 at the EOL (i.e., approximately 2 * 1.43 x 10^{19} n/cm^2 [E > 1.0 MeV]).

2-ER-165 (emphasis added). Further, the Staff found that this amended withdrawal schedule "complies with the criterion in ASTM E185-82 for withdrawal of the final capsule of a four capsule withdrawal program." *Id.*

Based on these findings, the Safety Evaluation stated:

> The NRC staff has reviewed PG&E's license amendment request to recover the low-power testing time that was performed during the initial startup of the [Diablo Canyon Units 1 and 2] reactors. The NRC has determined that authorization of the requested license may be granted *based on the following conclusions:*
>
> . . .
>
> (3) The [reactor vessel] surveillance capsule withdrawal schedules for [Diablo Canyon Units 1 and 2] remain in compliance with the requirements of 10 CFR Part 50, Appendix H, and the ASTM E 185 versions of record for the units.

Id. (emphasis added). Accordingly, the Safety Evaluation "concluded" as a general matter that:

> based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by

> operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

2-ER-167. The license amendment therefore changed the Unit 1 operating license expiration date from to September 22, 2021 to November 2, 2024. 2-ER-161. This three-year extension was equivalent to a new "end of life" EFPY of 35.2. 2-ER.[9]

Thus, in approving the license amendment to add three more years to the operating license of Unit 1, the NRC Staff upgraded the Unit 1 surveillance program from a *three-capsule program* compliant with *ASTM E 185-70* to a *four-capsule program* compliant with *ASTM E 185-82*. Further, there was no question that the Staff intended withdrawal of Capsule B to be carried out during the initial operating license term for Unit 1, because -- by PG&E's own assertion -- the license amendment proceeding had nothing to do with license renewal.[10]

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[9] In 2023, by operation of an NRC-issued exemption to the agency's timely renewal rule, 10 C.F.R. § 2.109(b), this termination date was changed to the date when the NRC makes a decision on PG&E's pending license renewal application. Petitioners' appeal of the exemption is pending before this Court in *San Luis Obispo Mothers for Peace, et al. v. U.S. Nuclear Reg. Comm'n,* No. 23-852.

[10] As PG&E stated in its license amendment application, "[t]he proposed amendments do not constitute license renewal." 2-ER-174.

F. NRC Decisions Extending Schedule for Withdrawal of Capsule B From Unit 1 Pressure Vessel

1. 2008 Extension Decision: from 2007 to 2009

In 2008, PG&E asked the NRC to extend the schedule for removing Capsule B from the Unit 1 pressure vessel, from 20.7 EFPY (2009) to 21.9 EFPY (2010). 2-ER-147. PG&E did not ask for an amendment to its operating license to change the Capsule B withdrawal schedule that the NRC Staff had incorporated into its 2006 license amendment. Instead, it simply stated that the current withdrawal schedule did not meet the requirement of NRC license renewal guidance document NUREG-1801, which requires that a reactor vessel surveillance program must have a "vessel material coupon that has a fluence exposure equivalent to 60 years of operation." 2-ER-148. According to PG&E, "[a] removal time of approximately 21.9 EFPY for Capsule B satisfies NUREG-1801." 2-ER-150.

In 2008, the Staff granted PG&E's request to extend the schedule for withdrawing Capsule B. 1-ER-029. But the Staff neither acknowledged that its decision constituted a license amendment, nor gave public notice of a hearing opportunity as required by § 189a of the Atomic Energy Act, 42 U.S.C. § 2239(a)(1)(A). To the contrary, the Staff's decision did not even mention that its 2006 license amendment had been based upon PG&E carrying out an ASTM E 185-82-compliant four-capsule reactor surveillance program that included Capsule B. Instead, the Staff erroneously stated that "the withdrawal and testing of Capsule

V [in 2003] fulfilled the third and final recommendation of ASTM E 185-70 for the current ... Unit 1 operating license." 1-ER-033.[11]

2. 2010 Extension Decision: from 2010 to 2012

That same year, in 2010, PG&E applied to the NRC for a second extension of the schedule for removing Capsule B from the Unit 1 pressure vessel, now seeking to change the date from 2010 to 2012. 2-ER-140. This time, PG&E asserted the extension was necessary because it had not been able to remove Capsule B in 2010. *Id.*

Once again, PG&E did not seek a license amendment, nor did it mention the NRC's 2006 license amendment decision. Instead, it asserted that it had "withdrawn and tested three capsules from Unit 1 that meet the three recommendations of ASTM E 185-70 and the approved supplemental surveillance capsule withdrawal changes listed in NRC staff Safety Evaluation dated September 4, 1992." 2-ER-141. Further, PG&E repeated the Staff's misrepresentation of its 1995-approved surveillance program as a three-capsule program asserting that the withdrawal of Capsule V in 2003 had "fulfilled the third and final recommendation of ASTM E 185-70 for the current DCPP operating license." *Id.* Instead, PG&E proposed to withdraw Capsule B at a fluence of "approximately 60 EFPY for the

[11] In 2009, following the First Extension Decision, PG&E applied for renewal of its operating license. 75 Fed. Reg. 3,493 (Jan. 21, 2010). *See also* Section G below.

reactor pressure vessel," to provide "fluence data for the period of extended operation for license renewal." *Id.*

The NRC approved the requested schedule change, again without providing public notice or an opportunity to request a hearing, or providing a reason for why it was abandoning the alteration of the vessel surveillance schedule on which it had conditioned the 2006 license amendment. 1-ER-023 – 028. And instead of acknowledging that in 2006 it had approved a four-capsule ASTM E 185-82-compliant surveillance program culminating with the withdrawal of Capsule B in 2009, the Staff again mischaracterized the Unit 1 surveillance program as a three-capsule program for which the third and already-removed Capsule V – not the fourth and still-remaining Capsule B – was the "final capsule." 1-ER-026. The Staff also incorrectly asserted that Capsule B did "not currently form a part of the licensee's surveillance program." *Id.* Instead, according to the Staff, Capsule B would become part of a future "surveillance schedule for the license renewal period." *Id.*[12]

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[12] The excision of Capsule B from the current surveillance program was further emphasized in the following paragraph:

> The surveillance capsule withdrawal plan spanning the initial license period has already been completed and, as such, forms no part of this evaluation. The DCPP LRA and the associated withdrawal schedule have not yet been approved; however, the NRC staff believes that the proactive consideration of Surveillance Capsule B for the period of extended operation adds to the consideration of this request and addresses it below.

Finally, in an apparent effort to obfuscate the obvious violation of ASTM E 185-82, the Staff strung together a partially false statement with a vague and misleading characterization of applicable ASTM standards:

> [T]he evaluation criteria of ASTM E-70 do not apply to Surveillance Capsule B, since it does not currently form a part of the licensee's surveillance program. The licensee's [reactor vessel] material surveillance program conforms to ASTM E 185.

1-ER-026. The Staff was correct in stating that the evaluation criteria of ASTM E-70 do not apply to Surveillance Capsule B – but incorrect about the reason. The reason for not applying ASTM E-70 to Capsule B was that the 2006 License Amendment upgraded the applicable ASTM E standard to ASTM E-82. And the statement that PG&E's surveillance program conforms to ASTM E 185 is vague and misleading, because it is not specific about what edition of ASTM E 185 is applicable to PG&E's surveillance program. The Staff failed to acknowledge that in granting the 2006 License Amendment, it explicitly decided that ASTM E-82 constituted the applicable industry standard for PG&E's surveillance program.

3. 2012 Extension Decision: from 2012 to 2022

In 2011, PG&E asked the NRC for a third extension of the schedule for removing Capsule B from the Unit 1 pressure vessel. 2-ER-134. This time, PG&E sought a *ten-year* extension, from 2012 to 2022. The new date would bring

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1-ER-026. As discussed in Sections F.3 and F.4 below, this same paragraph would appear as boilerplate language in the two additional extension decisions to follow.

withdrawal of Capsule B to within two years of the expiration of the Unit 1 operating license in 2024.

The requested extension did not relate at all to the NRC-approved surveillance program for Diablo Canyon Unit 1. Rather, it was "proposed to support data acquisition" for an industry-wide research program related to reactor license renewal. *Id.* For these research purposes, PG&E sought to delay removing Capsule B until it had accumulated "approximately twice the 60-year fluence" that would have been achieved by removing the capsule in 2012 – and which would take another ten years. *Id.*

PG&E's application did not request a license amendment or even mention the 2006 license amendment decision. Instead, PG&E again asserted that the withdrawal of Capsule V in 2003 had "fulfilled the third and final recommendation of ASTM E 185-70" for the "current" Unit 1 operating license. *Id.*

In 2012, the NRC Staff approved the schedule change. 1-ER-017. The Staff did not provide public notice or offer a hearing on this additional amendment to PG&E's amended operating license, nor did it mention the four-capsule surveillance program upon which the Staff had conditioned the 2006 license amendment. Instead, using the now-boilerplate language it had employed in the previous extension decision, the Staff asserted that the surveillance program for the initial license term had "already been fulfilled" and therefore formed "no part of this evaluation." 1-ER-020. *See also* note 12 above.

4. 2023 Extension Decision: from 2022 to 2023 or 2025 or indefinitely delayed

In 2023, PG&E requested a fourth extension for withdrawal of Capsule B, from 2022 until either late 2023 or sometime in 2025 – either the very eve of the extended license expiration or after expiration. 2-ER-125. Again, PG&E did not seek a license amendment or even mention the 2006 license amendment decision. Instead, PG&E characterized Capsule B as a "standby" capsule – a capsule with no scheduled withdrawal date at all. 2-ER-132.

The NRC Staff approved the extension. 1-ER-009. Once again, the Staff offered neither a rationale for abandoning its 2006 license amendment decision, nor public notice of its decision or an opportunity to request a hearing. The Staff agreed with PG&E that Capsule B was a "standby" capsule for which no withdrawal schedule existed and approved a new scheduled withdrawal date of 2023 or 2025. 1-ER-012. The Staff's approval letter also included the same boilerplate language that had appeared in the previous two extension decisions, stating that the surveillance program for the initial license period had been "completed" and thus formed "no part" of the Staff's evaluation. 1-ER-020. *See also* note 12 above.

The NRC Staff also left open the possibility that the schedule for removing Capsule B will be extended yet again, stating that the Staff "does not make any conclusion regarding the future use of the subject capsule in any potential future licensing applications or license periods." 1-ER-009.

G. Subsequent Treatment of PG&E's 2009 License Application: Submission, Withdrawal, and Re-Submission

PG&E applied for renewal of its operating license in 2009, 75 Fed. Reg. 3,493, but withdrew the application in 2018. 83 Fed. Reg. 17,688 (Apr. 23, 2018). After that, until 2022, PG&E made plans to close the reactors on their operating license expiration dates of 2024 and 2025. Perhaps for this reason, PG&E never sought an extension of the 2022 deadline for removing Capsule B; nor, to Petitioners' knowledge, did it attempt to remove Capsule B.

In 2022, following passage of state legislation offering a substantial public subsidy to encourage PG&E to seek license renewal once again, PG&E reversed its decision to retire the reactors. The company then obtained an exemption from the NRC's Timely Renewal Rule, allowing it to continue operating the Diablo Canyon reactors indefinitely pending NRC action on a forthcoming license renewal application.[13]

[13] As discussed above in note 9, Petitioners' appeal of the exemption is pending before this Court in *San Luis Obispo Mothers for Peace, et al. v. U.S. Nuclear Reg. Comm'n,* No. 23-852.

H. Petitioners' Hearing Request and Request for Emergency Action by the Commissioners

On September 14, 2023, after learning of the NRC's decision to extend the schedule for withdrawal of Capsule B to 2024, 2025, or later, Petitioners submitted a hearing request to the NRC Commissioners. 2-ER-036. NRC's response to Petitioners' hearing request gives rise to the issues currently before the Court.

Petitioners' hearing request included the following contention:

> PG&E's request to postpone the withdrawal and testing of Capsule B until 2025 should be denied, and the Staff's decision to approve it should be reversed, because it is inconsistent with NRC safety regulations 10 C.F.R. Part 50, Appendices G and H and 10 C.F.R. §§ 50.55a and 50.61 and poses an unacceptable risk to public health and safety in violation of NRC regulations and the Atomic Energy Act. Moreover, neither PG&E nor the Staff has any legal grounds for claiming that withdrawal of Capsule B relates only to license renewal and is unnecessary to maintain safety in the current license term.

2-ER-061. In the same pleading, Petitioners asked the Commissioners to order the immediate shutdown of Unit 1 for failure to obtain needed embrittlement data for over twenty years due to the repeated extensions of time for withdrawing and testing Capsule B. Petitioners asked the Commissioners to keep the reactor in a shutdown condition pending the removal and testing of Capsule B and a thorough assessment of the state of embrittlement of the Unit 1 pressure vessel. 2-ER-065 – 2-ER-068.

Both the hearing request and request for emergency enforcement action were supported by the declaration of Dr. Digby Macdonald, Professor in Residence at the University of California at Berkeley and a highly qualified expert on materials embrittlement in nuclear reactor pressure vessels. 2-ER-072. Dr. Macdonald's lengthy and detailed declaration explained the basis for his expert opinion that the current operation of Diablo Canyon Unit 1 "poses an unreasonable risk to public health and safety due to serious indications of an unacceptable degree of embrittlement, coupled with a lack of information to establish otherwise." 2-ER-076. In Dr. Macdonald's professional judgment, PG&E had inappropriately rejected data from Capsule V indicating that the Unit 1 pressure vessel could approach an unacceptable state of embrittlement by 2021; and furthermore, that PG&E had failed to withdraw and test Capsule B and therefore had no additional reactor-specific data on which it could rely. 2-ER-079 – ER-082. Therefore, Dr. Macdonald recommended that "the reactor should be closed until PG&E obtains and analyzes additional data regarding its condition." 2-ER-076.

Despite the gravity of the concern and detailed support provided by Petitioners and Dr. Macdonald for their charges, the Commissioners themselves did not respond to the hearing request or the request for emergency action. Instead, on their behalf, the Secretary of the Commission issued a brief three-page decision denying both requests.

The Secretary's grounds for denying the hearing request consisted of two paragraphs that merely restated the mischaracterizations of the record that had been repeated time and again in the Staff's four extension decisions:

> The Petitioners argue that they are entitled to a hearing because the Extension Approval constitutes a license amendment. But the Extension Approval, by its own terms, does not amend or otherwise affect Diablo Canyon's current license. The Extension Approval does not "grant the licensee any 'greater operating authority,' or otherwise 'alter the original terms of the license,'" the relevant factors in determining whether a Staff action constitutes a license amendment. In its evaluation of the schedule revision, the Staff specifically notes that
>
>> additional capsules are not needed to satisfy the requirements of Appendix H to 10 CFR Part 50 and ASTM E 185-70 for the current operating license period … the licensee's compliance with Appendix H to 10 CFR Part 50 and ASTM E 185-70 with respect to the current operating license period for Diablo Canyon, Unit 1 forms no part of the NRC staff's evaluation of the licensee's proposed revision to the withdrawal schedule for supplemental surveillance ….
>
> The Staff further observes that it "does not make any conclusion regarding the future use of the subject capsule in any potential future licensing applications or license periods."

Therefore, the Secretary concluded that "[b]ecause the current license for Diablo Canyon, Unit 1, has not been amended, the Extension Approval does not trigger an opportunity to request a hearing." 1-ER-005.

With respect to Petitioners' request for emergency action, the Secretary's only response was to state: "I refer Petitioners' underlying concerns to the Executive Director for Operations for consideration under 10 C.F.R. § 2.206." 1-ER-005. Thus, the Secretary referred Petitioners' request for emergency action to the very same agency officials who had unlawfully extended the schedule for withdrawing Capsule B, under the regulatory framework of a petition for discretionary enforcement action whose outcome would be unreviewable by this court or any other. *See, e.g., Safe Energy Coalition v. U.S. Nuclear Reg. Comm'n,* 866 F.2d 1473, 1479 (D.C. Cir. 1989) (citing *Heckler v. Chaney*, 470 U.S. 821 (1985) (holding that NRC's denial of an enforcement petition for revocation or modification of an existing license constitutes an unreviewable exercise of agency discretion).[14]

I. Petition for Review

On December 1, 2023, Petitioners submitted a petition for review of the NRC's decision denying their hearing request without a meaningful explanation.

STANDARD OF REVIEW

Under the APA, agency decisions will be set aside if "arbitrary, capricious, an abuse of discretion, or otherwise not in accordance with law." *Public Citizen v. NRC*, 573 F.3d 916, 923 (9th Cir. 2009) (citing 5 U.S.C. § 706(2)(A)). In reviewing "predominantly legal questions" rather than "factual ones," this Court applies a standard of "reasonableness." *Alaska Wilderness Recreation & Tourism v. Morrison,* 67 F.3d 723, 727 (9th Cir. 1995) ("[I]t makes sense to distinguish the

[14] The Staff denied the petition for emergency enforcement action on March 8, 2024. The decision post-dates the decisions on review and therefore is not part of the record.

strong level of deference we accord an agency in deciding factual or technical matters from that to be accorded in disputes involving predominantly legal questions."). *See also Northcoast Envtl. Ctr. v. Glickman,* 136 F.3d 660, 667 (9th Cir. 1998); *Price Rd. Neighbor. Ass'n v. U.S. Dept. of Transp*, 113 F.3d 1505 (9th Cir. 1997)); *San Luis Obispo Mothers for Peace v. NRC*, 449 F.3d 1016, 1028 (9th Cir. 2006).

When reviewing an agency's application of its own regulation, the agency's interpretation of its regulation must be given controlling weight unless it is plainly erroneous or inconsistent with the regulation. *Alaska Ctr. for the Env't v. United States Forest Serv.*, 189 F.3d 851, 857 (9th Cir. 1999). But an agency's interpretation of its own regulation will not be upheld if it "lacks the quality necessary to attract judicial deference." *Guard v. United States Nuclear Reg. Comm'n,* 753 F.2d 1144, 1148-49 (D.C. Cir. (1985). To determine whether agency action is arbitrary or capricious, a court must consider "whether the decision was based on a consideration of the relevant factors and whether there has been clear error of judgment." *Id.* at 859 (citing *Marsh v. Oregon Natural Resources Council*, 490 U.S. 360, 378 (1989)). Precedent behests this Court to reverse the NRC under the arbitrary and capricious standard if:

> [T]he agency has relied on factors that Congress has not intended it to consider, has entirely failed to consider an important aspect of the problem, or has offered an explanation for that decision that runs counter to the evidence before the agency or is so implausible that it

> could not be ascribed to a difference in view or the product of agency expertise.

Public Citizen, 573 F.3d at 923.

SUMMARY OF THE ARGUMENT

Petitioners challenge the NRC's wholesale abandonment of a condition in PG&E's amended operating license, without providing public notice, explanation, or any opportunity to challenge the NRC's abdication in a hearing. The license condition that the NRC abandoned was designed by the NRC to ensure that extending the operation of Diablo Canyon Unit 1 by three years past its 2021 expiration date would not pose an undue accident risk to the pressure vessel, which is "perhaps the most important single component in the reactor coolant system." 60 Fed. Reg. at 65,457. In that license condition, imposed via a 2006 License Amendment for Unit 1, the NRC upgraded the industry standard applicable to the Unit 1 reactor vessel surveillance program from ASTM E 185-70 to ASTM E 185-82, requiring a four-capsule surveillance program instead of a three-capsule program. And it required that Capsule B -- the fourth capsule -- must be withdrawn at 20.7 EPFY or approximately in 2009.

Over the following fifteen-year period, from 2008 to 2023, the NRC issued a series of Extension Decisions that not only postponed the schedule for removing Capsule B from the Unit 1 reactor vessel by fifteen years or perhaps indefinitely, but that attempted -- without explanation or rationale -- to erase the condition imposed by the NRC in the 2006 License Amendment as a predicate for adding three more years to Unit 1's operating license term, changing the expiration date from 2021 to 2023.

Instead of acknowledging the 2006 License Amendment or the condition it had imposed on the Unit 1 operating license, the NRC asserted that Unit 1 was governed by the outdated ASTM E 185-70 standard and PG&E's previous three-capsule surveillance program. And the Staff maintained that this three-capsule program had been fulfilled by the removal of Capsule V in 2003, thus making it unnecessary to remove Capsule B in the current license term.

The agency's abandonment of this important license condition in its four Extension Decisions and its refusal to grant Petitioners a hearing on those Decisions violated the Atomic Energy Act and the APA in three significant ways.

First, the NRC violated the Atomic Energy Act and the APA by refusing to grant Petitioners a hearing on the 2023 Extension Decision and its predecessor decisions. These decisions collectively amended conditions in the Unit 1 operating license to authorize PG&E to operate Unit 1 in a manner that exceeded the limits imposed by the 2006 License Amendment, thereby triggering the procedural obligations of the Atomic Energy Act to provide a hearing opportunity. *Citizens Awareness Network,* 59 F.3d at 295.

Second, the NRC violated the Atomic Energy Act's requirement that changes to operating licenses must be supported by findings that those changes will not pose an unreasonable risk to public health and safety. As required by 10 C.F.R. § 50.92(a), the NRC's review of license amendment applications must be "guided by the considerations which govern the issuance of initial licenses." These considerations include whether the license amendment will "protect the health and safety of the public." 42 U.S.C. § 2133(b).

Finally, the NRC's failure to support or even acknowledge its abandonment of the 2006 License Amendment violates the APA's requirement for reasonable decision-making on the primarily legal question under the Atomic Energy Act of whether it was required to justify a change to a previous safety determination. The NRC's unannounced and unexplained abandonment of the condition it imposed in 2006 was also arbitrary and capricious because the agency failed to provide any basis, let alone a reasoned basis, for its change of position. To the extent the Denial Order did attempt to explain the basis for the NRC's decision, its explanation "[ran] counter to the evidence before the agency" and was "so implausible that it could not be ascribed to a difference in view or the product of agency expertise." *Public Citizen*, 573 F.3d at 923.

The NRC's violations of the Atomic Energy and the APA have practical, significant ramifications for public health and safety as well as the credibility of the agency. With respect to public health and safety, PG&E has now operated Unit 1 for more than twenty years without withdrawing any capsule from the Unit 1 pressure vessel. And as discussed in Section D.2 above, PG&E has no data from the most recently withdrawn capsule – Capsule V in 2003 – that it considers credible. Further, given that the NRC has now dubbed Capsule B a "standby" capsule, it appears unlikely that Capsule B will be withdrawn any time soon.[15] In the meantime, the NRC has prevented Petitioners, if not the general public, from holding the agency accountable for this regulatory manipulation, by denying Petitioners' hearing request without a word of explanation.

[15] As stated in the NRC decision rejecting Petitioners' hearing request: The Staff further clarifies that it "does not make any conclusion regarding the future use of the subject capsule in any potential future licensing applications or license periods." 1-ER-005.

ARGUMENT

I. THE NRC VIOLATED THE ATOMIC ENERGY ACT WHEN IT AMENDED THE OPERATING LICENSE FOR DIABLO CANYON UNIT 1 WITHOUT PROVIDING PUBLIC NOTICE OR THE OPPORTUNITY TO REQUEST A HEARING AND WITHOUT FINDING THAT THE LICENSE AMENDMENTS WERE ADEQUATE TO PROTECT PUBLIC HEALTH AND SAFETY.

A. The NRC's 2006 License Amendment Decision Conditioned the Three-Year Extension of the Unit 1 Operating License on Specific Requirements for PG&E's Reactor Vessel Surveillance Program.

In 2006, following the analytical method set forth in SECY-98-296, the NRC Staff assessed the "impact" of the requested license extension on PG&E's surveillance program for the Unit 1 pressure vessel. 2-ER-163, 2-ER-165. As a result of that evaluation, in order to "provide[] adequate protection to the health and safety of the public" for the enlarged term, the Staff required that: (1) PG&E's program be upgraded from ASTM E185-70 to ASTM E185-82; (2) consistent with that upgrade, PG&E's program would be increased from three to four capsules, of which Capsule B was the last; and (3) also consistent with that upgrade, the projected withdrawal for Capsule B was amended to 20.7 EFPY (approximately 2009). *See* Section E.2 above.

By citing these explicit elements of the reactor surveillance program to justify the three-year extension of the Unit 1 operating license term, the NRC met the two-pronged test that established them as conditions of the Unit 1 operating license. First, the NRC relied on the elements of PG&E's reactor vessel surveillance program to support a license amendment that would grant "greater operating authority" to PG&E, *i.e.,* the authority to operate Unit 1 beyond 2021 to 2024. *In re Three Mile Island Alert,* 771 F.2d 720, 729 (3d Cir. 1985). *See also Citizens Awareness Network* , 59 F.32d at 295 (emphasis in original) (license amendment "undeniably *supplemented* [PG&E's] operating authority.").

Second, by establishing specific new surveillance requirements that must be carried out as a condition precedent to the extended operation permitted by the 2006 license amendment, the NRC "altered the original terms" of the operating license." *Deukmejian v. NRC,* 751 F.2d 1287, 1314 (D.C. Cir. 1984). *See also Union of Concerned Scientists v. NRC,* 711 F.2d 370, 382 (D.C. Cir. 1983) (holding that NRC amended reactor licenses by changing the "binding substantive norms.").

B. The NRC Has Repeatedly Amended the License Condition Imposed on PG&E by the 2006 License Amendment.

In four separate Exemption Decisions issued since 2006 -- in 2008, 2010, 2012, and 2023 -- the NRC Staff has amended the license condition imposed on PG&E by the 2006 License Amendment as a safety-based predicate for extending Unit 1's operating license term by three years past its 2021 expiration date. Without even acknowledging the license condition imposed in 2006, these Decisions have effectively discarded it by (1) extending the scheduled date for removal of Capsule B; (2) declaring that the applicable ASTM E standard was ASTM E 185-70 rather than the updated ASTM E 185-82; (3) declaring that PG&E's surveillance program was a three-capsule program instead of a four-capsule program, and (4) asserting that PG&E's surveillance program was completed with the removal of Capsule V in 2002.[16]

Fifteen years after the 2006 License Amendment decision, nothing remains of the license condition. The only part of the License Amendment that has any recognized effect is that PG&E has continued to operate the Unit 1 reactor for years past the pre-2006 expiration date of 2021, now unencumbered by the safety requirements on which that extension was based. Thus, the Exemption Decisions has "supplemented" PG&E's "operating authority." *Citizens Awareness Network,* 59 F.3d at 295.

C. The NRC Violated the Atomic Energy Act by Failing to Provide Public Notice or a Hearing Opportunity Each Time It Extended the Schedule for Withdrawal of Capsule B from the Unit 1 Pressure Vessel.

Before amending an operating license, the NRC must comply with the requirements of Section 189a of the Atomic Energy Act to provide public notice and the opportunity to request a hearing. 42 U.S.C. § 2239(a). *Citizens Awareness Network,* 59 F.32d at 295. The NRC violated this statutory mandate

[16] *See also* discussion above in Sections F.1 through F.4.

by failing to provide any public notice of the four Extension Decisions or to offer the public an opportunity to be heard.

D. The NRC Violated the Atomic Energy Act by Failing to Evaluate Whether Changes to PG&E's License Condition Would Provide Adequate Protection to Public Health and Safety.

By approving changes to PG&E's license as amended by the 2006 License Amendment without evaluating how those changes would affect public health and safety, the NRC violated the Atomic Energy Act. As required by 10 C.F.R. § 50.92(a), the NRC's review of license amendment applications must be "guided by the considerations which govern the issuance of initial licenses." These considerations include whether the license amendment will "protect the health and safety of the public." 42 U.S.C. § 2133(b). The NRC failed even to acknowledge the existence of the license condition, let alone address how changing it would affect public health and safety. Therefore, the Decisions are unlawful under the Act.

II. THE NRC VIOLATED THE ATOMIC ENERGY ACT AND THE ADMINISTRATIVE PROCEDURE ACT WHEN IT DENIED PETITIONERS A HEARING ON THE 2023 EXTENSION OF THE SCHEDULE FOR WITHDRAWING CAPSULE B.

Section 189a of the Atomic Energy Act, 42 U.S.C. § 2239(a), requires the NRC to "grant a hearing upon the request of any person whose interest may be affected by the proceeding." As discussed above in Section II.H, Petitioners demonstrated their interest in the proceeding by submitting standing declarations and by setting forth, in a specific and well-supported contention, the facts demonstrating that the NRC had amended PG&E's operating license by granting the 2023 extension and the three extensions preceding it. Nevertheless, the Secretary summarily denied Petitioners' hearing request. 1-ER-005.

The Secretary's Denial Order violated Section 189a of the Act by utterly failing to engage on, or even entertain Petitioners' claims that after granting the 2006 License Amendment, the NRC's decisions granting multiple extensions of the time for removing Capsule B "pivot[ed] sharply away" from the rationale for the 2006 License Amendment, to the point where the Staff "now considers withdrawal of Capsule B a discretionary task that PG&E may undertake on its own schedule." 2-ER-056 – 2-ER-059. The Secretary's decision is unreasonable because it addresses the legal question of whether the Staff impermissibly changed or discarded a license condition by simply parroting the demonstrably unacceptable language on which Petitioners seek a hearing. *Alaska Wilderness Recreation & Tourism,* 67

F.3d at 727. It is also arbitrary and capricious because it fails to consider "whether the decision was based on a consideration of the relevant factors and whether there has been clear error of judgment." *Alaska Ctr. for the Env't v. United States Forest Serv.*, 189 F.3d at 859.

III. THE NRC'S ABANDONMENT, WITHOUT A REASONED EXPLANATION, OF THE SURVEILLANCE PROGRAM IT IMPOSED ON PG&E AS A CONDITION OF EXTENDING THE TERM OF PG&E'S LICENSE WAS UNREASONABLE AND ARBITRARY AND CAPRICIOUS.

A. The Extension Decisions Were Unreasonable.

In all four Extension Decisions at issue, the NRC abandoned a duly-established license condition that it had imposed in 2006 in compliance with the substantive and procedural requirements of the Atomic Energy Act. By failing to even acknowledge the existence or applicability of the 2006 License Amendment, the NRC unlawfully abdicated its statutory duties under the Atomic Energy Act, which require support of decisions with a safety analysis and to provide public notice and a hearing opportunity.

Under Ninth Circuit jurisprudence, agency decisions that are "primarily legal in nature" are entitled less deference than those that are "factual" in nature. *Cal. Ex. rel. Lockyer v. USDA,* 575 F.3d 999, 1011 (9th Cir. 2009) (citing *Northcoast Environmental Center v. Glickman*, 136 F.3d at 667 and *Alaska Wilderness Recreation & Tourism,* 67 F.3d at). The NRC's unexplained abdication of the agency's statutory duty must be rejected because it is unreasonable.

B. The Extension Decisions Were Arbitrary and Capricious.

The Administrative Procedure Act invalidates agency actions that are "arbitrary and capricious." 5 U.S.C. § 706(2)(A). A long line of Supreme Court and lower federal court cases have concluded that, while an agency can change its legal position, "an agency must provide a reasoned explanation for any failure to adhere to its own precedents." *Hatch v. FERC*, 654 F.2d 825, 834 (D.C. Cir. 1981). *See also Encino Motorcars, LLC v. Navarro,* 579 U.S. 211, 22 (2017) (citing *FCC v. Fox Television Stations, Inc.,* 556 U.S. 502, (2009)). (agency must present "a reasoned explanation" for "disregarding facts and circumstances that were engendered by [a] prior policy."); *Motor Vehicle Manufacturers Association v. State Farm Mutual Automobile Insurance Co.,* 463 U.S. 29, 42 (1983) ("Accordingly, an agency changing its course by rescinding a rule is obligated to supply a reasoned analysis for the change beyond that which may be required when an agency does not act in the first instance.")). *See also Honeywell Int'l v. U.S. Nuclear Reg. Comm'n,* 628 F.23d 568, 578 (D.C. Cir. 2010) and *Guard v. United States Nuclear Regulatory Com.,* 753 F.2d 1144 (D.C. Cir. 1985) (NRC decisions held arbitrary and capricious for ignoring previous NRC positions without reasoned explanation).

Here, the NRC's action falls plainly afoul of this long line of cases. Without so much as acknowledgement, let alone a reasoned explanation, the agency arbitrarily and abruptly reversed position – on multiple occasions. In its 2006 license amendment, the agency conditioned a three-year extension of Unit 1's license on PG&E implementing the ASTM E 185-82 four-capsule surveillance program, including removal of Capsule B in approximately 2009; then the agency granted four separate extensions of the deadline for removing Capsule B over a period of 15 years, each one based on the NRC's assertion that the three-capsule surveillance program that had been supplanted in 2006 was the applicable program, and that it had been completed. The NRC offered no reason for its change of position: indeed, it did not even acknowledge that it had made any change.

CONCLUSION AND REQUEST FOR RELIEF

For the foregoing reasons, Petitioners respectfully request the Court to declare that the 2023 Extension Decision and the three preceding Extension Decisions constituted unlawfully issued amendments or revocations of the license condition imposed by the NRC in 2006 and reverse and vacate them. In addition, Petitioners request the Court to order the Commission to grant a hearing on whether it should have issued the 2023 Extension Decision or any of the previous Extension Decisions leading up to it. Finally, because these license amendments have cumulatively allowed PG&E to operate Unit 1 in violation of the license condition on which extended operation past 2021 is predicated, the Court should order the Commission to expedite the hearing and any other response to the Court's decision that may be required.

Respectfully submitted,

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March 20, 2024
Corrected March 25, 2024

PETITIONERS' CERTIFICATE OF COMPLIANCE

Pursuant to Fed. R. App. P. 32(a)(7)(C), the undersigned hereby certifies that this brief complies with the type-volume limitations of Fed. R. App. P. 32(a)(7)(B)(i) and Rule 29(a)(f).

1. Exclusive of the exempted portion of the brief provided in Fed. R. App. P. 32(a)(7)(B), the brief contains 10,041 words.

2. The brief has been prepared in proportionally spaced typeface using Microsoft Word in 14-point Times New Roman font. As permitted by Fed. R. App. P. 32(a)(7)(B), the undersigned has relied upon the word count feature of this word processing system in preparing this certificate.

Respectfully submitted,

/s/ Diane Curran
Diane Curran

March 25, 2024

PETITIONERS' CERTIFICATE REGARDING IDENTICAL COPIES

Undersigned counsel certifies that the contents of the paper copies of Petitioners' brief are identical to the contents of the corrected copies of the brief that Petitioners filed electronically on March 25, 2024.

Respectfully submitted,

/s/ Diane Curran
Diane Curran

March 25, 2024

DCISC

DIABLO CANYON INDEPENDENT SAFETY COMMITTEE

VIA EMAIL ONLY
macdonald@berkeley.edu

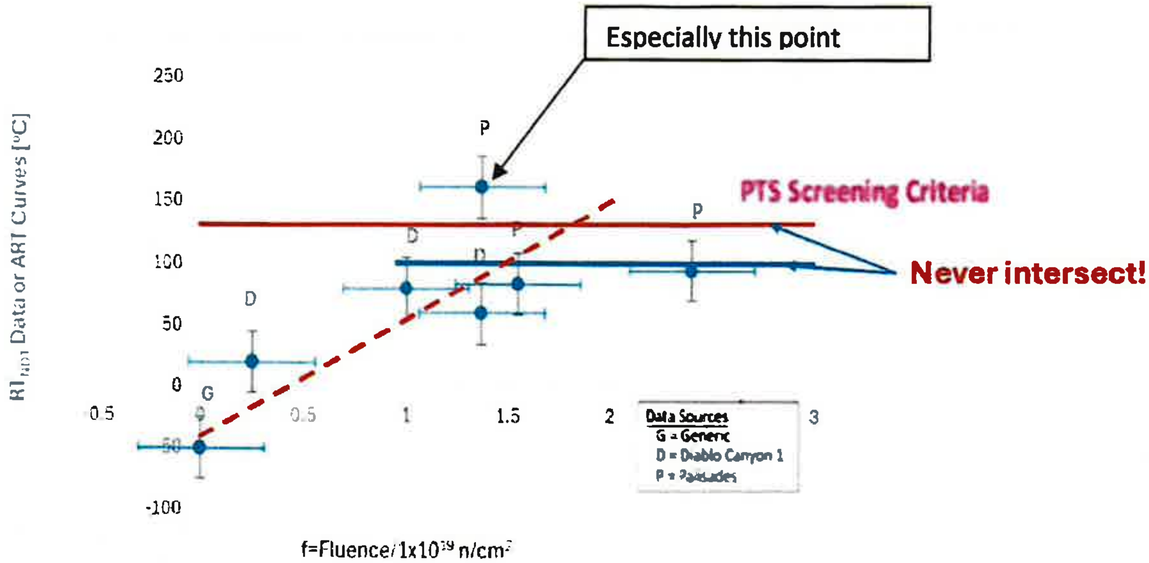
June 28, 2024

Dear Dr. Macdonald:

On behalf of the Members and Technical Consultants of the Diablo Canyon Independent Safety Committee please receive our thanks for your participation on behalf of San Luis Obispo Mothers for Peace, Friends of the Earth, and Environmental Working Group during the Committee's public meeting on June 20, 2024. Following the exchange on Thursday evening the Committee is now preparing to proceed to complete a more detailed review of your "Technical Evaluation of Reports by Dr. Mark Kirk Regarding Condition of Diablo Canyon Unit 1 Reactor Pressure Vessel" provided to the Committee on your behalf by Ms. Curran on June 14, 2024. To assist in this effort the Members have asked me to convey the following questions to you concerning your Evaluation and comments made last Thursday evening. As the Committee and Dr. Kirk complete their review it is possible they may have some additional questions which I will promptly convey to you.

1. Concerning your graph, an image of which appears on the next page:
   a. Please provide data sources for each datum plotted as well as the underlying properties (copper content, nickel content, unirradiated RT_{NDT}, measured ΔT_{41J}, plant of origin, fluence, irradiation temperature). Please also describe the basis for uncertainty bands for both fluence and ART.
   b. Does the point labeled "especially" come from the Palisades reactor capsule or some other place? We have reviewed your citation for this datum that appeared on page 6 of your June 14, 2024 document sent to the DCISC (i.e., the NRC's regulatory analysis that led to RG1.99-Rev 2, ADAMS ML102310298). We believe we found the basis for this datum on page 45 of that document. In a table on that page a value of RT_{PTS} at EOL for Palisades estimated using RG1.99R2 is given as 322 °F (or 161 °C, which corresponds to your plot). Is this the source for the datum you plotted? If so, are you aware that this value is a predicted value; it is not a measurement (all the other data points that appear on your graph are measurements)? Please let us know what justifies the inclusion of one predicted value along with five surveillance measurements (three from Diablo Canyon Unit 1 and two from Palisades) as part of your analysis?

c. Please explain the basis for the uncertainty bands shown in your figure. Also, please clarify whether the plotted uncertainty bands represent one standard deviation or some other measure.



d. Embrittlement data typically show a leveling off of the increase of ART with increasing fluence. Your figure shows a straight line. Was the line in your plot fit using the method of least squares, or by some other method? Also, please describe the basis for assuming that the variation of ART with fluence is linear for the Diablo Canyon Unit 1 weld.

2. Concerning your "extent of embrittlement" analysis methodology:
   a. Please provide evidence that the "Extent of Embrittlement" methodology proposed has been peer reviewed as a suitable metric for embrittlement either in a peer-reviewed journal or by scientific bodies such as the American Society for Testing and Materials and the American Society of Mechanical Engineers.
   b. Please provide empirical evidence that either EoE or $RT_{NDT,PoI}$ correlates better with the true transition temperature of RPV steels, as defined by a T_0 value estimated from fracture toughness tests performed according to ASTM E1921, than does T_{41J}.
   c. What are the acceptance criteria for "extent of embrittlement" and who developed and approved them?
3. Concerning hydrogen induced cracking:
   a. Please provide evidence that US nuclear plants have exhibited Hydrogen Induced Cracking and that this cracking is of sufficient extent to affect the integrity of the reactor pressure vessel during either routine operations or under postulated accident conditions.
   b. Please show evidence that the Unit 1 reactor pressure vessel contains cold worked 316L stainless steel, alloy 600, or alloy 182.
4. Concerning surveillance
   a. Please show where ASTM E-185-70 states that 5 capsules are required.
   b. Please provide documentation showing in the record where PG&E violated NRC's required capsule withdrawal schedule without NRC's approval of extensions.

5. General questions
   a. What materials are VVER Russian reactor pressure vessels made of? Are they similar to Diablo Canyon Unit 1? Do they experience neutron irradiation embrittlement by the same mechanisms as RPV steels made in the USA?
   b. What credit do you give industry wide data from all US operating and worldwide reactors in terms of embrittlement to support US NRC and ASTM data supporting understanding of the effect of radiation damage to reactor pressure vessel steels in the development of USE and PTS limits?
   c. On slide 10 of your presentation you suggest that K_{Ic} tests should be performed, we assume following ASTM Standard E399. Do you suggest to perform these in the irradiated or un-irradiated condition? Please describe how many tests you suggest performing and how the specimen size requirements of Standard E399 will be satisfied.

Should you wish to review the remarks you made to the Committee on Thursday evening you may access the video at https://slo-span.org/meeting/dcisc_20240620/ and review your presentation and the comments on it on the video between 5:16 and 5:38 on the footage counter followed by the subsequent public comments and DCISC committee discussion on embrittlement that went on for another hour-plus thereafter.

To enable the DCISC Members and Technical Consultants to have an adequate opportunity to thoroughly evaluate and consider the Evaluation and your responses to these questions prior to the next public meeting of the Committee on October 9-10, 2024, it is respectfully requested that written responses to these questions be received by email to info@dcisc.org on or before Thursday, August 1, 2024. I close by thanking you for your courtesy and cooperation in this matter and I hope to soon receive your reply. Please do not hesitate to contact me should you have any questions concerning the Committee's request.

Sincerely,

Robert W. Rathie
DCISC Assistant Legal Counsel

cc:
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DCISC Members and Technical Consultants

Technical Evaluation of Reports by Dr. Mark Kirk Regarding Condition of Diablo Canyon Unit 1 Reactor Pressure Vessel

By

Dr. Digby Macdonald[1]

June 14, 2024

Perhaps there has been no more contentious issue surrounding the fate of the Diablo Canyon Unit 1 (DCPP-1) reactor than the question now posed by the Diablo Canyon Independent Safety Committee (DCISC) as to whether the 2011 Westinghouse Pressurized Thermal Shock (PTS) Evaluation Report (WCAP-17315-NP, Revision 0 (2011)) was performed per NRC's regulations and procedures, and whether Dr. Mark Kirk is correct in endorsing those procedures. (Kirk Report (2024)). The subject is very complex, appearing to involve misinterpretation and misreading of the rules and some stretching of the imagination by which NRC procedures seem to be twisted to conform to desired mathematical outcomes. I shall assess where the errors have been made procedurally and mathematically. It is also apparent that Dr. Kirk reviewed his findings with PG&E and Westinghouse for technical accuracy while we were not given such an opportunity[2].

Although the DCISC has stated that it is no longer interested in historical documents, there is no way to evaluate such complex problems without considering the historical context, including prior evaluations that are omitted from Dr. Kirk's Report (Kirk Report (2024)) and misstatements he makes in the process of reassuring the public that the evaluation and safety measures incorporate "conservatisms," when in fact he has omitted key facts from his evaluation. He may argue that the Westinghouse PTS Evaluation Report (WCAP-17315-NP (2011)) stands alone because consideration of a new subset of data clears the slate of extraneous and sometimes inconvenient details regarding embrittlement, but, still, I would suggest this is more a result "of not wanting to see" the problem.

A key concern is Dr. Kirk's failure to mention the NRC's July 2006 letter (NRC letter (2006)), in which it is clear that based upon the surveillance data for DCPP-1 as of that date, with at least two sets of credible data per limiting material as required by (RG 1.99, Revision 2 (1988)), PG&E and the NRC both independently concluded that the projected RT_{PTS} for the most limiting weld (3-442C, Heat # 27204) would reach 258.8 °F by November of 2024. How is it possible that Dr Kirk doesn't even cite that important conclusion predating the Westinghouse Report by only five years? How is it possible that the Westinghouse PTS Report (WCAP-17315-NP, Revision 0 (2011)) incorrectly states that the end of the current license in 2024 corresponds to 32 effective full-power years when the 2006 NRC license approval clearly states this is 35.2 EFPY. Note that an RT_{PTS} of 258.8 °F is indistinguishable from the PTS screening limit of 270 °F when error bars of ± 28 °F are assigned to each data point (as shown in this report, Figure 1).

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1 **DISCLAIMER**. The author prepared this document on behalf of San Luis Obispo Mothers for Peace (SLOMFP), Friends of the Earth (FoE), and Environmental Working Group (EWG), while occupying a position at the University of California at Berkeley. The author makes no warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information or data disclosed herein. The views and opinions expressed in this document are those of the author.

2. Documentation (Cover Letter to the Kirk Report from Kadak Associates) shows that Dr. Kirk extensively reviewed his report with PG&E and Westinghouse for accuracy. This courtesy was not extended to me, SLOMFP, FoE, or EWG.

Because the 2006 summary of the NRC and PG&E calculations (NRC letter (2006)) doesn't show mathematical detail, it is hard to ascertain whether the surveillance data were credible and what margins were used. There is a bit of waffling between the use of Position 1 or Position 2 of (RG 1.99, Revision 2 (1988)) in the reports before 2006, and whether the data are judged to be "credible" is directly related to whether they fall within ± 1σ (standard deviation) about the embrittlement trend correlation (ETC), which directly affects the use of the margin (M) constant in the equations. One of the major features of the DCPP-1 surveillance data from Capsules S, Y, and V has been their credibility as determined by (RG 1.99, Revision 2 (1988)).

Given that the NRC granted a 37-month license extension from September 2021 to Nov. 2024 based on a well-corroborated calculation (NRC letter (2006)), it appears that PG&E and the NRC both assumed that the pre-2004 Charpy test data (ΔRT_{NDT}) were correct and "credible" according to (RG 1.99, Revision 2 (1988)). If this were the case, the lower margin values would have been employed using the correct NRC procedures. There is an apparent fallacy in this argument. All the pre-2004 stress test data results were consistent with mathematical predictions performed per NRC regulations (RG 1.99, Revision 2 (1998)). However, Table D-2 in (Capsule V Report, Appendix D (2003)) indicates that, at that time, one datum for the limiting weld was deemed 'not credible' as it fell 2 °F outside the standard deviation range. It had been deemed credible when it was published in 1992. All the reported results were calculated to be within the allowed uncertainty ranges and were regarded as credible when this report was issued in 2006. It is not plausible that the projected embrittlement values could shift by up to twice the uncertainty ranges in the prior calculations.

As discussed in the "Pressure and Temperature Limits Report (PTLR) for Diablo Canyon" (PG&E Letter DCL-05-016 (2005)), Units 1 and 2 are currently using the same heat-up and cooldown limits, which are based on the limiting-unit surveillance program results. To date, three surveillance capsules have been removed and analyzed from the Unit 1 reactor vessel and four from the Unit 2 reactor vessel. The Unit 1 surveillance results are currently limiting since the calculated delta Reference Temperature for Nondestructive Testing (ΔRT_{NDT}) data scatter does not fall within the one standard deviation of the predicted data as required by (RG 1.99, Revision 2 (1998)) criteria for 'credible "surveillance data. *Therefore, the DCPP heat-up and cooldown limits established in the PTLR are based on the generic limiting CF values in Tables 1 and 2 of (10 CFR 50.61 (2021)). If the RG 1.99 credibility criteria are met upon future surveillance capsule withdrawal and evaluation, then the RG 1.99 Position C.2 will be utilized using plant-specific surveillance data.* Dr. Kirk must explain why he is not bound by these constraints when selecting the CF values. He believes the 2011 PTS Report (WCAP-17315-NP, Revision 0 (2011)) supersedes all prior analyses, which he dismisses as "old news." My position is that (10 CFR 50.61 (2021)) calls for the licensee to consider a broad range of data before integration.

Regarding the number of Capsules that should have been irradiated during the DCPP-1 surveillance program, I note:

- According to (ASTM E185-70), the surveillance program at DCPP-1 should have involved five or more capsules that would be irradiated in the reactor to yield fluences up to the end of life (EoL), *i.e.,* the end of the initial licensing period.
- Two key reasons underscore the necessity for five or more capsules. First, the shift in the adjusted reference temperature (ΔRT_{NDT}) at the vessel inside surface was expected to exceed 200 °F due to high nickel and copper impurities in Weld Heat # 27204 (Westinghouse warned PG&E that such a problem existed). Second, a minimum of five ΔRT_{NDT} vs. fluence data are crucial to practically assure a satisfactory optimization of the

model described in (RG 1.99, Revision 2 (1988)). These data allow for predicting ΔRT_{NDT} (i.e., RT_{PTS}) at higher fluences and, hence, the fluence (reactor age) at which it might exceed the PTS screening limit.

- Interestingly, PG&E stated that the original surveillance program comprised eight capsules, three of which contained limiting weld metal (Heat # 27204) and five containing limiting base metal (PG&E Letter DCL-05-016 (2005)); it is surprising that PG&E, at NRC's insistence, did not replace the capsules containing limiting base metal as soon as it became apparent that the welds were the limiting beltline materials. The cost of irradiation time would have been minimal.
- In recognition of the presence of faulty welds, PG&E installed increasingly lower neutron leakage cores to decrease the rate of radiation embrittlement of the beltline welds (PG&E Letter DCL-05-016 (2005)) to ensure that RT_{PTS} does not exceed the PTS screening limit of 270 °F by the EoL. This assumption is a "first line of defense" that can be followed by more extreme measures, such as shielding the welds with stainless steel plates.
- Any capsules, like Capsule B, intended to be irradiated into or above the 20-year life extension are in addition to the five or more surveillance capsules. In 2006, the NRC clarified that Capsule B was for the initial OL period.
- Capsule B was inserted upon removal of Capsule Y at 5.86 surveillance EFPYs and was scheduled to be removed at 20.7 EFPYs as the fourth capsule in a four-capsule surveillance program as a condition for NRC granting the 37-month life extension to recapture the initial time of Low Power Operation (LPOL) (Amendment No. 188, License No. DPR-80 (2006)). However, due to the inability or unwillingness of PG&E to extract the capsule over the past 16 years, the fluence is now about $3.6\text{x}10^{19}$ n/cm^2. That high fluence renders Capsule B unhelpful for directly assessing weld embrittlement during the initial licensing process (fluence of $1.2\text{x}10^{19}$ n/cm^2). Still, it may have some utility for the extended life to 60 years (fluence of $2\text{x}10^{19}$ n/cm^2). Capsule B must be extracted and processed immediately because the greater the fluence over that of Capsules S, Y, and V, the less accurate the back interpolation (BI) will likely be. See discussion below.
- In performing the Charpy tests, I recommend that the temperature steps between the lower shelf energy (LSE) and the upper shelf energy (USE) in determining the fracture energy (FE) vs temperature (T) be tailored such that four temperatures are in the USE region so that the USE will be clearly defined. The FE vs. T data should then fit with the normal hyperbolic tangent curve, and the fitting parameters should be recorded. Despite past practices, no other graphical processing should be performed to "improve" USE, for example. The USE, RT_{41J}, and ΔRT_{41J} values will be determined per the FE vs. T hypo tangent graphs and (RG 1.99, Revision 2 (1988)), respectively. As noted above, the fluence for Capsule B at that point will be more than $3.6\text{x}10^{19}$ n/cm^2, whereas that of the pressure vessel at the end of the initial licensing period is $1.2\text{x}10^{19}$ n/cm^2.
- To determine RT_{41J} and ΔRT_{41J} at this lower fluence of $1.2\text{x}10^{19}$ n/cm^2, I would use the equations of (RG 1.99, Revision 2 (1988)) and (Odette, G R et al. (1984)) to perform a backward interpolation (BI) using data from Capsules S, Y, and V to anchor the fluence at the low end and Capsule B to anchor the fluence at the high end. Equation (3) or Equation (4) (*see* Appendix I of this Report) will be optimized on these data by minimizing the square root of the sum of the squares of the difference between the function and experiment (i.e., a "least squares" fit) at each point (i.e., the "residuals")). Details of BI are given in

Appendix I. This backward interpolation method has never been reported previously, so I cannot guarantee its success, but the prospects are excellent, and it will serve our purposes.

- Dr. Kirk attempts to address the problem of having only three data points for DCPP-1 (Capsules S, Y and V) by taking two data points, one of which is at a fluence of 2.3×10^{19} n/cm^2, which is significantly higher than that at the EoL of DCPP-1 (1.2×10^{19} n/cm^2), from the Palisades Nuclear Plant (PNP) and indeed greater than that at the end of the 20-year LE to 60 years (2.3×10^{19} n/cm^2) to reconstitute a 5-capsule surveillance program for DCPP-1. The critical points of Kirk's argument for using PNP data are:

  (1) "Sister" plant criteria do not need to be met. Note that the "sister" plant is an EPRI term not defined in 10CFR50, Appendix H. Instead, the NRC refers to sharing data by any two plants in conformity with this regulation as an Integrated Surveillance Program (ISP). Dr. Kirk claims this allows him to use PNP data without first demonstrating that PNP and DCPP-1 qualify as ISP partners as per (10 CFR Part 50, Appendix H (1995)). Still, his reasoning is murky and convoluted, at best, but NRC authorization is required by (10 CFR Part 50, Appendix H (1995)) to integrate the data from DCPP-1 and PNP, regardless. I have considered the ISP issue and concluded that DCPP-1 and PNP do not qualify as defined in (10 CFR Part 50, Appendix H (1995)). Accordingly, Dr. Kirk's attempt to integrate the data from the two plants is illegitimate on this basis alone. A summary of my analysis of the "sister" plant issue is contained in Appendix II.

  (2) Dr. Kirk "assures" conservatisms while reducing CF and M, which is counterintuitive when those adjustments seem to have already been made. It is disturbing that ANY adjustments of CF might be made, as CF is supposed to be a material property, like a melting point or density, and is not some adjustable parameter that can be changed at will to obtain a favorable result. Dr. Kirk's analysis is also inconsistent with Generic Letter 92-01 (GL92-01. US NRC (1992)) Case 5, which requires 5 of 6 capsules to be credible to integrate between two plants.

- All calculated and physically measured data must come with "experimental" uncertainties and preferable standard deviations, and these uncertainties must be displayed by error bars along with the data in any graphical form. The errors and standard deviations in the USE data for unirradiated steels have been determined to be 21-38 ft. lb and 18.1 to 23.8 ft. lb for observed and model-calculated data, respectively (Table 4.1 for ART_{NDT}, from (Eason, E D; Wright, J E; and Odette, MCS G R (1998)). For ART_{41J}, the standard deviation is 21-34 °F and 25.9 °F-38 °F for measured and calculated data, respectively. The "uncertainty" in the fluence is estimated at 10-20 % (CAP V Report, Appendix D (2003)), which, presumably, incorporates the uncertainty in whether flux changes due to all power changes are accurately accounted for.
  Neither the reference temperature nor USE nor the fluence are adjustable parameters, so they should not be treated as such to achieve a favorable result.

- In Figure 1 below, I have incorporated error bars into Dr. Kirk's plot of RT_{NDT} vs. fluence in his Figure 1 (Kirk Report (2024)). It will be noted that including the errors renders the DCPP-1 data essentially inadequate to assess where the locus of RT_{NDT} vs. fluence might intersect the PTS screening limit. Still, it will likely be within 1.2×10^{19} and 1.7×10^{19} n/cm^2 (just into the extended, 20-year operating period).

- A significant accomplishment of my work is to address the irrational definition of the transition temperature as being at a fracture energy of 30 ft. lb. (41J), which was first proposed by ASME and then accepted uncritically by the NRC and Dr. Kirk. In

Appendix III, Figure III.1--of this report-- it is seen that a negative shift in RT_{NDT} is produced by neutron irradiation, an impossible result. The problem is traced to the very definition of the transition temperature itself, originally formulated arbitrarily by ASME. A standard practice in analyzing Charpy impact fracture energy (FE) vs. temperature (T) data is to optimize a hyperbolic tangent function on the data. Thus, from the optimized Equation (1), I analytically calculate all definitions of the transition temperature and the Extent of Embrittlement from the following equations.

$$FE = A + Btanh\left(\frac{T-T_0}{C}\right) \tag{1}$$

where FE is the fracture energy at temperature T and A. B. C, and T_0 is the optimized constants obtained by least squares optimization of Equation (1) on the experimental (FE, T) Charpy impact data. From Equation (1), I find:

$$\text{USE} = A + B \tag{2}$$

$$\text{LSE} = A - B \tag{3}$$

$$FE_{PoI} = A \tag{4}$$

$$RT_{NDT,PoI} = T_0 \tag{5}$$

$$RT_{NDT,30} = T_0 + 0.5Cln\left[\frac{B-A+3}{B+A-30}\right] = RT_{NDT,41} \tag{6}$$

$$RT_{NDT,30} = \Delta T_{41J} = RT_{NDT,PoI} + 0.5Cln\left[\frac{B-A+3}{B+A-3}\right] = RT_{NDT,41J} \tag{7}$$

and

$$\text{EoE} = \left(1 + \frac{e^x - e^{-x}}{e^x + e^{-x}}\right)/2\text{ , where } x = \frac{RT_{NDTPOI} - T_0}{C} \tag{8}$$

Appendix IV illustrates how these equations can be applied to calculate transition temperatures, namely $RT_{NDT,PoI}$, the USE, and the Extent of Embrittlement (EoE), which Dr. Kirk has not achieved. For example, at the POI, the fracture surface is predicted to comprise equal amounts of ductile and brittle facets, thereby removing one variable from the analysis. The distinct advantage of the definition of the reference temperature as $RT_{NDT,PoI}$ is that it is not arbitrary but is well-based in fracture theory, and it avoids the issue identified by Dr. Macdonald. In contrast, Dr. Kirk's RT_{41J} definition achieves none of these.

- Although the reference temperature employed in assessing reference temperature shifts in the data from Capsules S, Y, and V used the conventional ASME definition, RT_{41J}, as does Dr. Kirk, a plot of the temperature shift in this quantity vs. fluence (Eason, E D; Wright, J E; and Odette, MCS G R (1998)) within experimental uncertainty describes a straight (broken, red) line that intersects the PST limit between a fluence of about 1.2×10^{19} n/cm^2 and about 2.4×10^{19} n/cm^2 (Figure 1) indicating that DCPP-1 will achieve an unacceptable state of radiation embrittlement to the limiting beltline welds sometime during the first 20-

year life extension. This analysis includes the three PNP datum points shown on the graph. However, removing these three PNP points minimizes the predictions.

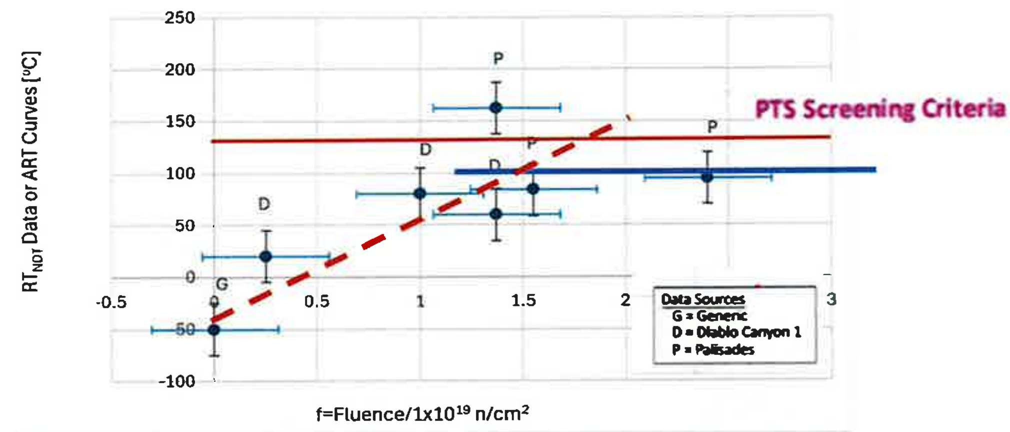


Figure 1. Summary of DCPP-1 RT_{NDT} vs. fluence data along with similar data integrated without NRC authorization by Dr. Kirk from the PNP, in addition to a data point for PNP from and (RG 1.99, Revision 2 (1988)), with each point being assigned error bars as per (Eason, E D; Wright, J E; and Odette, MCS G R (1998)).

- Dr. Kirk's addition of the two PNP data (those below the screening limit) has the effect of "pulling down" the trajectory of RT_{NDT} vs. fluence. In his presentation, datum points without error bars (as should be considered (Eason, E D; Wright, J E; and Odette, MCS G R (1998)) and the data from (RG 1.99, Revision 2 (1988))) from Diablo Canyon (D) and Palisades (P) were used. The absence of error bars and the omission of a datum point from Palisades and above the PTS screening line (red line) results in RT_{NDT} vs. fluence never intersecting the PST screening limit (red line). In other words, Dr. Kirk claims that DCPP-1 is safe to operate forever. However, the PNP data employed by Dr. Kirk appear unusually low, and Dr. Kirk seems to have "cherry-picked" his data because he omitted the high point (point above the red line) from the "1987 analysis of RPVs" that accompanied the publication of RG1.99 R2 (RG 1.99, Revision 2 (1988)). My best fit to the available data, including the two highest fluence data from PNP, is represented by the broken red line, and it indicates that the PTS screening limit will be met at a fluence ranging from about 1.2x10^{19} n/cm^2 and about 2.4x10^{19} n/cm^2. Appendix V discusses my assessment of the lack of conservatism in Dr. Kirk's analysis.
-

For the above reasons, it should be evident to everyone that the present status of DCPP-1 is uncertain, as we currently have no limiting weld embrittlement data. Capsule V was pulled in 2003, and it was noticed that the status of DCPP-1 is unconfirmed by the Westinghouse PTS Report (WCAP-17315-NP, Revision 0 (2011)) because of unaccounted uncertainties and the unacceptable lack of reliable data. Furthermore, we have had no ultrasound (UT) examination performed since 2005, as an inspection scheduled for 2015 was not performed. As a result, the Licensee has produced no surveillance data for the past 19 to 21 years. Therefore, it is my professional opinion that the continued operation of Diablo Canyon Unit 1 poses an unreasonable risk to public health and safety, and DCPP-1 should not be in operation.

I am unaware of why the NRC allowed PG&E to skirt every regulation regarding the minimum number of Capsules needed for the surveillance program. (ASTM E185-70) specifies the need to have five or more Capsules at a minimum. PG&E disregarded the 2006 requirement to pull Capsule B in 2009 and failed to pull Capsule B --with the highly questionable excuse that a gravity-held, 2.5-pound stainless steel plug got "stuck" (a unique event in the nuclear power industry). Still, we then learned, from correspondence, that despite receiving a 37-month license extension based on the withdrawal of Capsule B in 2009, PG&E had never truthfully considered Capsule B to be part of the initial license assessment but an opportunity to obtain data in support of life extension. In doing so, they broke a commitment to the NRC to include Capsule B as the fourth surveillance capsule for the initial licensing period at DCPP-1 in exchange for a 37-month life extension in 2006. And every time PG&E sought to delay the removal of Capsule B, the NRC appeared to "rubber-stamped" the requested change in contravention of its own NRC regulations.

Therefore, the immediate shutdown of DCPP-1, is strongly recommended. If PG&E seeks to reopen DCPP-1, it should take the following measures:

a) Withdrawal and analysis of the contents of Capsule B. Given the high fluence of Capsule B, the above-described backward interpolation (BI) method should be used to assess the reference temperature (RT) vs. fluence during the license renewal term, as well as fluence assessments of Capsules C and D (previously withdrawn but not analyzed) to determine if their contents are helpful. All RT assessments should be conducted by an entity independent of PG&E and Westinghouse, according to strict standards of scientific integrity, and all data, analyses, and results should be provided to the public.
b) WOL specimens in Capsules C and D and archived surveillance capsules should be evaluated. (Among the contents of Capsules Y and V are several WOL fracture mechanics specimens that I presume have been archived). WOL specimens are normally used to determine the stress intensity factor at which crack arrest occurs. However, provided that the specimens are still compliant, they may be pulled to failure in a suitable load machine (e.g., an INSTRON load machine) to determine the fracture toughness, K_{IC}. Additionally, mini-C(T) specimens must be machined from the remnants of the archived Charpy specimens from Capsules S (if any are available), Y, and V, from which K_{IC} may be determined.)
c) PG&E should conduct a comprehensive UT inspection of reactor vessel beltline welds. UT inspections are preferably to be performed on a five-year basis. Appendix VI of this report presents my analysis of UT inspections and why they are essential.
d) The data from the 2015 UT inspection of reactor vessel beltline welds should be published, and the embrittlement trend will be evaluated by comparing them with current data.
e) I also ask PG&E to perform nanoindentation (NI) studies on the fractured remnants of the Charpy specimens from Capsules S, Y, and V. (Hosemann et al. (2015)) have previously demonstrated that NI is a valuable method for assessing the state of embrittlement of ferritic steels as has the considerable advantage over various "macro-scale" methods such as the Charpy V-notch impact techniques in that a statistically significant number of measurements may be made on small specimens.

f) A robust re-evaluation of data credibility from Capsules S, Y, and V that fully complies with NRC guidance and scientific principles is needed.
g) PG&E should take any follow-up steps that may be appropriate for finding credibility of the data from Capsules S, Y, and V, including compliance with (10 CFR 50.61 (2021)).
h) PG&E should provide the NRC, the Advisory Committee on Reactor Safeguards (ACRS), and the public with all data and analyses obtained or performed and a description of any remedial steps taken by PG&E to address the condition of the Unit 1 reactor pressure vessel.

By applying these measures, I believe the state of weld embrittlement at DCPP-1 may still be determined.

In addition, due to the severe deficiencies in the methodology used by PG&E, Westinghouse, and NRC to evaluate reference temperatures, I strongly recommend that PWRs, particularly DCPP-1, change from a reference temperature (RT)-based surveillance program to one based on fracture mechanics and volumetric (UT) examination (FM_UT). This new program has several important advantages over the current RT-based program:

- It directly measures the needed quantity, the critical stress-intensity factor for unstable crack growth, K_{IC}.
- It involves well-established technology and techniques.
- It should be no more costly than an RT-based surveillance program and be no more intrusive on reactor operation.
- Correlation with data from the entire PWR fleet is not immediately required (i.e., the Embrittlement Trend Correlation, ETC) because a theoretical correlation between K_{IC} and fluence has yet to be defined (but which undoubtedly exists). At this point, the analysis must be applied on a case-by-case basis.
- It will require careful definition of the acceptable limit, K_{IC}^{crit}, below which K_{IC} should not go. This requirement may be done to a large extent by calculation.
- UT is a well-established technique for detecting and characterizing fissures that might nucleate brittle fractures in an RPV in response to PTS following an LOCA.
- FM-UT is equally applicable for assessing radiation embrittlement during the initial license period and life extension.

Suppose two halves of the Charpy test specimens are from capsules Y and V (and possibly from S). In that case, it should be possible to machine up to twenty mini-C(T) fracture-mechanics specimens of each material (assuming ten Charpy specimens for each capsule). This number is sufficient to determine the "standard deviation" for K_{IC} and K_{IC}^{crit}. Characterizing the magnitude of the error in each K_{IC} is critically important in assessing the state of embrittlement.

Appendix I. Backward Interpolation (BI) for Determining RT_{NDT} at Intermediate Fluence Levels.

If successful, BI could be used to determine the state of embrittlement (RT_{41J}, ART_{41J}, and USE) for any fluence, including that at the end of the first license period ($1.2x10^{19}$ n/cm^2). The only assumption behind BI is that the state of embrittlement, as measured by RT_{41J}, ART_{41J}, or USE, is a continuous function of fluence. In (RG 1.99, Revision 2 (1988)) the equations accurately represent their dependence on fluence,

$$F(f) = f^{[0.4449+0.0597log(f)]} \tag{1}$$

or

$$F(f) = f^{(0.28-0.1logf)} \tag{2}$$

where

$$\Delta RT_{NDT} = CFxF(f) \tag{3}$$

this seems to be true for the 400+ PWRs in the NRC database. Note that the fluence is in units of 10^{19} n/cm^2 and that CF represents the chemistry factor, which accounts for the concentrations of Ni and Cu that redispose the welds to premature embrittlement. Values for CF are given in tabular form in (RG 1.99, Revision 2 (1988)). When using Equation (1) for the fluence factor (Eason, E D; Wright, J E; and Odette, MCS G R (1998)) derived an improved model for the shift in the transition temperature $\Delta RT_{NDT,41J}$ (or ΔT_{41J} in Kirk's terminology) with accumulated fluence as

$$\Delta RT_{NDT,41J} = Aexp\left(\frac{1.906x10^4}{T_C+460}\right)(1 + 57.7P)F(f) + B(1 + 2.56Ni^{1.358})h(Cu)g(f) \tag{4}$$

where the fluence factor is given by Equation (1). The other factors in the more sophisticated model of (Eason, E D; Wright, J E; and Odette, MCS G R (1998)) Equation (4) are given by

$$g(f) = 0.5 + 0.5tanh\left[\frac{log(f+5.48x10^{12}t_i-18.290)}{0.600}\right] \tag{5}$$

and

$$h(Cu) = \left\{\frac{\dfrac{0, Cu \le 0.72\ Wt.\%}{(Cu-0.072)^{0.678}, 0.072<Cu<0.300\ Wt.\%}}{0.367, Cu \ge 0.300\ Wt.\%}\right\} \tag{6}$$

For welds A = 1.10x10-7, B = 209; for plates A = 1.24x10-7, B = 172; for forgings A = 8.98x10-8, B = 135. $\Delta RT_{NDT,41J}$ and Tc (temperature of irradiation) are in °F; Cu, Ni, and P are in wt%; f is in n/cm2 (E > 1 MeV), and ti is in hours. All the non-integer constants in the above equations are fitting parameters determined by the least squares method. The adjusted reference temperature is expressed as:

$$\Delta RT = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \tag{7}$$

where the first term on the right side of Equation (7) is the reference temperature for the unirradiated material, the second is given by Equations (3) or (4), and the Margin is written is

$$margin = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2} \tag{8}$$

In this expression, σ_I is the standard deviation for the initial RT_{NDT}. If a measured value of initial RT_{NDT} for the Material in question is available, all is to be estimated from the precision of the test method. A generic mean value for that material class will be used if it is not. If a suitably large data set is available, it may be used to establish the mean and standard deviation. The standard deviation for ART_{NDT}, σ_Δ, is 29 °F for welds and 17 °F for base metal, except that σ_Δ need not exceed 0.50 times the mean value of ART_{NDT}. The two models differ primarily by how the weld impurities (Cu, Ni, and P), which predispose the welds to radiation embrittlement, are included in the formulations.

Appendix II. Definition of a "Sister" Plant (ISP Partner)

The scarcity of reactor-specific surveillance data from DCPP-1 has led PG&E to argue that the Palisades Nuclear Plant (PNP) is sufficiently similar to DCPP-1 that they may be regarded as "sister" plants (an EPRI term). Dr. Kirk also contends that PNP is a "sister plant" to DCPP-1, but the term doesn't appear anywhere in the regulations, so it is puzzling that he states that they do not need to be qualified by the ISP partner criteria in the regulations. Still, Dr. Kirk offers no convincing evidence to support that claim. Is his claim reasonable? Four important aspects of this issue need to be addressed. First, are the reactors of the same design? Second, are the operating histories of DCPP-1 and Palisades sufficiently alike? Third, are the limiting (weld) materials sufficiently alike in chemical composition (Ni and Cu content) and physical properties (reference temperature, USE, fracture toughness, etc.)? Fourth, how do the fluences compare for the operating periods of the two reactors? A requirement to consider these issues before comparing data from different reactors and a clarification of the criteria for qualifying "sister plants" are explicit in NRC's Fracture Toughness Requirements for Light Water Reactor Vessels in ((10 CFR 50.61 (2021)), (10 CFR 50, Appendix H (1995)), (10CFR, Vol. 60, (1995)), and (10 CFR 50.61 (2021))).

1. In an integrated surveillance program (ISP), the representative materials chosen for surveillance for a reactor are irradiated in one or more other reactors with similar design and operating features. Integrated surveillance programs must be approved by the Director, Office of Nuclear Reactor Regulation, on a case-by-case basis. Criteria for approval include the following:

   a. The reactor in which the materials will be irradiated and the reactor for which the materials are being irradiated must have sufficiently similar design and operating features to permit accurate comparisons of the predicted amount of radiation damage.

   b. Each reactor must have an adequate dosimetry program.

   c. There must be adequate arrangements for data sharing between plants.

   d. A contingency plan must ensure that the surveillance program for each reactor will not be jeopardized by operation at a reduced power level or by an extended outage of another reactor from which data are expected.

   e. Substantial advantages, such as reduced power outages or reduced personnel exposure to radiation, must be gained because surveillance capsules are not required in all reactors in the set.

2. No reduction in the requirements for the number of materials to be irradiated, specimen types, or number of specimens per reactor is permitted.

3. No testing reduction is permitted unless previously authorized by the Director, Office of Nuclear Reactor Regulation.

The scarcity of reactor-specific surveillance data from DCPP-1 has led PG&E to argue that the Palisades Nuclear Plant (PNP) is sufficiently alike to DCPP-1 that they may be regarded as ISP partners. Regulation (10 CFR 50.61 (2021)), which incorporates (RG 1.99, Revision 2 (1988)), is quite specific about the criteria that must be met in assessing whether two PWRs are sufficiently alike that surveillance data obtained on one of the reactors can be used with confidence in determining the state of RPV embrittlement of the other. My professional judgment is that the plants are not alike within the parameters dictated by (10 CFR 50.61 (2021)). If that is the case, the entire method of integrating data from the two plants is called into question.

In my opinion, it is evident that this integration should not be allowed and should not serve as a substitute for more plant-specific surveillance data, as would have been required by (ASTM E185-70) or (ASTM E185-82). As quoted above, (10 CFR 50.61 (2021)) clearly states that integrated surveillance programs are not to be used to reduce the scope of the required surveillance data as dictated by the regulations. Furthermore, Dr. Kirk falsely assumes that the two plants have operated at near full capacity and that temperature adjustments to the data will account for all operational differences, which is not the case.

DCPP-1 and PNP are distinct designs from the outset and have been operated differently throughout their operational lives.

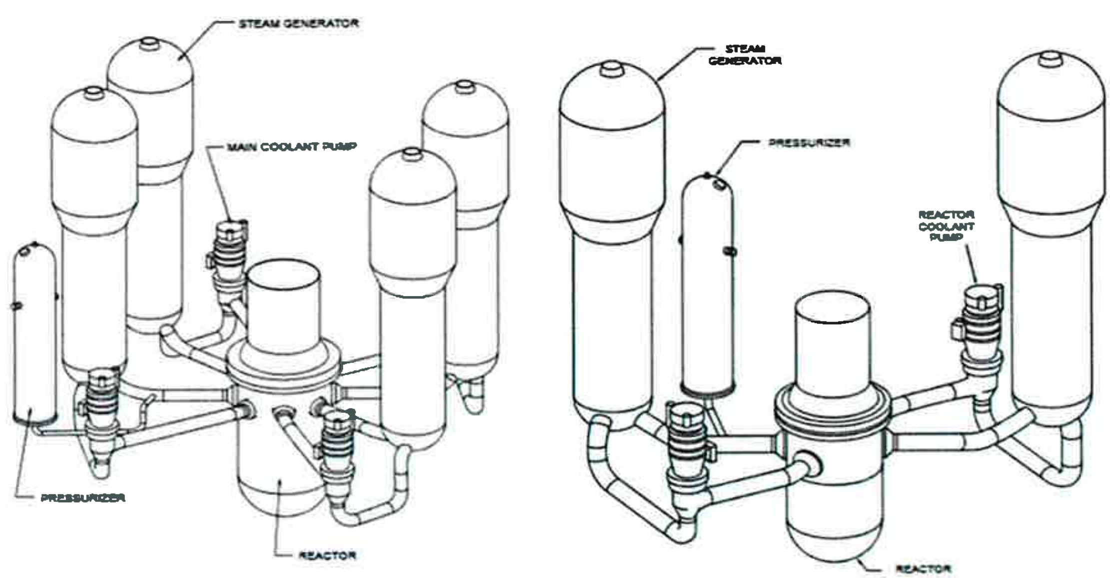


Figure II.1. Schematics of a Westinghouse-designed four-loop Pressurized Water Reactor (PWR, like DCPP-1(left), and a two-loop model like PNP (right)).

The left frame of Figure II.1 shows a schematic of a four-loop Westinghouse power plant, like Diablo Canyon, Unit 1. This design has four steam generators, four reactor coolant pumps, and a pressurizer. The four-loop units in the United States include Braidwood 1 and 2, Byron 1 and 2, Callaway, Catawba 1 and 2, Comanche Peak 1 and 2, D. C. Cook 1 and 2, Diablo Canyon 1 and 2, Indian Point 2 and 3, McGuire 1 and 2, Millstone 3, Salem 1 and 2, Seabrook, Sequoyah 1 and 2, South Texas Project 1 and 2, Vogtle 1 - 4, Watts Bar 1, and Wolf Creek. On the other hand, the right frame of Figure II-1 shows a two-loop Westinghouse plant, like Palisades Nuclear Plant, which has two steam generators, two reactor coolant pumps, and a pressurizer. The two-loop units in the United States are Ginna, Kewaunee, Point Beach 1 and 2, Palisades, and Prairie

Island 1 and 2. Each of these plants has 121 14 x 14 fuel assemblies arranged inside a reactor vessel with an internal diameter of 132 inches. DCPP-1 and PNP do not have similar designs and have operated very differently, allowing them to evolve as unique units in operating practice and experience. According to the criteria in Appendix H, this would suggest they are not good candidates for integrating their surveillance data following the methods prescribed by 10 CFR50.61. The operational differences should be more thoroughly analyzed and not assumed appropriate for integration, especially given the long history of weld material embrittlement at both plants.

In both reactors, the limiting materials are beltline welds near the center of the reactor pressure vessel. In DCPP-1, the limiting weld is an axial weld with weld wire Heat # 27204, which has a screening limit of RT_{PTS} of 270 °F, while in PNP, one of the limiting welds is an axial weld using W5214 material, not the circumferential weld, which is a close second. The circumferential weld utilizes weld-wire heat # 27204 with an RT_{PTS} of 300 °F. The difference arises from the thin-wall cylinder analysis of an RPV, which shows that the axial stress is less for a given internal pressure than the circumferential (hoop) stress. This stress difference is compensated for by assigning a circumferential weld a higher RT_{PTS} (300 °F) than that for an axial weld (270 °F). It is also important to note that axial welds at PNP used a different but compositionally similar weld material (W5214) that was even more embrittled than the PNP circumferential weld. Thus, the Heat # 27204 weld material was not the most limiting material in the case of PNP. This fact is an essential difference between DCPP-1 and PNP. However, it is important to note that the circumferential weld at PNP (9-112) was projected to reach an RT_{PTS} of 280 °F by about 2011, and this RT_{PTS} was significantly higher than that of the axial welds at DCPP-1 using the same # 27204 material. Although there are differences in fluence due to the location of the welds, which are accounted for with transport calculations, the PNP weld 9-112 would not have met embrittlement screening limits under (10 CFR 50.61 (2021)) had it been an axial weld. Palisades was even more embrittled than DCPP-1. As stated previously, these details suggest that the PNP and DCPP-1 capsule sets should not be integrated in a manner that yields such a significant shift in the data. Their embrittlement trends were well understood despite the limited data available.

All complex systems, including nuclear reactors, operate in a way that quickly distinguishes them from their peers. That difference may be due to planned or unplanned operations involving systems at various power levels, such as nuclear reactors. Thus, if two reactors (or any other complex system) are of identical design in all aspects, including configuration, materials, welding, etc., the two quickly evolve as individuals because of different operating histories and conditions alone. Since DCPP-1 and PNP have various designs and operating histories, they do not qualify as "sister" plants. Because Palisades was rapidly trending toward embrittlement, fuel configuration, and power factor changes were significantly changed. Still, most importantly, shielding around some parts of the fuel but not others introduced another variable completely independent of the temperature adjustments, which Dr. Kirk falsely assumes will account for operational differences.

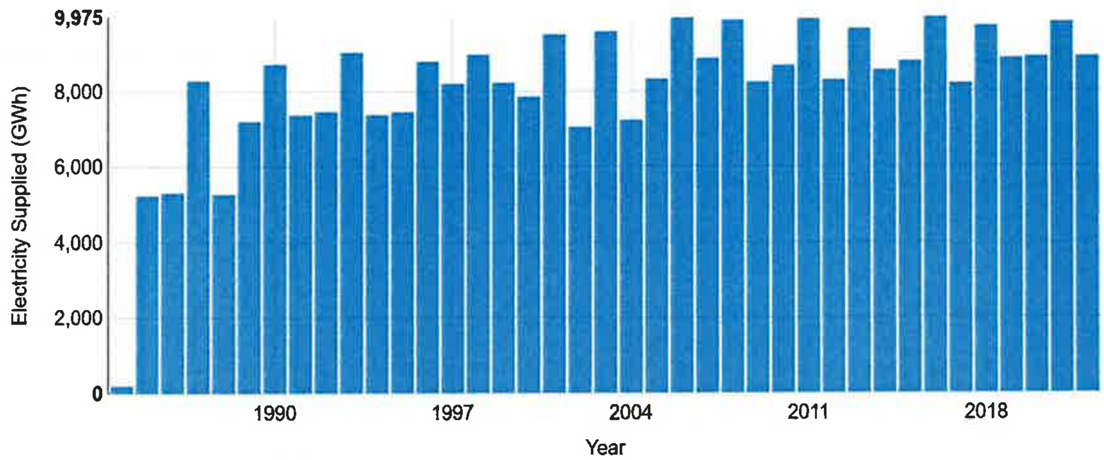


Figure II.2. Electricity supplied by DCPP-1 from start-up in 1984 to 2022.

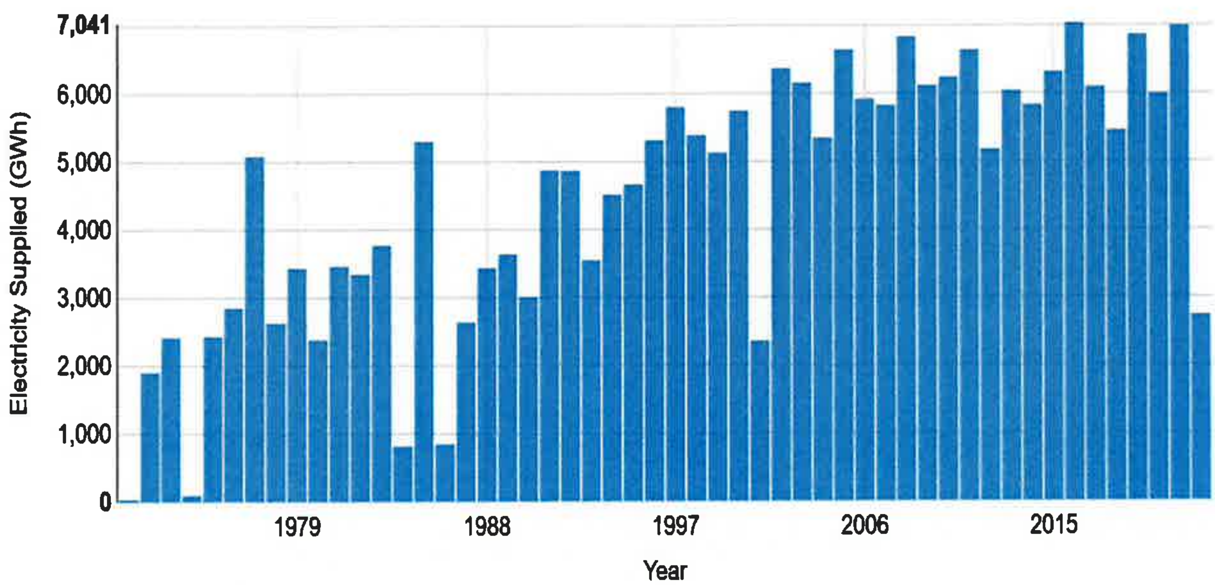


Figure II.3. Electricity supplied by PNP from a start-up in 1972 to shut down in 2020.

Differences in operating history also complicate fluence comparisons between reactors. Figure II.2 and Figure II.3 show the electricity supplied to the DCPP-1 and Palisades NP, respectively, from 1975 to 2020. The load in the case of PNP is erratic, reflecting frequent unplanned outages and excursions, and is quite different from that shown by DCPP-1 in Figure II.2.

Because the electricity supplied by DCPP-1 has remained stable, which is reminiscent of a baseload plant, with only small excursions typical of refueling outages only, the conversion from calendar years of operation to effective full power years (EFPYs) of operation is a factor of 0.8.

On the other hand, for the Palisades reactor in Michigan, that factor is 0.70. The difference here is that for a given calendar time, DCPP-1 accumulates more EFPYs than PNP. Thus, if the high energy neutron flux under full power conditions were the same (an unlikely valid assumption because the flux depends on the design of the core, which is different for DCPP-1 and PNP), DCPP-1 would be expected to accumulate a higher fluence than PNP for a given calendar time. The above argument is simplistic because it assumes full power operation, which is not the case for either reactor but especially for PNP. I examined the Westinghouse fluence reports for both DCPP-1 and PNP. Still, it is unclear whether power excursions and unscheduled outages were considered in the fluence calculations. Dr. Kirk should clarify this critical issue.

Other energy sources statistics show the same pattern as in Figures II.4, II.5, and II.6 below. At this point, it is necessary to note that the two reactors have entirely different designs. Thus, DC PP-1 is a four-loop reactor while PNP is a two-loop reactor model, both of which are Westinghouse designs. The two reactors have different core (fuel) configurations, which yield different high energy fluxes and hence different beltline fluence for a given full power operating time, among other factors. Most importantly, the operating histories of the two reactors are quite different.

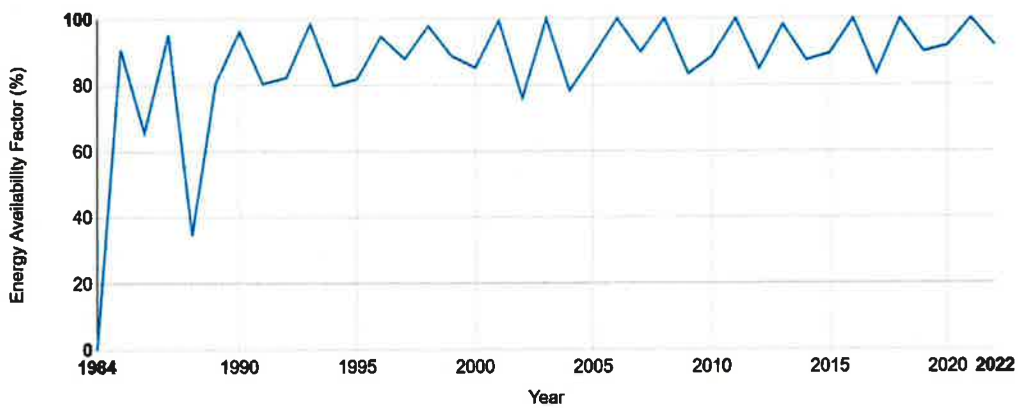


Figure II.4. Energy availability factor for DCPP-1 from startup to 2022.

Because the reactor designs are substantially different, it cannot be assumed that the temperature adjustments capture the differences in fluence, as Dr. Kirk assumes. There doesn't appear to be any analysis of these critical considerations in either the Westinghouse PTS Report (WCAP-17315-NP, Revision 0 (2011)) or the Kirk Report (Kirk Report (2024)) other than Kirk's rather general assertion that it is safe to assume that both plants are sufficiently similar and that they have operated at nearly 100% throughout their operational history, which we know not to be the case. The differences in the plant designs and the impact of plant design differences on their qualification as "sister plants" should be more thoroughly analyzed.

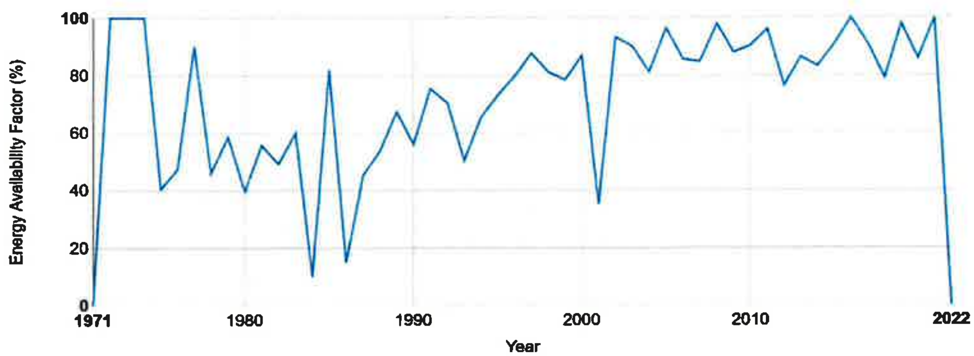


Figure II.5. Energy availability factor for PNP from startup to 2022.

Thus, DCPP-1 has operated as a base load plant, whereas PNP appears to have a much less consistent operational history and has experienced considerable unplanned outages, both intended and unintended.

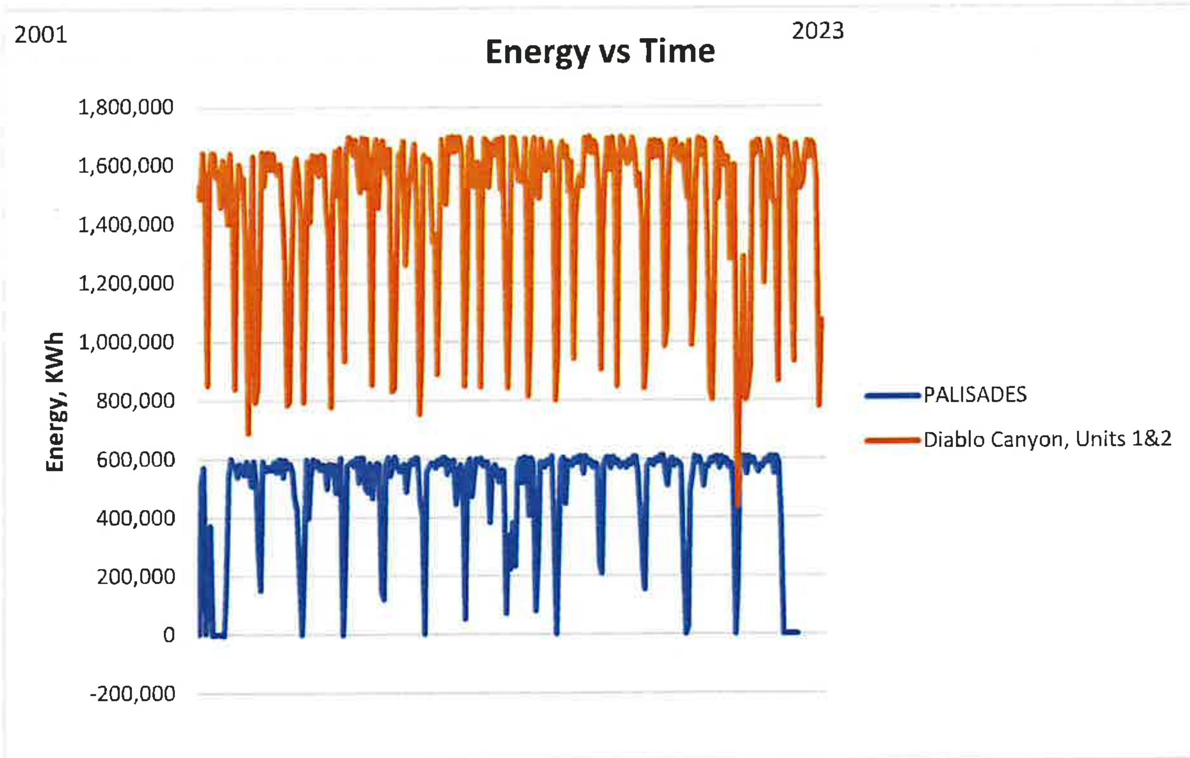


Figure II.6. Comparison of the monthly energy produced by DCPP-1 and PNP over the past ca. twenty-five years. Data from the US Energy Information Administration (EIA).

Other operational conditions between the two plants (PNP and DCPP) are significantly different and not at all similar to what Dr. Kirk's report suggests. Earlier capsule testing at Palisades indicated rapid embrittlement of the RPV welds, and there was concern that the plant would have to close as early as 1999. For this reason, and from 1999 onwards, Palisades implemented an "Ultra Low Leakage Core Strategy" to reduce the flux (radiation exposure) at the critical welds. Doing so involves changing the fuel configuration, reducing generation capacity to 90% (to reduce radiation damage), and adding stainless-steel shielding to protect the most vulnerable welds. It is unclear whether the dosimetry, temperature adjustments, and transport equations performed to integrate the DCPP-1 and PNP data sets accounted for all these variables. Although DCPP-1 operated at 97.9% of its current capacity (3338MWt) until the 10th fuel cycle (about 1992), it has operated at its current 100% capacity (3411MWt) since then. Still, it has employed increasingly lower neutron leakage cores to help maintain $RT_{PTS} < 270$ °F until the EoL (PG&E Letter DCL-05-098 (2005)). Most significantly, DCPP-1 has not employed ultra-low neutron leakage cores or the stainless-steel shielding of its limiting welds as PNP has, and there is no accounting of this in the Westinghouse PTS Evaluation Reports (WCAP-17315-NP, Revision 0 (2011)). From these details alone, one would suspect that the embrittlement at DCPP-1 is worse than the pre-2003 surveillance data indicates. Since the capacity was increased in 1992, and the most limiting weld has not benefited from shielding, the Palisades data raises greater embrittlement concerns than that for the DCPP-1 RPV since the former may be far more embrittled than the earlier PNP data indicate. In any event, the operational differences increase the need to properly qualify PNP as a "sister plant" and account for previously unaccounted variables. Even if the NRC doesn't flag such an oversight, it violates their regulatory requirements as defined in the 1995 update to (10 CFR50 and Appendix H (1995)).

Thus, the above information shows that DCPP-1 has been operated far more consistently than PNP. These data are evidence from the frequent, presumably unplanned power excursions in Figures II.4 and II.6. The data in Figure II.6 demonstrate that the power excursions were not from long-term outages (longer than one month) because the monthly energy production did not reach zero. In contrast, most such excursions in PNP reached zero, qualifying them as outages over which the plant produced no energy. Accordingly, I argue that DCPP-1 and PNP are not sufficiently alike in design or operating experience to qualify as "sister" plants.

Operating Palisades at low power for a significant fraction of its life means that for a given calendar time, the fluence experienced by the beltline components and the extent of RPV embrittlement should be lower for Palisades than for DCPP-1 since the latter operated at "full power" for its entire life except for the period of low power testing in the beginning when the fluxes of high-energy neutrons ($E > 1$ MeV) were probably comparable at the beltlines. Assuming that the beltline neutron fluxes are similar, using data from Palisades to argue that DCPP-1 is not close to embrittlement screening limits is questionable because of the highly embrittled state of the PNP circumferential weld and noting that the axial limiting weld at DCPP-1 is probably at least as embrittled.

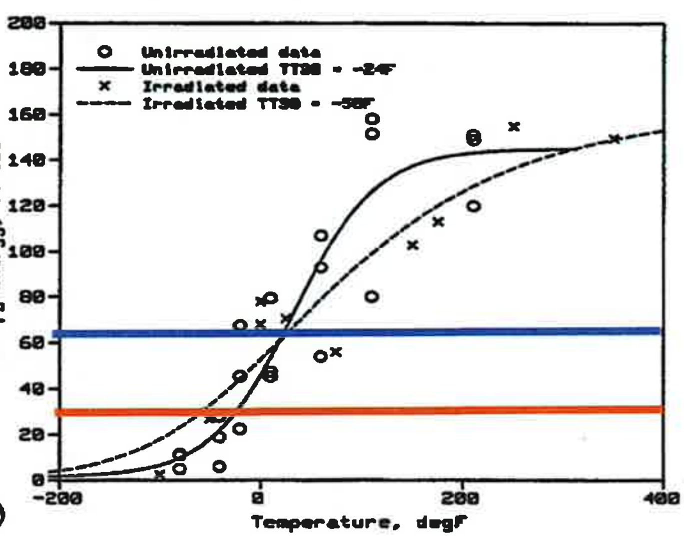
Appendix III. Errors in Experimental and Calculated Data.

All experimentally measured data are characterized by uncertainty (error), and the Reference Temperature and USE obtained from surveillance samples in operating nuclear reactors, such as DCPP-1, are no exception. A competent scientist or engineer pays close attention to the magnitude of the error and the distribution within the error because these factors may determine whether a particular state has been achieved within a level of statistical certainty. Dr. Kirk fails to account for significant uncertainties in assessing fluence, Reference Temperature, and USE in three key respects:

a) Determining whether the use of a standard deviation is appropriate.
b) Establishing appropriate error bars for fluence, Adjusted Reference Temperature (ΔART_{41J}), and upper shelf energy (USE) and displaying those error bars on any data plots.
c) Addressing the fact that the arbitrary definition of the reference temperature at a Charpy fracture energy of 30 ft. lb gives rise to negative shifts in the transition temperature in the example shown in (Eason, E D; Wright, J E; and Odette, MCS G R (1998)) --see Figure III.1--which appears to be non-physical. This is because there is a negative shift in RT_{41J} upon irradiation (orange line), indicating that, somehow, the ductility of the steel has improved upon neutron irradiation, not lessened as predicted theoretically. Note that the RT_{PoI} (blue line) does not indicate this anomaly; instead, it correctly shows a small positive shift and, hence, a slight embrittling of the steel upon neutron irradiation.

Because of these unaccounted-for uncertainties, it is even more concerning that margins and CF values are lowered without clear scientific justification. Dr. Kirk contends that lowering the CF and M values is procedurally correct, but does it stand up in light of other unaccounted variables?

Assuming that the data are normally distributed and that the error bars correspond to $\pm 1\sigma$ for both ART_{41J} and ART_{PTS}, which are typically of the order of 28 °F (Eason, E D; Wright, J E; and Odette, MCS G R (1998)) then the overlap of the two 1σ error bars (one for ART_{41J} and the other for ART_{PTS}) will occur at 210 °F. When the mean value of ART_{41J} is within 20 °F of the mean value of ART_{PTS}, the overlap is such that the probability of failure is essentially one (100%). Dr. Kirk and the NRC have ignored the errors associated with ART_{41J} and ART_{PTS}, even though these errors have been quantified in (Eason, E D; Wright, J E; and Odette, MCS G R (1998)) and consequently, they have missed this critical point.



b)

Figure III.1. Illustration of the effect of neutron irradiation on the mechanical properties of a pressure vessel steel as revealed by Charpy Impact Testing (CIT). The orange line represents RT_{41J} and the blue line RT_{POI}. (Eason, E D; Wright, J E; and Odette, MCS G R (1998)).

The question for Dr. Kirk, then, is: "What probability of fracture via PTS with overlapping error bars (probability of fracture < 1) compared with 100 % probability with overlapping mean values of $\overline{ART_{41J}}$ and $\overline{ART_{PTS}}$ (at which the probability is one) is an unacceptable position?" Is a high probability (say 68 %) with $\overline{ART_{41}} < \overline{ART_{PTS}}$ acceptable, or are you willing to recognize a much lower probability corresponding to $\overline{ART_{41}}$ and $\overline{ART_{PTS}}$ being much further apart? In the latter case, the risk is negligible but still finite.

Further issues on this point must also be addressed, as they relate to accounting for unfactored uncertainties that would argue against reducing margins in a case where impure weld materials are a known factor:

1. The arbitrary definition of the transition temperature at a Charpy fracture energy of 30 ft. lb (41J) gives rise to negative shifts in the transition temperature in the example shown in (Eason, E D; Wright, J E; and Odette, MCS G R (1998)), which appears non-physical, i.e., cannot occur by any known physical means. On the other hand, a slight positive shift occurs in RT_{POI}, for example, seemingly avoiding this non-physical issue. Thus, by using RT_{POI}, we observe a slight increase in embrittlement, but the use of RT_{41J} indicates that embrittlement has decreased upon exposure of the steel to neutron irradiation (E > 1 MeV). Although (Eason, E D; Wright, J E; and Odette, MCS G R (1998)) do not give the conditions of irradiation (or the fluence), the notion that further irradiation should result in a negative shift in RT_{NDT} is nonsensical. Thus, radiation "healing" is unknown to me and should be recognized as an artifact of the arbitrary definition proposed by ASME and adopted uncritically by the NRC.
2. The errors and standard deviations in the USE data for unirradiated steels have been determined to be 21-38 ft. lb and 18.1 to 23.8 ft. lb for observed and model-calculated data, respectively, from (Eason, E D; Wright, J E; and Odette, MCS G R (1998)). These errors should always be considered when analyzing the experimental data.
3. For ART_{41J}, the standard deviation is 21-34 °F and 25.9 °F to 38 °F for measured and calculated data, respectively.
4. The ART_{41J} is found to be only a weak function of fluence, particularly at high fluence where the slope $d(RT_{41J}/d\varphi t)$ is low as RT_{41J} approaches an upper, steady state limit as the fluence $\varphi t \rightarrow \infty$.
5. These errors should always be quoted with data used for regulatory purposes.

(Eason, E D; Wright, J E; and Odette, MCS G R (1998)) reports a rigorous analysis of the transition temperature errors (standard deviations) and the USE (Upper Shelf Energy) for the transition temperature. Since the USE data are not discussed in depth here, the reader is referred to (Odette, G R et al. (1984)) for details. The issues mentioned above are related to reference temperature, and the techniques used directly apply to the USE. However, a comparable probabilistic USE data analysis was not performed because of a lack of time.

The remaining issue defines the exact fluence at which ΔART_{41J} exceeds the prescribed limits 270 °F for longitudinal welds. The predicted ΔART_{41J} and the ΔART_{PTS} [the "Δ" indicating that these are "adjusted" quantities by the inclusion of the "margin"] have associated uncertainties indicated by "error bars." In the Ideal World, the parameter of interest (e.g., ΔART_{41J}) will have been sufficiently replicated to apply rigorous statistical analysis, but we do not live in an "Ideal World." Typically, a capsule in a surveillance program at DCPP-1 yields a single ΔART_{41J} with no associated standard deviation. Still, because ΔART_{PTS} is determined from a database of > 400 reactors, the standard deviation in that quantity is well established. Thus, it is *assumed* that *the same standard deviation characterizes ΔART_{41J}.* For example, these errors incorporate the fluence (f) error. However, the error in f may be substantial because, from its definition $f = \varphi t$, where φ is the neutron flux see changes (n/cm^2.s) and $t(s)$ is the time. From the Westinghouse PTS Report (WCAP-17315-NP, Revision 0 (2011)), it is uncertain how well the errors in the fluence are determined. For example, all reactors shut down every 18 months or so for refueling, and I assume that the fluence is adjusted for those events. But what about unscheduled power excursions and shutdowns that all reactors experience? Dr. Kirk needs to show us how the errors in the flux and time of irradiance are incorporated in his analysis and to provide us with an accurate assessment of the error in the fluence and its impact on RT_{41J}.

Suppose Unit 1 embrittlement projections confirmed by the NRC in 2006 suggested that it was reaching the 270°F screening limit by November of 2024. In that case, it is highly implausible that the reactor could operate safely for another 20 years. Dr. Kirk endorses the Westinghouse PTS Report (WCAP-17315-NP, Revision 0 (2011)) as the embodiment of rigorous NRC procedure while claiming it has built-in "conservatism." Still, he falls short of explaining the details of the mathematics. He doesn't explain it because I suspect that all conservatisms have been removed by lowering both the CF and M factors, i.e., by removing all the conservatisms he claims are built into the formula. If correct, it is hard to imagine that such a misrepresentation is accidental.

While Dr. Kirk dismisses my recalculation methods as unapproved by the NRC, he also employed his method, which he admits is also unapproved by the NRC (apparently now under development with an ASTM committee). Remarkably, Dr. Kirk's recalculation suggests that the upper shelf energy values of DCPP-1 will exceed regulatory limits by year 2029, a result that affirms the accuracy of the pre-2004 surveillance data and indicates that the Unit 1 reactor is neither safe for 20 more years of operation, nor is it a good investment for the state of California. However, he dismisses that even his calculation projected that in 2029, USE would fall well below 50 ft. lb, a critical (10 CFR 50.61 (2021)) screening limit. How can he then state that no threat to the reactor exists and endorse the Westinghouse conclusion that DCPP-1 is safe to operate through 2044? This contradiction in Dr. Kirk's report should be a red flag to the DCISC and a clear sign that the report incorporates misrepresentations and contradictions.

Appendix IV. Redefinition of RT_{NDT}.

Dr. Kirk attempts to denigrate and dismiss my attempt to define a more rational reference temperature than the temperature of the arbitrarily chosen value of the Charpy Impact Fracture Energy of 30ft. lb or 41J. ASME chose this value arbitrarily because it was sufficiently low and had minimal risk of interfering with the USE. However, it has no theoretical basis and even indicates, absurdly, that irradiation may increase ductility (Eason, E D; Wright, J E; and Odette, MCS G R (1998)). His attempt is fundamentally incorrect and displays a lack of understanding of what I did. To begin with, Dr. Kirk and I use the same data (Charpy impact testing) and the same procedure for optimizing the hyperbolic tangent function on those data performed by Westinghouse ((Capsule V Report (2003)), (WCAP-15958 (2003))). The only difference is that I extract the final results analytically rather than empirically and define the transition temperature at the point of the inflection of the hyperbolic tangent function used to represent Fracture Energy vs. temperature data. The fracture displays equal amounts of ductile and brittle facets at the inflection point.

In his "Figure 28," reproduced below (Figure IV.1), Dr. Kirk presents two graphs showing that my approach is wrong or inferior. He makes this determination from the wider scatter band of the data shown in Figure IV.1, but that impression can be corrected by simply changing the scale on the vertical axis. Furthermore, the parameter that should have been plotted is $RT_{NDT,PoI}$ compared with T_{41J} rather than the extent of embrittlement (*EoE*), a derived quantity with no equivalence in Dr. Kirk's formulation. That would have allowed him to make a legitimate link to Fracture Toughness through his T_0. He may not have appreciated that my T_0 ($RT_{NDT,PoI}$) is not the same quantity as his.

As the reader can note, in Figure IV.1, the comparison is between two different curves; on the left side, it is a curve of T_o (units of degrees) versus *EoE*, while the right frame presents a plot of his T_0 (not indicated, we assume it is this one) versus T_{41J} (units of °C). The reader will note that for a comparison of the left and right frames to be valid, the axes on both frames must be the same. They are not, as the EoE is a dimensionless quantity. Furthermore, the reader will note that his analysis involves the extra parameter, T_0, which is the fracture toughness transition temperature as defined by T_{41J}.

This parameter is not considered in my analysis. Instead, I use the temperature at the point of inflection of the hyperbolic tangent ($RT_{NDT,PoI}$), as derived by PG&E and Westinghouse (Capsule V Report (2003)). Comparing the two graphs is not an "apples-to-apples" comparison; rather, it is better classified as an "apples-to-oranges" comparison. A fair comparison would be to simply replace T41J in the analysis with $RT_{NDT,PoI}$, and replot Figure IV.1.

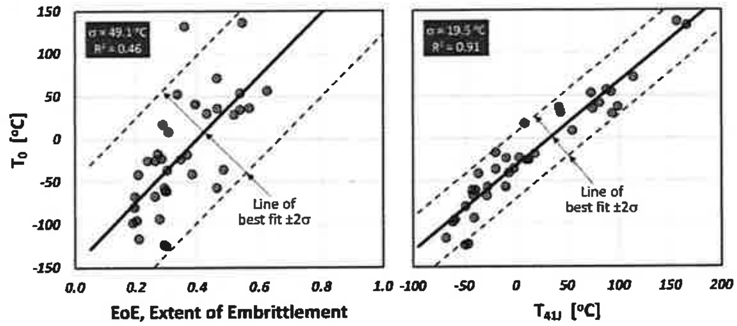


Figure IV.1. Graphs presented by Dr. Kirk support his assertion that my approach to calculating the embrittlement parameters is inferior to his (from (Kirk Report (2024)), Part 2, p. 57).

Dr. Kirk opines: "This comparison shows that EoE is not as well correlated with the fracture toughness transition temperature, T_0, as is T_{41J}." Also, "if *EoE* were used to predict T_0 the uncertainty in that prediction would be 2.-times greater than the uncertainty associated with a prediction of T_0". However, there is no direct relationship between EoE and T_0, so the broader scatter apparent in the left frame in Figure IV.1 is due to the relationship between EoE and ΔT_{41J}; that is, the scale on the EoE axis is magnified relative to that on the ΔT_{41J}. If that is the case, then the apparent difference shown in Figure IV.1 is probably an artifact and hence is invalid. Finally, the data sources are not identified, but a close examination of the data in both graphs suggests that the data differs. Accordingly, the plots in the two frames cannot be compared, and his analysis is fundamentally incorrect. Since my analytical technique has never been published, and considering Dr. Kirk's criticisms, I feel it necessary to describe the basis of the method below.

Thus, the values for the various parameters I sought by analytical calculation by manipulating Equation (1) are summarized as follows.

$$E = A + B\tanh\left(\frac{T-T_0}{C}\right) \quad (1)$$

where E is the fracture energy at temperature T and A. B. C, and T_0 are the optimized constants obtained by least squares optimization of Equation (1) on the experimental (E, T) Charpy impact data. From Equation (1), we find:

$$\text{USE} = A + B \quad (2)$$

$$\text{LSE} = A - B \quad (3)$$

$$FE_{PoI} = A \quad (4)$$

$$RT_{NDT,PoI} = T_0 \quad (5)$$

$$RT_{NDT,30} = T_0 + 0.5C\ln\left[\frac{B-A+3}{B+A-30}\right] = RT_{NDT,41J} \quad (6)$$

$$RT_{NDT,30} = \Delta T_{41J} = RT_{NDT,PoI} + 0.5Cln\left[\frac{B-A+3}{B+A-3}\right] = RT_{NDT,41} \quad (7)$$

and

$$\text{EoE} = \left(1 + \frac{e^x - e^{-x}}{e^x + e^{-x}}\right)/2 \text{ , where } x = \frac{RT_{NDTPOI} - T_0}{C} \quad (8)$$

Values for these parameters, as calculated from the coefficients given by PG&E (CAP V Report, Appendix D (2003)) are published in my Declaration of September 14, 2023. Again, it is essential to recognize that the T_0 in EoE is not the same parameter employed by Dr. Kirk (fracture toughness transition temperature, T_0, the arbitrarily defined T_{41J}) but is equal to $RT_{NDT,PoI}$. There is, however, an analytical relationship between $RT_{NDT,30}$ (Dr. Kirk's ΔT_{41J}) and $RT_{NDT,PoI}$ as given by Equation (7). I encourage NRC to settle on a single terminology for the various parameters to correct the considerable confusion in the field. The concepts are difficult enough for most engineers to grasp without being complicated by disparate names.

As noted above (Equations (6) and (7)), the arbitrary choice of 30 ft-lb as a metric for determining the value of RT_{NDT} (designated here as $RT_{NDT,30ft.lbs}$) is problematic in the author's viewpoint because, for the same material at different fluences, the fracture exhibits varying degrees of fracture character. Thus, the arbitrary definition of $RT_{NDT,30}$, as the metric by which dissimilar materials (e.g., irradiated vs. unirradiated) are to be compared ignores the potentially significant changes in the character of fracture character. Contrariwise, the use of $RT_{NDT,PoI}$ is consistent because, as shown by Equation (8), when the $RT_{NDT,POI} = T_0$, $EoE = 0$. This point corresponds to equal areas of brittle and ductile facets on the fracture surface, which defines the fracture character. Thus, when $RT_{NDT,POI} < T_0$, more brittle facets than ductile facets cover the fracture surface, but when $RT_{NDT,POI} > T_0$ more ductile facets than brittle facets cover the fracture surface. At the LSE and USE, the surfaces are purely brittle and ductile, respectively. Thus, by defining the reference temperature as $RT_{NDT,POI}$,the surface has equal coverage of ductile and brittle facets, so the fracture character is removed from the assessment when comparing one irradiation state with another or different materials.

Figures IV.2 to IV.4 present this analysis's findings. Furthermore, as noted by (Eason, E D; Wright, J E; and Odette, MCS G R (1998)) $RT_{NDT,30}$ (or $RT_{NDT,41J}$) may indicate a negative shift in the reference temperature with neutron irradiation (E > 1 MeV), seemingly indicating that neutron irradiation increases the fracture toughness, a nonsensical result. I have never found that to be the case with $RT_{NDT,PoI}$.

Examination of the data in Figures IV.2 to IV.4 shows that the embrittlement of the materials may be ranked as follows: Weld >> HAZ > Plate ≈ SRM. Figures IV.2 and IV.4 express these rankings more explicitly, especially in Figure IV.2. Finally, it is to be noted that the unirradiated materials are assigned fluence values of $1\text{x}10^{17}$ n/cm^2 because of the mathematical impossibility of assigning a value of zero fluence in a linear-log graphical format. The assigned value is too low to display any sign of embrittlement. Regardless of the arbitrary assignment of fluence for an unirradiated material, the results remain valid, as seen from the parameters for the three higher fluence values.

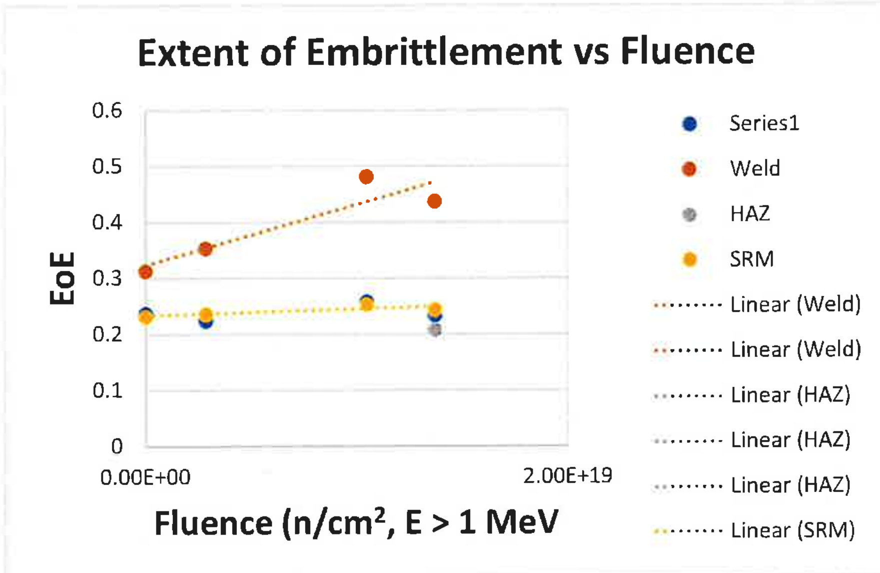


Figure IV.2. Plots *EoE* vs. Fluence for plate, weld, HAZ, and SRM specimens from Capsules S, Y, and V in Diablo Canyon, Unit 1 Nuclear Power Plant. Note that in some cases, "healing (reduction in the EoE)" occurs at high fluence (CAP V Report, Appendix D (2003)).

The above results are expressed in terms of the extent of embrittlement (*EoE*) as the materials accumulate radiation damage. The data plotted in Figures IV.2 and IV.4 provide this information. Figure IV.2 shows that while *EoE* is only weakly dependent on fluence for the plate, HAZ, and SRM materials, the *EoE* for the weld is strongly so. The variation in embrittlement is seen even more clearly in Figure IV.2, where *ΔEoE* is plotted against log(fluence). This illustrates that the weld material is the most susceptible to radiation embrittlement of the four materials exposed in the core in Diablo-Canyon, Unit 1, during the reactor surveillance program.

This conclusion is best supported by the data in Figure IV.3, which plots the change in EoE against fluence in a linear-log format. While the shift in *EoE* data for the plate, HAZ, and SRM are all less than 0.036, regardless of fluence, that for the weld exceeds 0.175 at the highest fluence.

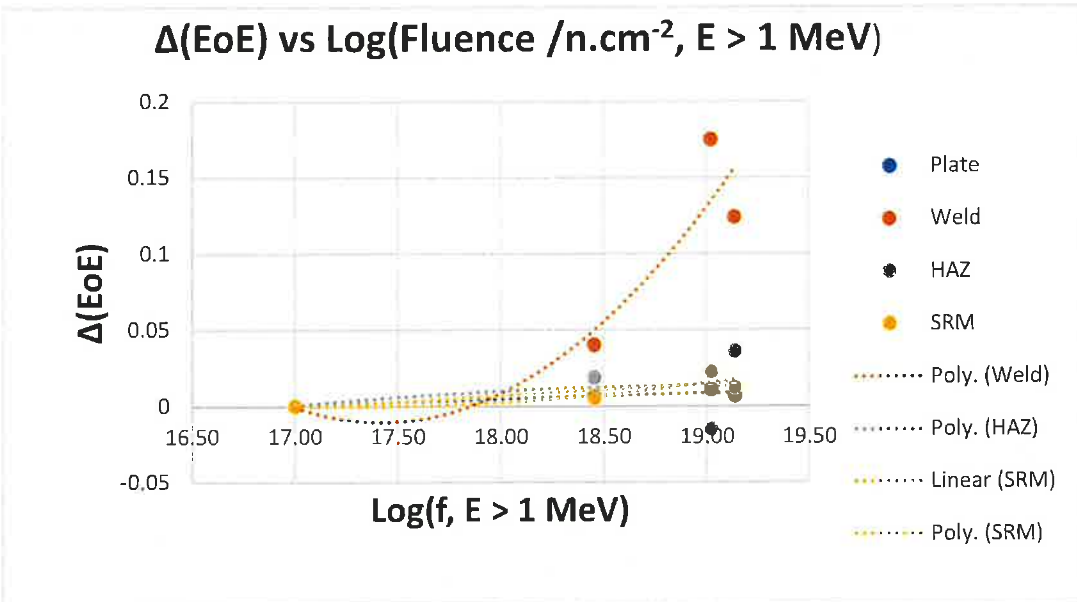


Figure IV.3. Plots ΔEoE vs. log(Fluence) for weld, plate, HAZ, and SRM specimens from Capsules S, Y, and V in Diablo Canyon, Unit 1 Nuclear Power Plant (Macdonald, D D (2023)). Note that the solid lines are included to emphasize that in some cases, "healing (reduction in the EoE)" apparently occurs at high fluence (CAP V Report, Appendix D (2003)), and Calculations by D. D. Macdonald, unpublished.

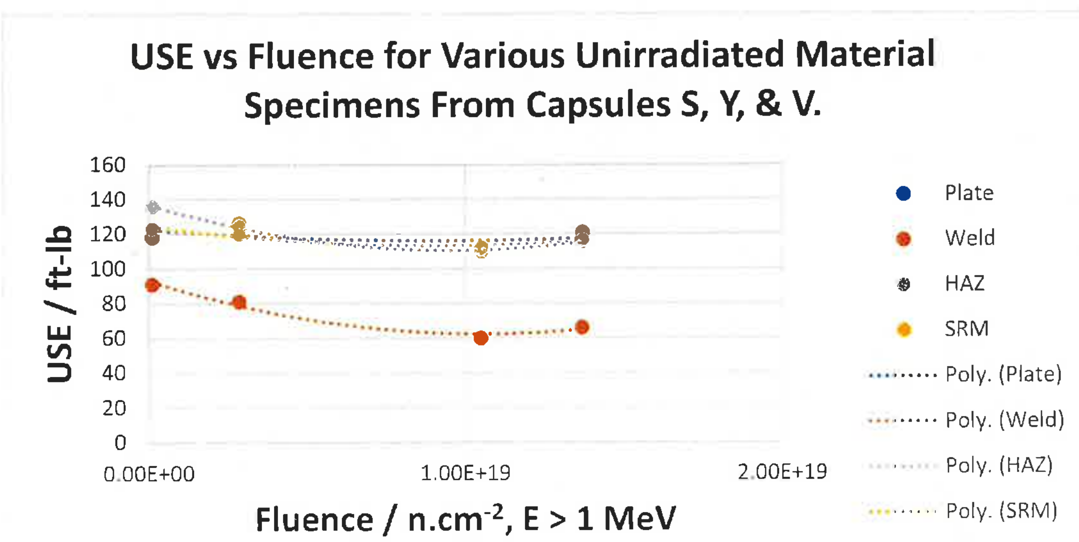


Figure IV.4: Plots USE vs Fluence for plate, weld, HAZ, and SRM specimens from Capsules S, Y, and V in Diablo Canyon, Unit 1 Nuclear Power Plant (CAP V Report, Appendix D (2003))

Figure IV.4 displays plots of the USE vs. fluence for the plate, weld, HAZ, and SRM specimens from Capsules S, Y, and V in Diablo Canyon, Unit 1, Nuclear Power Plant. As seen, the USE for the plate, HAZ, and SRM materials is only weakly affected by neutron irradiation at a fluence of up to $1.37\text{x}10^{19}$ n/cm^2, E > 1 MeV, but that for the weld material is more so with the USE being reduced from about 90 ft-lb for the unirradiated material to about 60 ft-lb at a fluence of $1.05\text{x}10^{19}$ n/cm^2, E > 1 MeV. Interestingly, all data indicating the embrittlement of the materials, including $RT_{NDT,30ft.lb.}$, $RT_{NDT\text{-}PoI}$, EoE, and ΔEoE vs. fluence, suggest that either the materials are not becoming highly embrittled or that the Charpy Impact Test (CIT) is not very sensitive to the extent of embrittlement. This issue has yet to be resolved. However, it points to the need to identify and develop other methods that might replace or complement CIT.

The data plotted in Figures IV.2 to IV.4 indicate that the degree of embrittlement passes through an extremum (USE through a minimum and EoE and ΔEoE through maxima) as a function of fluence. This issue, discussed briefly above, appears to be at odds with theoretical expectation, as it indicates that healing of the embrittlement damage occurs by continued neutron irradiation at very high fluences. It should be noted that the data for the two highest fluences might also be why PG&E declared the surveillance program data to be "not credible." At this point, I must ask: Were the data from Capsules V and Y somehow transposed during the analysis as postulated above? If so, transposing those data back yields a dataset that agrees with theoretical expectations. This possibility requires further analysis by examination of the original records.

Thus, I counter Dr. Kirk's characterization of my new method for analyzing Charpy data. The data employed in my analysis are the same data from DCPP-1 employed in his analysis, except that he introduces fracture toughness transition temperature, T_0. In essence:

1. I have manipulated Equation (1) to derive analytical expressions for the parameters of interest, noting that analytical expressions always yield more precise values for any physical parameter than visual examination.
2. By defining $RT_{NDT,PoI}$, I avoid the arbitrarily defined $RT_{NDT,41J}$, as introduced by ASTM, ASME, and NRC, and identified with ΔT_{41J} by Dr. Kirk. I argue that the former has an important physical basis, as the reference temperature has the same amounts of brittle and ductile fracture on the fracture surface.
3. To compare with Westinghouse and Dr. Kirk's empirically extracted data, I also calculate $RT_{NDT,41J}$ analytically, using the information in the whole curve as expressed by the coefficients in the optimized Equation (1).
4. Finally, I have no idea where the data used in his analysis (Figure 28) came from, as Dr. Kirk does not cite a source. Unless this is known, comparing the empirical ("eyeballing") method and the analytical process is premature.

Therefore, I argue that my method of extracting information from the Charpy curve is superior to that of Dr. Kirk, PG&E, and NRC because of its analytical nature. Finally, had Dr. Kirk wanted to compare his work fairly with mine, he would have replaced ΔT_{41J} in his analysis with $RT_{NDT,PoI}$, because that is where the two approaches differ.

Appendix V. Are Dr. Kirk's Estimates Conservative?

While the Kirk Report offers assurances of "conservatism" and strict adherence to NRC protocols, some questions arise regarding input values used in the Westinghouse Report, in which he expresses the utmost confidence. The (Kirk Report) reads: "All analysis performed herein that follow NRC requirements show PG&E has correctly assessed the Unit 1 RPV and it meets all current regulatory requirements to 60-80 years of operation (Kirk Report (2024)), Part 2, p.4). We strongly dispute Dr. Kirk's conclusion, as articulated elsewhere in this report. As already discussed, there are several reasons why PG&E appears not to follow NRC regulations in its 2011 PTS Evaluation ((PG&E re-evaluation Report (2011)) and (WCAP-17315-NP, Revision 0 (2011))), and there is insufficient surveillance data because of PG&E's postponements. For example, PG&E failed to extract Capsule B, which was the fourth capsule of the 4-capsule surveillance program at DCPP-1, in a timely manner, rendering it of only marginal use for assessing the state of embrittlement of the critical welds at the end of the initial 40-year licensing period as was intended. Similarly, they also missed the 2015 UT volumetric examination of the beltline welds, arguing instead for postponement. The UT inspection is now planned for 2025, but it has not been included in PG&E's license renewal application, and therefore, it is not clear whether it will be carried out at all in 2025.

UT inspection is essential because neutron irradiation causes damage through two mechanisms. At low fluence, the first is the cascade generation of vacancies or voids in the molecular lattice, many of which condense on foreign particles (e.g., Ni and Cu and various intermetallic particles containing these impurities). The resulting voids act as barriers to the movement of dislocations, thereby significantly decreasing the material's ductility (i.e., the material becomes more brittle). At higher fluence, the voids grow, commonly as "flying saucer" shaped fissures that, if adequately oriented, will have the central axis perpendicular to the principal stress axis (e.g., the stress caused by thermal shock during a LOCA) and will be characterized by a high Mode 1 loading stress intensity factor, K_I, at the periphery of the saucer. If it is above K_{ISCC}, the high K_I will induce slow (subcritical) crack growth at the saucer's periphery, with the radius growing and K_I increasing in value. At some point, K_I will exceed the fracture toughness of the steel K_{Ic}, *and the crack will propagate unstably and possibly result in a through-wall crack (TWC) and, hence,* rupture of the vessel. Thus, contrary to Dr. Kirk's assertions, high neutron bombardment is directly related to the formation of voids and embrittlement, increasing the probability and severity of through-wall crack propagation. Suppose the void is pressurized with a gas (e.g., He or mainly hydrogen). In that case, that pressure is added to the mechanical stress acting upon the crack to accelerate crack growth at stress intensity values significantly below K_{Ic}. The principal effect of the gases that may segregate into the fissure is that they reduce the time taken for the nucleation of a supercritical, unstable crack (i.e., one for which $K_I > K_{Ic}$). Hydrogen gas (25-35 cc/kg(H_2O)) is added to the primary coolant via the pressurizer. Additionally, the radiolysis of the coolant due to the intense neutron, γ-photon, and α-particles (He nuclei) radiation fields in the core results in the formation of the damaging hydrogen atom (H), which is separated from the RPV steel by only a 7-mm thick stainless-steel liner. Atomic hydrogen readily passes through austenitic stainless steel. Therefore, disregarding the possibility of HE/HIC by postponing UT inspections indefinitely is foolish. Dr. Kirk mistakenly asserts that this is not a potential safety issue requiring monitoring.

Theoretical models that accurately predict sub-critical ($K_I < K_{Ic}$) crack growth rate in embrittled Alloy 600 and 182 have been developed by the author and his colleagues and should be applied to predicting crack growth rate in embrittled ferritic PV steel accurately (Fekete, B, Ai, J, Yang, J, Han, J S, Maeng, W Y and Macdonald, D D (2018)) and (Shi, J; Fekete, B; Wang, J; and Macdonald, D D (2018)). Importantly, these models incorporate the recombination of atomic hydrogen in the cavities to form hydrogen gas, H_2. The models predict brittle fracture based on the electrochemical corrosion potential (ECP) at the primary coolant/Alloy 600 interface. This is discussed in some greater detail below.

I return now to the question of conservatism. For example, Dr. Kirk makes the statement that by deeming the PG&E pre-2004 surveillance data as "not credible" as per the credibility criteria defined in (RG 1.99, Revision 2 (1988)) the public should be comforted to know that the NRC procedures assure greater "conservatism" in projecting the reference temperature at the end of the 20-year license extension (RT_{PTS} values). It is, in fact, true that by deeming the pre-2004 surveillance data to be non-credible, operators are required to use more conservative chemistry factor calculations, as Dr. Kirk suggests. Table 6.1-2 (WCAP-17315 (2011)) - indicates that while pre-2004 data were "considered" and "integrated" (10 CFR 50.61 (2021)) into the new projections of RT_{PTS} for limiting weld 3-442C (weld wire Heat # 27204), it also seems that the more conservative CF value of 226.8 °F was not used. This more conservative value from Position 1.1 of and (RG 1.99, Revision 2 (1988)) is required when surveillance data is deemed "not credible." So it is clear that while Dr. Kirk assures "conservatisms" in the general description, these were minimized or removed. Even though PG&E and Westinghouse deem the capsule S, Y, and V data to be not credible, the less conservative 214.1 °F value, which should only be used if your data are credible, has become the input into the equation used to determine the RT_{PTS}. Even if the NRC eventually approves this PTS Evaluation despite failing to meet the sister plant criteria, the question should be, is this appropriate given the well-documented safety concerns being considered? Is it ethical to reduce chemistry factors when there are known deficiencies in the welds, with the apparent sole aim of minimizing safety program costs? Is this "conservative"? Only in terms of saving money would I argue that it is.

Suppose Dr. Kirk would argue that this is allowed because analysis of the new subset of 5 capsule tests finds the previously non-credible DCPP-1 data to be now credible. In that case, he cannot also claim that this new calculation incorporates "conservatisms" when, in fact, conservatisms have been eliminated through the latest "integrating" of a larger subset, which in theory would "reduce uncertainties" if it were not so evident that all of the materials in question have a history of embrittlement that cannot be denied. Kirk recommends and endorses a conscious decision to integrate sister plant data and lower chemistry factors without compliance with Appendix H and despite the history of weld metal embrittlement.

Thus, the Westinghouse PTS Report, at least regarding this variable, appears less conservative than the (Kirk Report (2024)) suggests. It seems that the "conservatisms" that Dr. Kirk assures us of, at least concerning this variable, may not have been delivered. Therefore, his assurances of "conservatism" being built into the calculations are false because this integration is performed without accounting for uncertainties arising from integrating sister plant data and meeting Appendix H requirements.

The precise composition of the welds had previously been tested (around 1992 at DCPP). The percentage by weight of nickel and copper impurities that directly cause the reactors' predisposition to premature embrittlement was reported to be 0.196% copper and 1.03% nickel -- Capsule Y, Report, Table 4-4, Pg. 4-5 in (Terek, E; Anderson, S L; and Madesky, A (1993)). Those values should not change over time. However, there will always be slight variations from sample to sample. The regulations incorporate tables to determine "chemistry factors" (CF) based on material property analysis (RG 1.99 Revision 2 (1988), pg1.99-4). Dr. Kirk has suggested that "Table 2" and chemical testing values should not change. If so, why do the CF values appear different within the same Westinghouse document? In effect, changing the table-based chemistry factors (CF) allows the data to fall into a tighter standard deviation, which in turn can also reduce the margin (M). Given that the calculation of the CF can be somewhat complex, minor changes in the variables can add decades to the projected life of a plant. As noted in (Wichman (1998)), tiny changes in CF result in comparatively significant changes in the resulting RT_{PTS} calculations and operational life of the plant. Should such minor changes in the (RG 1.99 Rev 2. (1988)) Table CF values be allowed or go unnoticed when they substantially impact the calculated results? How much do such errors reduce the actual margins of error built into the calculations? These questions should be addressed.

Table 6.1-2 in the (WCAP-17315 (2011)) and (Terek, E; Anderson, S L; and Madesky, A. (1995)) show that these variables (CF: Chemistry Factor, and Margin) play a role in determining the 60-year (54 EFPY) projection of embrittlement. It appears likely that they (Dr. Kirk, PG&E/Westinghouse) begin the interpolation with generally high embrittlement, as reflected in the RT_{41J} of the various capsules that have been tested, and in the end, the result is substantially lower than any of these inputs as a direct result of lowering the margins and removing "conservatism." This does not appear to reflect the conservatism that Dr. Kirk assures us of. One must question a procedure or an interpolation of higher embrittlement values that produces results that suggest more significant safety margins. In the case of Capsules S, Y, and V, the original data would appear close to the maximum allowed RT_{41J} at the end of 40 years (32 EFPY). Yet, the result of the interpolation is a significantly lower value that happens to be within the regulatory limit of $RT_{41J} < 270°F$ for 60 years of operation. (Terek, E; Anderson, S L; and Madesky, A (1993)) Terek et al. show the Chemical Composition of the Diablo Canyon Unit 1 Charpy Specimens from Capsule V. The chemistry factors (CFs) for Positions 1.1 and 2.1 (RG 1.99 Revision 2 (1988)) are summarized in Table 7 of (WCAP-17315-NP (2011)). These CFs should generally be followed, but as indicated above and articulated in greater depth below, I question how these variables and "conservatisms" were compromised.

To determine whether the calculations and embrittlement projections --in the Westinghouse PTS Report-- have been executed as per NRC procedure must be addressed:

1) Whereas the DCPP plant-specific S, Y, and V data are deemed "not credible" (three out of five sets that are integrated into the recalculation of CF values), would the guidance in (RG 1.99 Rev 2. (1988)) require that the more conservative CF value be used? The CF factor in Table 6.1-2 of (WCAP-17315 (2011)) is the value for Position 2.1 (RG 1.99, Revision 2 (1988)), the less conservative variable reserved for only credible data sets. The guidance in (GL92-01. US NRC (1992)) suggests that all six sets may be considered if five data sets appear credible. Only two of five sets, or in the best case, four of five sets, appeared credible in this case, yet the less conservative CF variables are used. This point seems to directly violate NRC guidance.
2) There would be a problem if the data were deemed "not credible." Still, the Westinghouse PTS Evaluation does not seem to incorporate the more conservative inputs dictated by its tables, so there would be a problem. Perhaps the justification is that the Palisades capsule data were deemed "credible" and integrated into this set. The new subset of five capsules is all considered credible for this exercise. If this is the case, it should be clearly stated in both Westinghouse and Dr. Kirk that this is the case. In any event, Dr. Kirk's claim that the

more conservative variables are incorporated appears false, and all suggestions that the Westinghouse calculations incorporate "conservatism" should be removed from his report. This statement appears in numerous places in the report.

3) It is questionable whether PG&E would be allowed to use a procedure incorporating ISP partner data without obtaining permission from the NRC. It is clearly stated (10 CFR 50, Appendix H (2024)) that this is required as a matter of procedure. I have not discovered any document showing that permission was granted or that the operational dissimilarities between Palisades and Diablo Canyon were adequately evaluated. Dr. Kirk suggests this is a trivial matter and that the operational differences are captured by temperature adjustments that have been made. However, some variables are not captured by these adjustments, and failing to analyze these operational differences fully introduced greater uncertainties when the new calculations lowered the margins of error in the form of the CF and M variables. This seems to open PG&E to potential miscalculations due to uncontrolled variables. I believe this violates the NRC procedure, which is being smoothed over.
4) If (10 CFR 50.61 (2024)) requires potential ISP partner plants to operate substantially similarly, having similar power levels and outage schedules (10 CFR Part 50, Appendix H (1995)), how similar must these be to ensure data integration? It is not entirely clear that 10 CFR 50 defines a threshold for similarity, but the bar is much higher than Dr. Kirk suggests. The fact that Palisades operated at partial power for years and that shielding was incorporated into altered fuel configurations would suggest that fluence would have been changed relative to operating temperatures. Dosimetry may not pick up on how shielding impacted exposure at the welds of most concern, depending upon the position of dosimeters. In short, variables are not accounted for in the temperature adjustments Dr. Kirk refers to. Dr. Kirk suggests these are trivial issues and that both plants can be assumed to run at 100% capacity, but DCPP-1 ran at partial power for about three years when it was undergoing equipment testing. Palisades was notorious for unplanned outages and power excursions, and it was run at partial power for years toward the end of its operation. The operational differences were, therefore, quite significant, and all these differences would suggest that the integrated program methodology that allows the CF and M values to be substantially lowered should be questioned. The fact that Dr Kirk dismissed this constitutes a violation of the NRC procedure, which the DCISC and the NRC should be more careful about investigating.
5) Dr. Kirk suggests that Capsule S is inherently less credible because of its low fluence levels, but is there regulatory guidance as to what low-fluence capsules are deemed not credible? I am not aware of guidance from the NRC that suggests all low-fluence capsules should be considered "not credible," and there is a strong correlation between the projected RT_{41J} (i.e., RT_{PTS}) of Capsule S and Capsule V (258°F and 250.9°F, respectively). Capsule S may be at the lowest fluence. Still, it is the capsule that exhibits the highest differential $\partial \Delta T_{41J}/\partial f$ implying that in Capsule S, ΔT_{NDT} is the most sensitive to f and, hence, carries the most information. Dr. Kirk is in error in summarily dismissing Capsule S.
6) The Guidance in (GL92-01. US NRC (1992)) suggests that data should not be deemed "not credible" without serious analysis of why a datum may appear an outlier. It is unclear that there is documentation dating back to the 2003-2011 period of such an analysis in the case of Capsule S, Y, and V data being deemed not credible at the time of publication of the (Capsule V Report (2003)). It is questionable why all three sets of data were considered not credible at the time when the data in Table D-2 in the (Capsule V Report (2003)--

Appendix D) indicated that there were at least two credible sets of data for each limiting material and in 2006, the recalculation of RT_{PTS} for the limiting weld (3-442C) performed by PG&E again seemed to assume that all three sets of data were credible. One potential reason why the Capsule Y data for the limiting weld appeared 2°F out of range was the unirradiated RT_{NDT} was -67°F when, in every other report, it seems to be -56°F. This would have shifted the RT_{PTS} by 12°F and more than accounted for why the data appeared non-credible. Documentation showing that the outliers were thoroughly analyzed when the (Capsule V Report (2003)) was issued would directly affect the Westinghouse Report conclusions. Such documentation should be added as a supplement to the (Kirk Report (2024)) and the Westinghouse PTS Report (WCAP-17315-NP, Revision 0 (2011).).

Appendix VI. Why UT Examination is Essential.

Introduction

Dr. Kirk dismisses the importance of PG&E missing a scheduled UT examination of the beltline region of DCPP-1, but let us look at the facts and consequences of such an omission. I will show below that his assessment is wrong, and his position must be rejected.

Radiation embrittlement involves two broad phenomena: the collision of high-energy neutrons with atoms in the metal to produce a vacancy and a "hot" (high-energy) interstitial metal atom. This high-energy interstitial collides with other metal atoms to produce further vacancy-interstitial (Frenkel) pairs. Thus, the population of Frenkel pairs multiplies in a cascade manner throughout the metal mass. Simultaneously, some metal interstitials and vacancies recombine to regenerate ordinary metal atoms in regular lattice positions and eliminate vacancies. You can think of these as billiard balls getting knocked out of one pocket and into another. However, the remaining vacancies will diffuse through the metal and become trapped at foreign particles, such as Ni and Cu in weld metal and at intermetallic particles formed by those entities with each other and with other metals in the system (e.g., Fe) to create not only barriers to the movement of dislocations but also to form nano-voids. That is why there is a direct relationship between a weld and plate metal impurities and the formation of voids or cracks in the metals. Dr. Kirk seems to dismiss this relationship.

Since plastic deformation occurs via the formation, movement, and annihilation of dislocations, any barrier to their movement will decrease the ductility of the metal, increase its hardness, and render it prone to brittle fracture at stresses that are considerably lower than if the metal had retained its ductility. Thus, there is also a direct relationship or at least a high correlation between impurities in the steels and reduced ductility. Kirk is incorrect in suggesting UT inspections do not indicate embrittlement because voids and nano-fissures in steels subject to fast neutron bombardment occur later. As the fluence increases over time, these processes proceed with the continued condensation of vacancies on the surface of the nano-void, thereby leading to void growth until the voids have evolved in size and shape, with the latter often taking on the shape of a "flying saucer" with the minor axis oriented parallel to the principal stress axis. In this configuration and shape, the stress intensity factor at the periphery of the void can be written as:

$$K_I = A\sigma\sqrt{l} \tag{1}$$

Where A is a constant for the system's geometry, σ is the tensile stress coincident with the minor axis of the saucer, and l is the distance from the load line to the saucer's periphery. If the saucer is filled with helium or hydrogen (He and/or H_2), the pressures of these gaseous components must be added to the mechanical tensile stress, σ. Accordingly, Equation (1) becomes:

$$K_I = A(\sigma + p_{He} + p_{H_2})\sqrt{l} \tag{2}$$

Equation (2) is essential because it provides the link between mechanics (σ) and chemistry ($p_{He} + p_{H_2}$) As shown below in the Hydrogen-Induced Cracking (HIC) phenomenon.

The events described thus far occur at low fluence and under low loading conditions such that $K_I < K_{ISCC}$, where K_{ISCC}, is the minimum stress intensity factor for slow (subcritical) crack growth (SCCG). As the micro-voids become filled with and pressurized by He and H_2, K_I increases. When $K_I > K_{ISCC}$, subcritical crack growth begins with the crack, and K_I increases with time during the subcritical crack growth period (Figure VI.1). This accelerates the crack growth rate, so the micro-crack size continues until $K_I > K_{Ic}$. At this point, unstable supercritical crack growth occurs, and the crack propagates at a significant fraction of the sound velocity in the material, as noted above. In the event of a PTS following a LOCA, the thermal stresses resulting from the rapid cool-down of the vessel and the high internal RPV pressure may become augmented by $p_{He} + p_{H_2}$ to lessen the time to transition from subcritical to supercritical crack growth, exacerbating the entire process.

At higher fluence, the processes of void growth and subcritical crack growth tend to continue. However, metal interstitial and vacancy generation via cascading still occur, provided that the steel is still being irradiated by high energy neutrons (E > 1 MeV); these processes are much closer to a steady state, as witnessed by any property sensitive to the damage approaching a horizontal axis asymptotically. As the voids and subcritical cracks grow, they, too, tend to combine to form microscopic fissures that are detectable by UT. Thus, the importance of UT is that it provides the last line of detection of fissures that are large enough that they might convert into unstable, supercritical cracks in the event of PTS following a LOCA.

Even though Ultrasonic Testing (UT) is required on a ten-year schedule, Dr. Kirk dismisses its usefulness. Thus, according to Dr. Kirk, "*SLOMFP have provided no evidence that the large number of UT indications detected in the Doel 3 and Tihange 2 (The Greens (2013)) reactors in Belgium in 2013 can plausibly exist in Diablo Canyon Unit 1. The root cause of these flakes was tied to unusual aspects of the initial manufacturing process that caused hydrogen flakes to exist in the Doel and Tihange forgings (Electrabel (2012)). A review of the information from Doel 3 and Tihange 2 by EPRI led to the conclusion that 'it is unlikely that conditions similar to those observed at Doel 3 exist in U.S. PWRs; and even if substantial* [flake] *indications are postulated to exist in beltline ring forgings in U.S. PWRs, the potential for vessel failure is acceptably low.' The NRC later concurred with this assessment. Also, in 2015, the Nuclear Regulatory Agency in Belgium (FANC) convened panels of national and international experts to review the concerns of Professors Macdonald and Bogaerts (Bogaerts, W F and Macdonald, D D (2022)); (FANC (2015)).*" These FANC-sponsored proceedings were not unlike the present one of assessing DCPP-1's fitness for service, with the operator ((Electrabel (2012)) and (Electrabel (2013))) working closely with the regulator (FANC) to synthesize a theory for the origin of hydrogen flakes. As a result of those proceedings, FANC concluded:

"The only theoretical propagation mechanism for the flaw indications in Doel 3 and Tihange 2 RPVs is low cycle fatigue, which is considered to have a limited propensity. Other phenomena (such as hydrogen blistering or hydrogen-induced cracking) have been evaluated and ruled out as possible mechanisms of in-service crack growth. The evaluation of significant evolution over time of hydrogen flakes due to the operation of the reactor units is unlikely. The comparison between the inspection data from the 2012 and 2014 UT inspections, applying the same parameters and reporting thresholds, does not (provide) evidence of crack growth. However, the time between the restart in 2013 and the shutdown in 2014 is too short to claim that there is definitive experimental evidence of no in-service fatigue crack growth. Therefore, the FANC requires the Licensee to perform follow-up UT-inspections, using the qualified procedure on the

entire reactor pressure vessels wall thickness at the end of the next cycle of Doel 3 and Tihange 2, and after that at least every three years."

As Dr. Kirk notes, Professor Bogaerts and I were independent parties to the analysis of the Doel-3 and Tihange-2 PWRs in Belgium, and our findings were somewhat different from the official version of FANC. Both FANC and Dr. Kirk miss our point: hydrogen from *any source*, including moist air when hot forming the steel ingots into shells for the RPV or from corrosion during operation of the reactor or from the radiolysis of the hydrogenated coolant, is very damaging to the steel. This has been well-documented in the oil and gas industry, where massive explosions of natural gas transmission pipelines appear to be an almost regular occurrence. Previous reports have been made of the possibility of the hydrogen embrittlement of PWR RPVs with the hydrogen derived from corrosion. This issue is explored later in this report.

The point of this discussion is that UT examinations of the beltline materials in PWRs are critical, especially toward the end of their license period when the formation of larger voids and cracks is increased. The UT technology uses carefully placed ultrasound emitters and sensors that pick-up reflections of the ultrasound waves from discontinuities in the mechanical impedance, such as that offered by a void or fissure. *UT examination is the only technique to detect and characterize the defects that might eventually transform into unstable, brittle fractures resulting from PTS following a LOCA. UT examinations are scheduled on a ten-year basis; the last was performed on DCPP-1 in 2005. PG&E missed the UT inspection planned for 2015, with the permission of the NRC, and has now rescheduled the next for 2025. In my expert opinion, missing such an important examination is inexcusable, given the consequences of the through-wall cracking of the RPV if a near-critical crack exists in the wall in the event of PTS following a LOCA. I have seen in the DCISC record that PG&E often makes the public statement that its safety track record is among the best in the nation. When they have delayed such fundamentally important safety inspections, it is not honest to make such representations. The DCISC should REQUIRE more regular UT inspections for both DCPP-1 and DCPP-2 moving forward.*

Reactors Doel-3 and Tihange-3 in Belgium

The inner surface of most western PWR reactor pressure vessels (RPV) is clad with austenitic stainless steel to reduce corrosion rates and CRUD formation. Of instances, this clad has been reported to develop cracks, which will create direct contact of the base RPV metal (ferritic low-alloy steel) with the high-temperature coolant (boric acid + lithium hydroxide + hydrogen) in the primary circuit and possible progression of the cracks into the base material. In June 2012, a pivotal moment in nuclear reactor safety was marked by the implementation of ultrasonic, in-service inspections (UST) at the Belgian Doel 3 nuclear power plant (Electrabel. (2013)). This new technique/instrumentation was used to meticulously check for under-clad cracking in the reactor pressure vessel, a concern previously identified at Tricastin 1 in France.

Figure VI.1 shows a schematic illustration of the Doel 3 RPV. The vessel's total height is approximately 13 meters (incl. the spherical top lid), with a diameter of 4.4 meters and a wall thickness of the cylindrical part of 205 mm. The primary water side of the RPV is clad with a stainless-steel Type 308/309 lining of approx. 7 mm thickness. The RPV base material is a SA 508 Cl. 3 low-alloy Mn-Mo-Ni steel (i.e. 1.2-1.5% Mn, 0.45-0.60% Mo, 0.40-1.00% Ni, max. 0.25% Cr, max. 0.25% C).

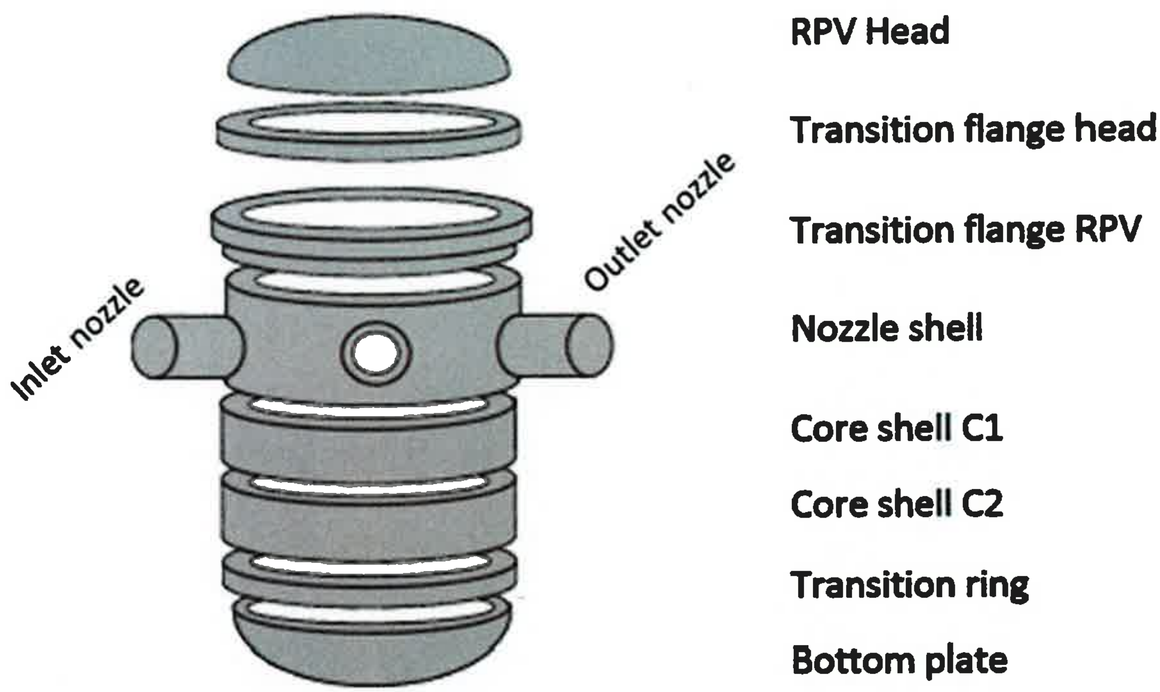


Figure VI.1: Illustration translated from ((FANC (2013)) and (FANC (2015))) showing the original forged steel ring sections of the RPV separated for clarity. These rings are welded and clad internally with a stainless-steel liner to form the reactor pressure vessel.

No under-clad cracking defects were detected. However, to our surprise, atypical UST "indications" in the RPV shells were found in the first 30 mm of the material depth of the irradiated part of the Doel 3 RPV core shells (Electrabel (2013)). This unexpected discovery prompted the operator to order a full-thickness RPV shell inspection in July 2012, which confirmed high numbers (thousands) of similar "indications" ((Peachey, C (2012)) and (TIHA-169 (2013))) down to a depth of 120 mm into the material, measuring from the reactor's primary water side.

The flaws appeared "almost circular in shape" with a reported average diameter of 10-14 mm (approximately ½"), "although some had diameters more than 20-25 mm" (approximately 1") Some available data, however, showed significantly higher values. (cf. see Figures VI.3 and VI.4) (Electrabel (2012)). It was also observed that the detected defects are oriented in the rolling direction of the plate in both vessels (Doel 3 and Tihange 2 reactor pressure vessels, 2013; Defects in the reactor pressure vessels of Doel 3 and Tihange 2. (The Greens (2013).) and that their position and orientation showed a pattern similar to a zone of assumed macro-segregations. (Van Walle, E (2013)). Bridging was found to occur only between flakes very close to each other.

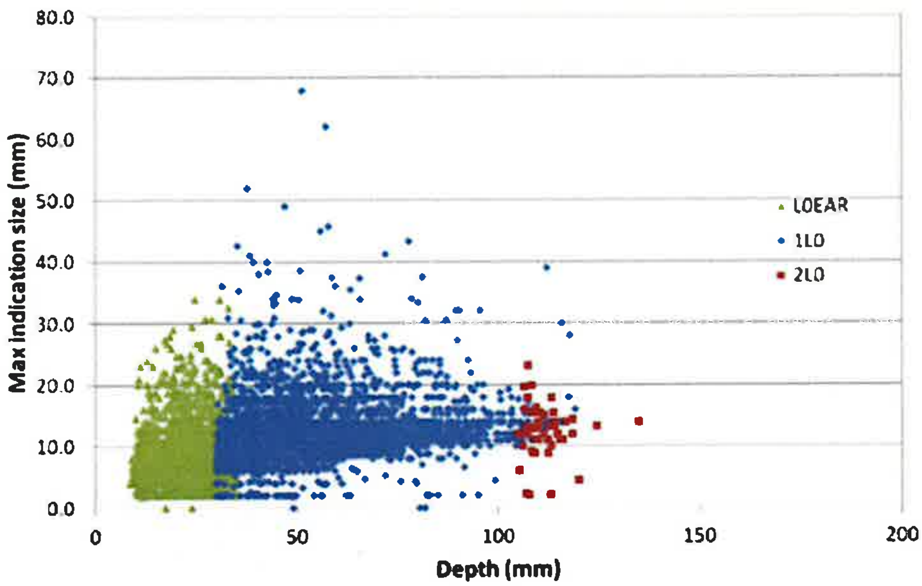


Figure VI.2: Doel 3 – Size of indications [max (x, y)] versus depth into the RPV steel wall (data 2012). (Electrabel (2013); The Greens (2013))

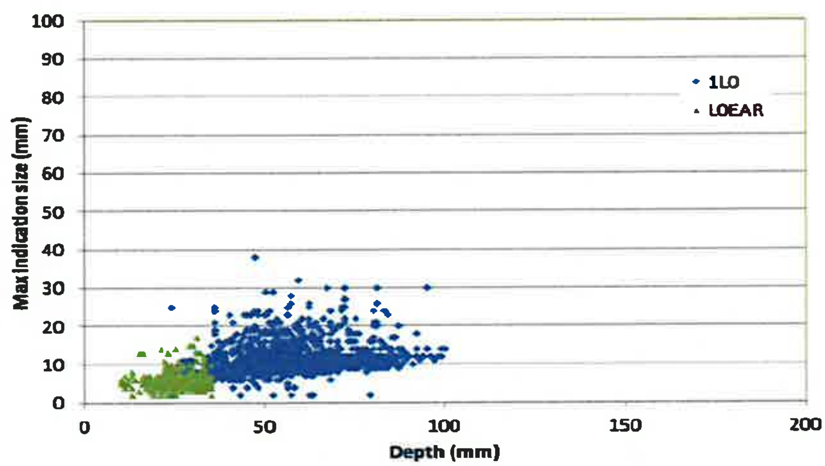


Figure VI.3: Tihange 2 – Size of indications [max (x, y)] versus depth into the RPV steel wall (Electrabel (2012).).

In 2014, there were a total of 13,047 "hydrogen flaws" in Doel 3 and 3,149 in Tihange 2 (Electrabel (2012)) (Compare those numbers to Figures VI.2 and VI.3). However, three years after the first detection of these "thousands of hydrogen flaws" in the RPV shells and after new investigations, both reactors received authorization to restart in December 2015. Since then, the affected reactors have been plagued by several scrams and unforeseen shutdowns. Given all this, the potential problem of (hydrogen-related) crack growth in the RPVs and the related longer-term aging problems of the reactors are still imminent and deserve further attention—probably more than ever. Although Dr. Kirk dismisses the relevance of this case study, it is pertinent to DCPP-1 precisely because of the known metallurgical inadequacies inherent in the design and fabrication of the RPV. This example illustrates why delaying UT inspections for 20 years is risky when standard NRC protocols dictate otherwise. The NRC has been negligent in allowing such inspections to be waived or delayed when the history of the plant and the only available surveillance data indicate premature embrittlement of the RPV.

In the case of the Belgian plants, for example, it remains unclear if the cracks found in the NPPs Doel 3 and Tihange 2 are "only" manufacturing artifacts or if there is also an "operational component" contributing to the current problems and operational risks, i.e., whether the cracks are still progressing (as they seem to be from a comparison of Figures VI.2 and VI.3) and whether there are other phenomena, e.g., similar to 'hydrogen blistering' or hydrogen-induced cracking (HIC) processes, contributing to the problem. Additional hydrogen might come from the cathodic corrosion reactions occurring on the primary water side of the reactor pressure vessel or from radiolytic hydrogen of primary water decomposition. During operation, there is a permanent flux of (corrosion-originating or radiolytic) atomic hydrogen – although the flux might be small – and this hydrogen could quickly become trapped into the present voids or are created at inclusions in the wall of the RPV. An eventual pressure build-up in the flakes will result in growing cracks and other material degradation phenomena.

It is evident that not just Doel 3 and Tihange 2 in Belgium could be affected. The RPVs were fabricated by the now-bankrupt RDM (Rotterdamsche, Droogdok, Maatschappij, in the Netherlands), which also manufactured RPVs for at least 20 other reactors that are operating in seven countries around the world, including some 10 in the United States. Also, more recently, phenomena like those in the Belgian reactors have been detected in the Swiss reactor Beznau-1 and, to a lesser extent, in Beznau-2. These RPVs were fabricated by a different manufacturer, demonstrating that other factors, like steel supplier, cladding process, and final assembling, may have played an essential role in developing the observed damage. It should be recognized that a pressure vessel with a density of flaw "indications," as found in 2012 in both Belgian RPVs, would not have been accepted at the fabrication time. If some of the hypotheses discussed above were to be proven true, it might significantly impact currently operating PWRs worldwide. The relative uncertainties about the rate of void and crack growth over time and the causes specific to the operating conditions of a particular plant are precisely why UT inspections should be performed at regular intervals at any facility. Only by mapping subcritical flaws and cracks over time can an operator have confidence about the operating condition of the plant.

Causes of the Cracks

Since the flaws were first discovered, different investigations have been carried out (Flaw indications in the reactor pressure vessels of Doel 3 and Tihange 2 (FANC (2015)) (Fruehan, R J (1997)). They have highlighted so-called *'hydrogen flakes'* as the root cause of the problem. These hydrogen flakes might arise during the fabrication of large steel ingots. Solidification of a large mass of steel is characterized by the significant development of micro- and macro-defects in the ingot structure and the changing solubility of different elements during cooling. For example, the solubility of hydrogen (e.g., originating from thermal dissociation of water molecules from damp scrap, fluxes, atmospheric humidity, etc.) decreases during solidification and cooling down of the steel ingot. The solubility of hydrogen in steel at room temperature is approx. 0.1 ppm, compared with 30 ppm in molten steel.

Hydrogen atoms possess high mobility in the steel matrix but are collected at internal voids, such as non-metallic inclusions (sulfides, oxides), shrinkage pores, cracks caused by internal stresses, etc. Hydrogen atoms collected at such internal micro-voids combine and form gaseous hydrogen molecules H_2, which may cause the formation of cracks ("flakes" in the traditional steel jargon) when the gas pressure exceeds the steel strength. The "hydrogen flakes" are internal cracks extending radially in all directions from a center (e.g., inclusion), with the typical characteristics of a hydrogen-induced (trans crystalline) brittle fracture.

Hydrogen flakes (sometimes called '*shatter cracks*': internal fissures seen in large forgings due to segregated hydrogen) are well-known from the past, and their possible formation is hazardous for parts fabricated from large ingots. A potential remedy is to use vacuum ladle degassing methods to decrease the hydrogen content to < 2 ppm, which should avoid or mitigate flake formation. Still, this was not done to prepare the ingots from which the RPVs for the Doel and Tihange PWRs were fabricated.

Not all forged components of the Doel 3 and Tihange 2 RPVs contain the same number of flaws. Based on an analysis of the ingot size and the combined sulfur and hydrogen content, there appears to be a good correlation between the intrinsic susceptibility to hydrogen flaking and the number of flakes found in each forged component.

The critical question remains about the possible evolution of these so-called "hydrogen flakes" over time. The position of the regulatory authorities and the operator, so far, has been that the defects found in the Doel and Tihange RPV "are usually associated with manufacturing and are not due to aging" and that it is "improbable" that the flaws have evolved since their formation. The only theoretical propagation mechanism still considered is 'low cycle fatigue.' Also, the limited experience concerning the influence of irradiation on flaw propagation in zones with hydrogen flakes is, however, recognized.

One of the main reasons for concluding that it is unlikely there has been a significant evolution of the voids over time is the claim that "there is currently no source of hydrogen anymore," which could cause the propagation of the cracks. Recognizing that the inner surface of the RPV is in contact with an aqueous solution (primary coolant), this is an incredible and erroneous conclusion.

Boonen et al. (Boonen, R and Piers, J (2016)) also discuss hydrogen segregation during steel production and manufacturing as the only cause of the current cracks. The so-called 'Safety Case Report' for Doel 3 states that the concentration of hydrogen decreases from 1.5 to 0.8 ppm during the cooling of the steel. A first estimate for the high-density flaw region of the lower shell shows that this would equal about 61 mL H_2 in total. To obtain an approximation of the amount of hydrogen needed to generate all the flaws in this section of the RPV, Boonen et al. (Boonen, R and Piers, J (2016)) have carried out several linear elastic finite element simulations to estimate the total flow volume. Based on this theoretical calculation, they came up with a needed H_2 volume of 604 mL, approximately ten times the stated released amount of hydrogen from the steel. These calculations appear realistic since they would also result in a calculated average flaw/crack opening width of 0.25 to a few micrometers.

This study concludes that traditional "hydrogen flaking" cannot be the only cause of the present flaws. Either another cause has resulted in the flaking, or the flaws have been growing. Boonen et al. (Boonen, R and Piers, J (2016)) have also severely questioned the validity of the fracture mechanics approach used by the operator. The interaction of the many multiple cracks, as seen in the current case, is not covered by the ASME approach.

Water chemistry, corrosion effects, and hydrogen sources.

<u>Hydrogen from cathodic partial corrosion reaction</u>

The flaking phenomenon described above is reminiscent of the well-known 'hydrogen blistering,' 'water blistering,' or hydrogen-induced fracture phenomena from corrosion in the chemical and petrochemical industries. Hydrogen blistering can occur when hydrogen enters steels due to the reduction reaction (hydrogen evolution via water and/or proton reduction) on a corroding metal surface. In this process, single atoms of "nascent" hydrogen (H atoms) diffuse through the metal until they react with another atom, usually at inclusions or defects. The resulting diatomic hydrogen molecules are too large to migrate through the metal lattice and become trapped. Eventually, a gas blister or internal crack builds up and may split the metal, as schematically illustrated in Figure VI.4. Practical examples are shown in Figure VI.5.

In the presence of already existing cracks (cf. the "hydrogen flakes"), the newly generated hydrogen may be responsible for further crack growth in two ways: either through the pressure build-up by molecular hydrogen or through the concentration of hydrogen atoms and embrittlement phenomena at the crack tips and on grain boundaries. Both processes will have the same deleterious effect on the metal's structural properties.

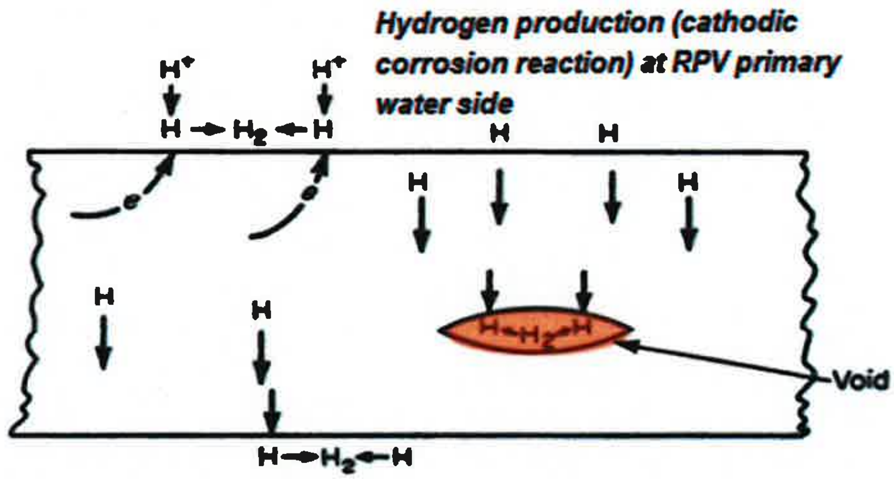


Figure VI.4. Schematic diagram of hydrogen diffusion and blister formation.

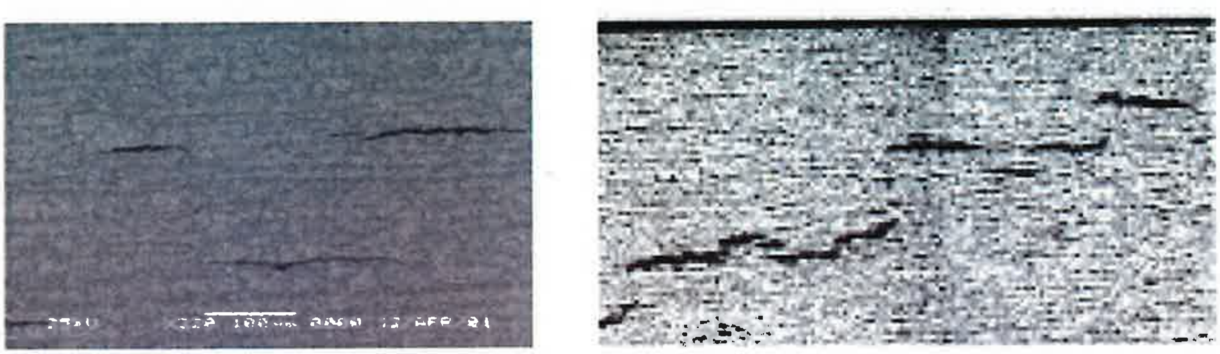


Figure VI.5. Typical hydrogen-induced cracks (An Atlas of Corrosion and Related Failures (1987)).

Hydrogen blistering or cracking is controlled by minimizing corrosion. It usually is not a problem in neutral or alkaline environments, but it is with high-quality steels with low impurity and inclusion levels. Nevertheless, also under the primary water chemistry conditions of the reactor coolant system (RCS) of PWRs, with a typical pH_T of approx. 6.9 -7.4 at the operating temperature of ca. 300 °C (corresponding to a room temperature pH of around 10), the primary cathodic corrosion reaction will be $H_2O + e^- \rightarrow H + OH^-$, because of the low proton concentration compared with water. Even the meager corrosion rates of the stainless-steel cladding (e.g., 0.1 to 1 micron/yr) will result in significant quantities of corrosion-generated hydrogen atoms that will evolve and may enter the base metal of the RPV (> 10^{24} - 10^{25} atoms/yr). In this respect, (Tomlinson, L (1981)) has shown that in oxygen-free, high-temperature water, more than 90% of the hydrogen generated in the cathodic corrosion reaction is absorbed by the steel.

Austenitic stainless-steel cladding is sometimes considered to prevent hydrogen diffusion and potential hydrogen-induced cracking problems in pressure vessels. To our knowledge, this has never been proven experimentally adequately.

(Mazel, R E; Grinenko, V G; and Kuznetsova, T P (1980)) wrote that, at most, the cladding probably has only a "delaying" effect in transferring the nascent hydrogen to the cladding/base metal boundary and further into the RPV steel matrix. The presence of flaws in this matrix (cf. "hydrogen flakes") represents ideal, irreversible sinks (traps) for the hydrogen injected into the metal from the cathodic corrosion reaction.

In addition to the corrosion-generated hydrogen, hydrogen radicals are formed as a result of the radiolysis of water and the reactions of H_2 with the radiolysis products (e.g., $OH\cdot + H_2 \rightarrow H\cdot + H_2O$) (Macdonald D D and Urquidi-Macdonald, M (2007)). Hydrogen is used in the RCS to suppress radiolytic oxygen and hydrogen peroxide formation. These effects are discussed elsewhere (Bogaert, W F; Zheng J H; Jovanovic A S and Macdonald, D D (2015)).

Finally, earlier observations of hydrogen-induced blister cracking have been reported in nuclear structural materials (Namboodhiri, T K G (1984)), and there has been considerable debate about the issue in the past. An old, specific example of failures attributed to hydrogen occurred in retaining rings used to connect inlet assemblies to the reactor process tubes in a Hanford, water-cooled production reactor. Failures occurred in carbon steel and Type 420 stainless steel. The reported hydrogen sources were the fabrication process, hydrogen generated during the corrosion of the ring by the process water, and the galvanic coupling (Westerman, R E (1961)).

Aging risks and crack growth

Given all the above, the "trapping" of cathodically generated hydrogen (due to primary water corrosion reactions) or radiolytic hydrogen inside existing "hydrogen flakes" is not improbable at DCPP Unit 1. Moreover, the (original) flakes may act as stress raisers, enhancing the hydrogen diffusion to the stressed areas in the metal, possibly also resulting in hydrogen stress cracking (HSC).

Also, the additional effect of irradiation is still largely unknown. Traditionally, it has been assumed that hydrogen significantly affects the fracture properties of pressure vessel steel in both the unirradiated and irradiated states at hydrogen contents above two ppm. Some studies have measured the hydrogen content of the cladding and found it to be 3-4 ppm after prolonged irradiation in PWR water (Koutsky, J and Splichal, K (1986)). This is also assumed to be the equilibrium content at the cladding/base material interface. Although hydrogen diffuses quickly in the RPV steel at high temperatures, the presence of efficient hydrogen traps, such as the "hydrogen flakes," poses a severe threat. The effect of irradiation and the importance of hydrogen in some observed low fracture toughness values requires further research.

Hydrogen Embrittlement in PWRs

The possibility of hydrogen segregation into the voids and fissures in the beltline region raises the ugly head of hydrogen embrittlement or hydrogen-induced cracking (HIC), which must be addressed. This is because hydrogen-induced cracking, a form of hydrogen embrittlement, is already recognized as the mechanism of the primary-side failure of cold-worked Alloy 600 steam generator tubes at bends, expansion joints into the steam generator tube sheets (Figure VI.7) and in penetrations through the carbon steel support plates in the event of denting corrosion (Macdonald, D D and Engelhardt, G (2022)). Ni-base alloys, including Alloy 600, Aloy 182, Alloy 690, and Inconel 800, are used extensively in PWRs. The reader will note that all primary water stress corrosion cracking in PWR steam generator tubing occurs in regions where the tubing is intentionally deformed (bending or expanding into the lower tube support plate) or unintentionally stressed, such as in denting corrosion. Sometimes, PWSCC is called IGA/IGSCC (Inter Granular/Intergranular Stress Corrosion Cracking, OD (outer diameter), but these forms of attack are all the same phenomenon: hydrogen-induced cracking.

This is a concern for the DCPP-Unit 1 RPV because -- even though the ferritic RPV steel is not in direct contact with the coolant (being separated from it by a 7 mm thick stainless-steel liner) -- hydrogen can transverse the liner with ease. Accordingly, the possibility of a "hydrogen" component to the embrittlement of the RPV steel should always be expected possibility of a "hydrogen" component to the embrittlement of the RPV steel should always be expected. Hydrogen is also responsible for the brittle fracture of other materials in PWRs, such as "stress corrosion cracking" of nickel-base alloys, such as Alloy 600 steam generator tubing and Alloy 800 components, Alloy 182 welds, and cold-worked Type 316 stainless steel barrel bolts, all of which occur on the coolant side but which, nevertheless, are fundamentally caused by hydrogen embrittlement (HE). These examples illustrate the ubiquitousness of HE in the failure of embrittled materials in PWRs, and there is no reason to expect that a PWR RPV is immune. Indeed, (Toribio J et al. (2017)) have recently characterized the harmful role of HE in the embrittlement of the pressure vessel structural materials in a WWER-440 nuclear power plant (a Russian PWR), and this is likely only the tip of the iceberg.

Figures VI.6 (a) and (b) show cutaway views of a typical recirculating steam generator employed in Westinghouse PWRs. Figure VI.6 (a) shows the flow paths of the primary coolant (black) and the secondary side coolant from the entry point as liquid water to exit at the top of the steam generator as dry steam to the high-pressure steam turbine.

Figure VI.6 (b) shows the regions of different forms of corrosion. The forms identified as Primary Water Stress Corrosion Cracking (PWSCC) and IGA/IGSCC (Intergranular Attack/Intergranular Stress Corrosion cracking) on the tube ID (i.e., in contact with the primary coolant) are all the same phenomenon: hydrogen-induced brittle fracture, as noted above.

The conditions in the primary coolant of a typical Westinghouse PWR are summarized in Table VI.1 of this Report. The composition of the coolant and the amount of added hydrogen are essential to the present discussion. The latter is important because hydrogen is a reducing agent and shifts the electrochemical corrosion potential (ECP) in the negative direction, thereby exacerbating hydrogen-induced (brittle) cracking (HIC). Under irradiation conditions (Figure

VI.6), this phenomenon becomes even more prevalent because of the generation of atomic hydrogen, which is the species that enters the metal and diffuses to voids, recombines to form hydrogen gas ($2H \rightarrow H_2$), which pressurizes the voids and fissures, to enhance significantly subcritical ($K_I < K_{Ic}$) crack growth rate.

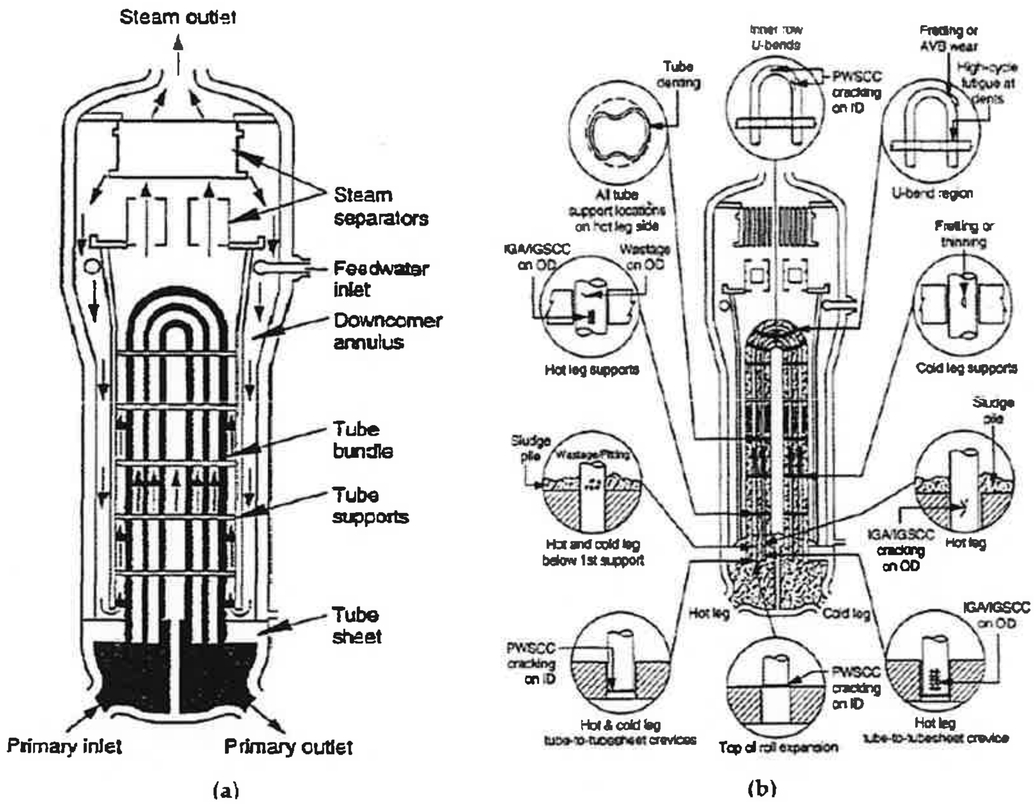


Figure VI.6. Corrosion problems in PWR steam generators. (a) General layout of a recirculating steam generator. (b) All PWSCC (Primary Water Stress Corrosion Cracking, including OD IGA/IGSCC) form a brittle fracture.

Table VI.1. Typical conditions exist in the main loop of the primary coolant circuit of a PWR. (Macdonald D D and Urquidi-Macdonald, M (2007)).

Property	Value	Comment
Temperature	295°C–330°C	Typical
Pressure	150 bar (2250 psi)	Typical
Coolant composition	4000–0 ppm B as boric acid, 4–1 ppm Li as lithium hydroxide, depending upon the burn-up of the fuel and the vendor	Li-B trajectory over a typical fuel cycle is shown in Figure 27(a).
Hydrogen concentration	25–55 cc(STP [standard temperature and pressure])/kg(H_2O)	Some noncommercial units operate with [H_2] as high as 70 cc(STP)/kg(H_2O)
Core channel dose rate		Typical
γ-Photon	3×10^5 Rad/s	
Neutron	6×10^5 Rad/s	
α Particles	3×10^5 Rad/s	
Coolant Mass Flow Rate	18,000 kg/s	Typical

Compared with the work on modeling BWR primary coolant circuits, much less work has been reported on assessing electrochemical effects in PWR primary circuits. This situation reflects that cracking has not been as significant a problem in PWR primary coolant circuits as in BWR primary coolant circuits. However, the primary water stress corrosion cracking (PWSCC) of mill-annealed Alloy 600 steam generator tubes, pressurizer components, control rod drive tubes, and baffle bolts (highly cold-worked Type 316 SS) have been severe, recurring issues in PWR operation, for example. Because of the high hydrogen concentration [typically 25-50 cc(STP)/kg(H_2O), corresponding to 1.12×10^{-3} m to 2.24×10^{-3} m or 2.23 to 4.46 ppm] employed in a PWR primary circuit to "suppress radiolysis," and given the lack of sustained boiling, it was generally believed that the hydrogen equilibrium potential dominates the ECP and hence that the coolant circuit acts as a "giant hydrogen electrode." If so, an approximate value of the ECP is readily calculated from the known pH, which, in turn, is easily estimated from the boron and lithium contents of the primary coolant and the known hydrogen concentration using the Nernst equation. Considering subsequent modeling, this picture is not entirely accurate; more importantly, PWRs are not free from cracking in their primary circuits, and the cracking observed is brittle and very potential-dependent. For example, Primary Water Stress Corrosion Cracking (PWSCC) of Alloy 600 steam generator tubes has plagued operators for many years, as noted above, and cracking of core barrel bolts (highly cold-worked Type 316 SS) has also been a recurring problem. A typical brittle fracture in highly cold-worked Alloy 600 in PWR primary coolant is shown in Figure VI.7, proving that brittle fracture occurs within many materials in PWR primary circuits. This issue is discussed further below.

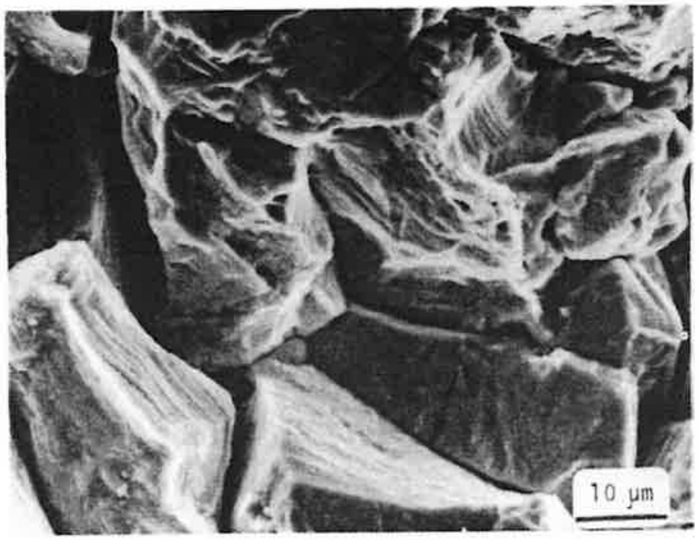


Figure VI.7. IGSCC in Alloy 600 after exposure to simulated PWR coolant at 350 °C and 0.1 MPa of H_2. Note that the solution was saturated with H_2 at ambient temperature and then pumped into the autoclave at 350 °C at sufficient pressure to suppress boiling. The specimen failed in a CERT.

While there are significant material differences between BWR and PWR primary circuits, in both cases, it has gradually become evident that the electrochemistry of the coolant is a prime factor in the nucleation and propagation of corrosion damage. Many lessons should have been learned from experiences within the fossil-fueled power community that appear not to have been heeded in the early days of reactor operation. Thus, at the dawn of the fission reactor age, when it was regarded that a fission reactor was like a conventional thermal (fossil-fueled) plant "with a different heat source," materials were employed that were prone to failure mainly because of the presence of oxidizing radiolysis products, such as O_2, H_2O_2, OH, etc., in the coolant. This led to many corrosion problems, such as IGSCC in sensitized stainless steels, particularly in BWRs, that have plagued fission plants over the past fifty years.

To understand the chemistry of a PWR primary circuit coolant, we have employed a proprietary radiolysis model to calculate the evolution of various chemical species with time upon "switching on" the fission processes typical of a PWR operating at full power. The concentrations of all species are then plotted as a function of time in Figure VI.8. Of particular interest are the concentrations of H_2, H, and OH, all three potent reducing agents, one of which may enter a steel lattice through the surface without the need to be formed by the dissociation of hydrogen, $H_2 \rightleftarrows 2H$. From this equilibrium, we see that the hydrogen atom concentration is predicted to be about $2\text{x}10^{-8}$ M. From thermodynamic guesstimates, the concentration of hydrogen atoms in equilibrium with molecular hydrogen at a pressure of 1 atm at 300 °C is calculated to be about $1\text{x}10^{-20}$ m. The equilibrium constant for this reaction is written as:

$$K_p = \frac{M_H^2}{p_{H_2}} \tag{3}$$

where K_p is estimated to be about 10^{-40} M^2/atm. We may turn this equation around and calculate the pressure of hydrogen needed to obtain an H concentration of $2\text{x}10^{-8}$ M. The result is an astounding $4\text{x}10^{24}$ atm. No such pressure has ever been generated on Earth.

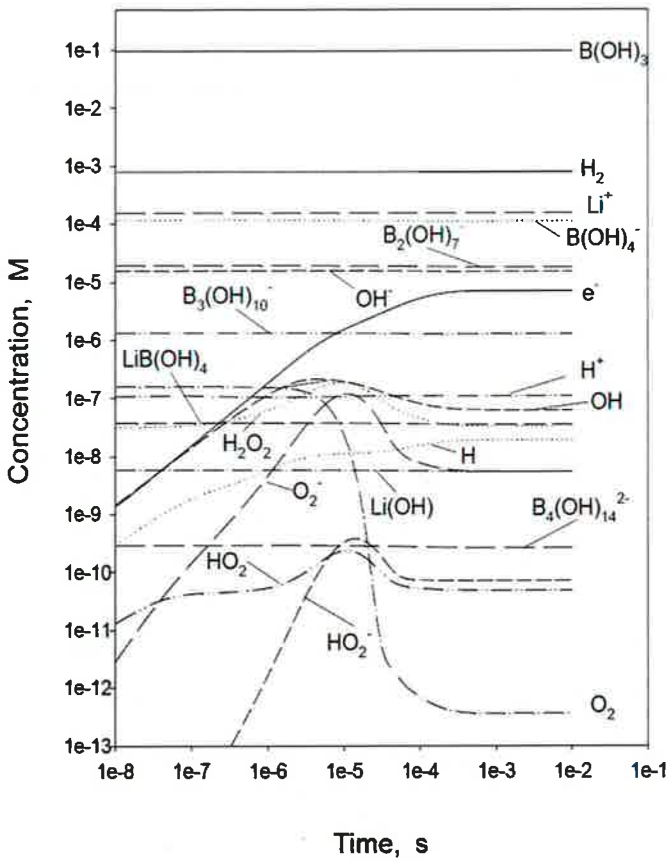


Figure VI.8. Predicted concentrations of radiolysis and pH control species in the primary coolant of a PWR as a function of time towards achieving a local steady state for $C_{B,T}$ = 1500 ppm, $C_{Li,T}$ = 1.5 ppm, T = 300 °C, [H_2] = 25 cm^3/kg STP, $\Gamma_n = 1\times10^{20}$ eV/cm^3s, and $\Gamma_\gamma = 3\times10^{21}$ eV/cm^3s.

We have modeled an operating French PWR's primary heat transport system close to a Westinghouse reactor's design. Some of our data showing the predicted ECP is presented in Figure VI.9. This plot indicates that the most negative ECP is at the bypass grid (S7) rather than in the core, fuel channels, core return, top of the core, bypass tube guide, or bypass hot zone. The value predicted can be a function of the local conditions, such as temperature, flow velocity, etc. In any event, the most negative ECP is -0.80 V_{she}. Similarly, we predicted very negative ECP values of about -0.80 V_{she} for the steam generator tubes, particularly for the hotter entry region. ("hot leg").

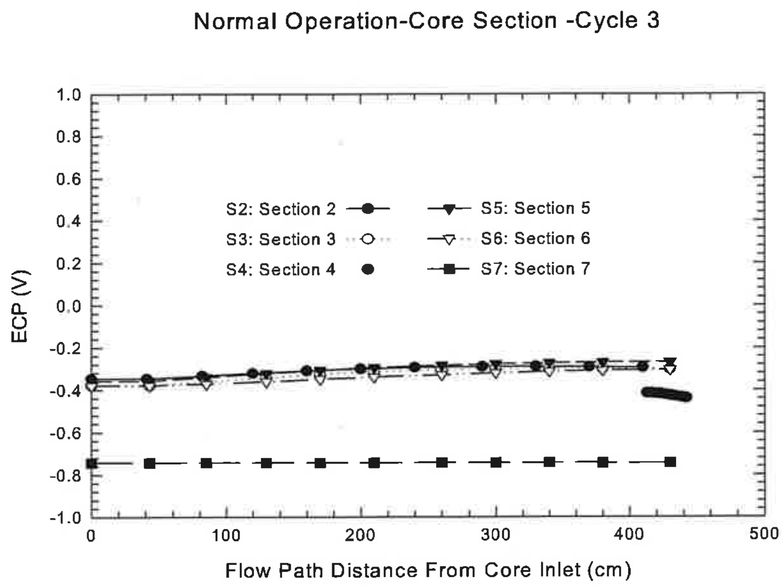


Figure VI.9. Predicted ECP vs distance in a PWR under normal, full power operation. S1 = Core return, S2 = Fuel channels, S3 = Hot zone, S4 = Top part of the core, S5 = By-pass Tube Guide, S6 = By-pass hot zone, S7 = By-pass grid.

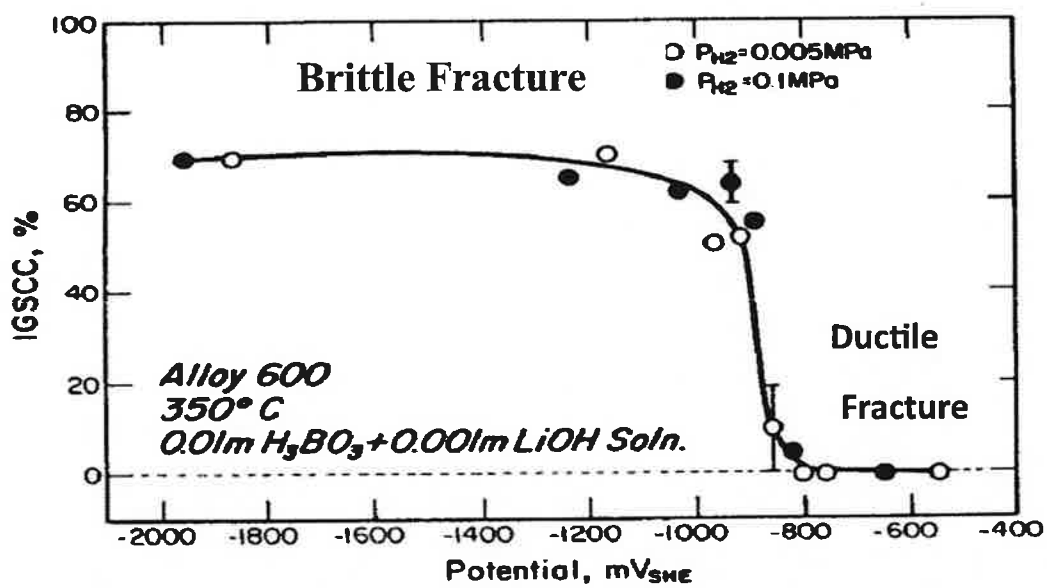
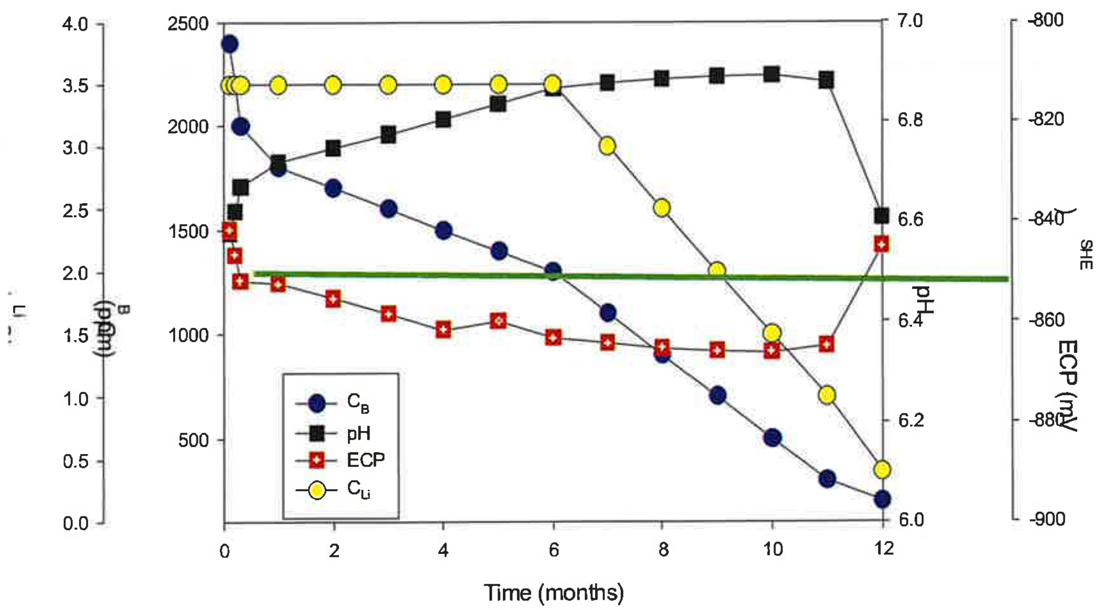


Figure VI.10. Illustration of electrochemically induced hydrogen embrittlement in Alloy 600 in hydrogenated PWR primary coolant. (Totsuka, N and Szklarska-Smialowska, Z (1987)).

Totsuka and Szklarska-Smialowska performed CERT experiments on Alloy 600 samples in PWR environments as a function of (applied) potential, and their data are plotted in Figure VI.10 (Totsuka, N and Szklarska-Smialowska, Z (1987)). They found a transition from ductile to brittle fracture as the potential is more negative. This is a direct confirmation of the theory that Macdonald and coworkers had put forward that the brittle failures experienced in a variety of materials in the primary coolant of PWRs could be traced to the corrosion potential being too negative, either because the pH was too high or, more importantly, because the hydrogen concentration was too high. Indeed, Bertuch, Pang, and Macdonald found (Bertuch, A; Pang, J; and Macdonald, D D (1995)) that only five cc(STP) H_2/kgH_2O is needed to "suppress radiolysis," in contrast to the industry-recommended levels of hydrogen of 5 to 7 times as much. Had the industry accepted our lower figure, also proposed later by some other research groups, the enormous cost suffered by PWR operators might have been avoided.

Following the work of Totsuka and Smialowska, who demonstrated the critical roles of electrochemistry and HIC in PWSCC, (Kim, H S (2007)) modified the PWR_ECP code of Macdonald and Urquidi-Macdonald (Macdonald, D D and Urquidi-Macdonald, M (2007)). to address PWSCC in Korean PWRs. Of specific interest was the relationship between the water chemistry protocol during fuel burnup and the occurrence of PWSCC. Thus, it is known that mill-annealed Alloy 600 steam generator tubes suffer HIC at potentials that are more negative than about -0.85 V_{she}, as shown in Figure VI.11. The potential does indeed approach this critical value in typical PWRs [Figure VI.11 (a)], which depicts the standard "coordinated water chemistry protocol (CWC)" that is in use in PWRs Worldwide. Additional modeling work of (Kim, H S (2007)), shows that under CWC, the ECP of the Alloy 600 steam generator tubes is displaced well below (i.e., is more negative) than -0.85 V_{she} toward the end of a fuel cycle --Figure VI.11 (a), thereby accounting for the rash of PWSCC that has plagued PWR operators in recent years. Those calculations indicate operators may avoid the problem by tailoring the primary circuit chemistry over the fuel cycle. This led to devising the "adjusted water chemistry (AWC)" protocol shown in Figure VI.11 (b). The ECP is predicted to be more positive than the critical potential for brittle fracture over the entire fuel cycle. From Figure VI.11, a significant brittle fracture does not develop until the ECP is more negative than about -0.85 V_{she}—the green lines in Figure VI.11 mark this potential.



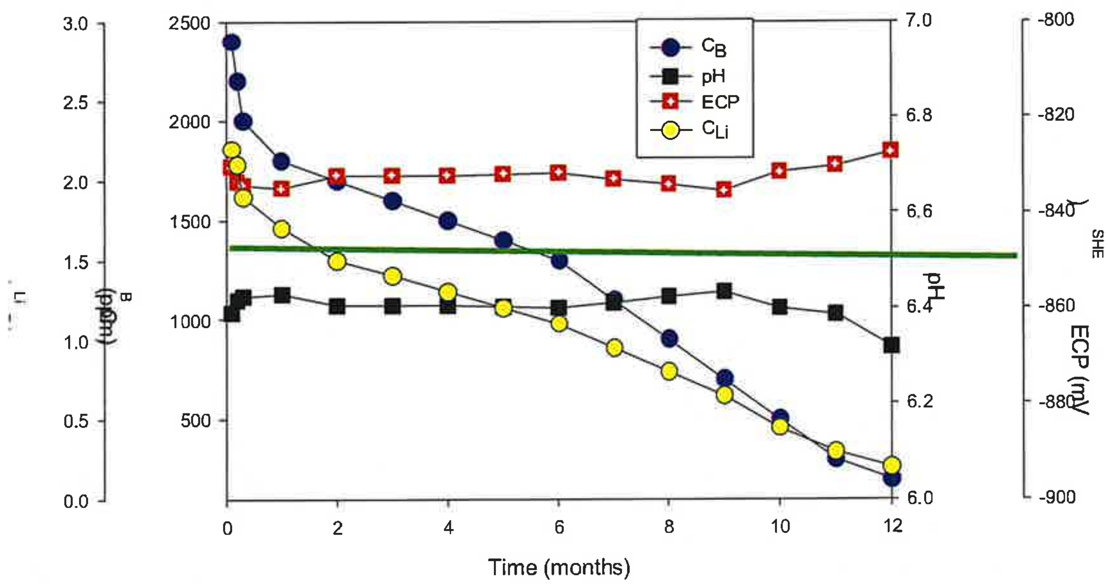


Figure VI.11. (a) Predicted ECP and pH over a fuel cycle for a PWR operating on coordinated water chemistry and (b) adjusted water chemistry. $[H_2] = 25$ cm^3 (STP)/kgH_2O, T = 320 °C (Kim, H S (2007)). The dark green lines are the critical ECP for brittle fracture.

Modeling the type described in this review may mitigate serious problems in reactor operation, as illustrated here regarding PWSCC in PWR Alloy 600 steam generator tubing. PWSCC in the tubing has plagued the operation of those reactors over several decades. Thus, under coordinated water chemistry as defined by the trajectories of the concentrations of boron and lithium over a fuel cycle, which is widely practiced --Figure VI.11 (a), the ECP of the Alloy 600 steam generator tubes in the hot leg remains below (i.e., more negative than) the critical value for PWSCC of -0.85 V_{she} for almost the entire fuel cycle. Hence, the tubes are in a perpetual state of cracking.

On the other hand, if the reactor is operated under the proposed "adjusted water chemistry" protocol, as shown in Figure VI.11 (b), the ECP remains at or above the critical potential of -835 mV_{she} over the entire fuel cycle, thereby mitigating PWSCC in the steam generator tubes. Of course, "balance of plant" issues would need to be addressed to determine whether any unintended consequences could arise from such a change in water chemistry. For example, it is seen from Figure VI.11 (b) that the pH must be reduced by as much as 0.4 of a unit to affect the desired change in the ECP and the effect that this change would have on activity and mass transport and general corrosion in the primary circuit would have to be determined. Nevertheless, the example given above is intended to illustrate how detailed physical-electrochemical modeling may be used to refine the operation of nuclear reactors to avoid costly materials degradation phenomena, and it is hoped that the same can be achieved in the development of fusion power.

I end by summarizing what has been discovered about the impact of water radiolysis in PWR primary coolant and the effect that it is likely to have on the brittle fracture of PWR RPVs. Thus:

- Radiolysis is predicted to produce hydrogen atom concentrations that would require hydrogen pressures far over what has been created on Earth without radiolysis. Thus, the driving force for hydrogen atoms to enter the steel is enormous, and a 7 mm stainless steel liner is expected to be ineffective in preventing neutron flux. Accordingly, we should not be surprised if copious amounts of hydrogen are present in RPVs.
- Suppose hydrogen is found to enter the RPV ferritic steel. In that case, it will diffuse to and recombine to form molecular hydrogen that will pressurize the voids, thereby greatly accelerating the subcritical crack growth rate and hence will shorten the time for the transition of a subcritical crack with $K_I < K_{Ic}$ to a super-critical crack with $K_I > K_{Ic}$. At that point, the crack will propagate unstably and may threaten the integrity of the vessel.
- I recommend that the NRC fund a program to quantify the amount and source of hydrogen in operating reactor RPVs. I believe the best approach would be to use a barnacle electrode on the outside of the vessel.
- The brittle fracture occurs in PWR internal components, and for the same reason, it will happen in the RPV: the entry of atomic hydrogen into the alloy. In the case of the reactor internals in direct contact with the coolant, too much hydrogen in the coolant displaced the electrochemical corrosion potential (ECP) to an excessively negative value. We have shown that redesigning the water chemistry protocol may prevent this form of brittle attack on steam generator tubes.
- The lesson for the RPV community is that even though the ferritic RPV steel is not in direct contact with the coolant (being separated from it by a 7 mm thick stainless-steel liner, that should bring little comfort as hydrogen can transverse the liner with ease. Accordingly, the possibility of a "hydrogen" component to the embrittlement of the RPV steel should always be expected.

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