# EXHIBIT 3 – DECLARATION OF DIGBY MACDONALD

**UNITED STATES OF AMERICA**
**NUCLEAR REGULATORY COMMISSION**
**BEFORE THE SECRETARY**

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| In the matter of | |
| Pacific Gas and Electric Company | Docket Nos. 50-275-LR |
| Diablo Canyon Nuclear Power Plant | 50-323-LR |
| Units 1 and 2 | |

**DECLARATION OF DIGBY MACDONALD, Ph.D**

Under penalty of perjury, I, Digby Macdonald, declare as follows:

## I. PURPOSE AND DESCRIPTION OF MY DECLARATION

1. I have been retained by San Luis Obispo Mothers for Peace (SLOMFP), Environmental Working Group (EWG) and Friends of the Earth (FOE) to evaluate Pacific Gas and Electric Company's (PG&E's) license renewal application of November 7, 2023 with respect to its program for surveillance of the Diablo Canyon Unit 1 reactor pressure vessel. The purpose of my declaration is to explain why, in my expert opinion, PG&E's license renewal application does not contain an adequate plan to monitor and manage the effects of aging due to embrittlement of the Unit 1 reactor pressure vessel (RPV).[1]

2. As discussed below, PG&E's aging management program for the Unit 1 RPV is based upon and continues the surveillance program that PG&E has used during the initial operating license period. In the summer and fall of 2023, I evaluated PG&E's program and found that it was deficient in multiple significant respects, to the point that operation of 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. In a declaration submitted to the NRC on September 14, 2023, I recommended to the U.S. Nuclear Regulatory Commission (NRC) that Unit 1 should be closed until PG&E obtains and analyzes additional information regarding its condition.[2]

3. I continue to hold the opinions stated in my 9/14/23 Declaration, which I hereby adopt and incorporate by reference into this Declaration. As discussed below, I am concerned that the significant defects in PG&E's current RPV surveillance program are perpetuated in the LRA without being addressed or corrected. Therefore, the LRA fails to

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[1] This declaration pertains only to the Unit 1 RPV. I have not studied the Unit 2 RPV in any detail and therefore have not formed an opinion about PG&E's program for managing the aging of that reactor.

[2] Declaration of Digby Macdonald, Ph.D in Support of Hearing Request and Request for Emergency Order by San Luis Obispo Mothers for Peace and Friends of the Earth (Sept. 14, 2023) ("9/14/23 Declaration") (NRC Accession No. ML23257A302). A copy of my 9/14/23 Declaration is attached to this Declaration as Attachment 1.

demonstrate that the effects of aging on the Unit 1 RPV will be managed in a way that is adequate to protect public health and safety. In my expert opinion, Unit 1 should be closed pending withdrawal of Capsule B and other testing to provide reasonable assurance that the Unit 1 RPV is safe to operate now and when renewed operation commences.

4. Section II below provides a statement of my expert qualifications. Section III provides background information on the role of a pressure vessel in a nuclear reactor and NRC guidance and regulations. Section IV provides background information regarding PG&E's existing RPV surveillance program and its license renewal application. Section V summarizes the technical analysis of my 9/14/23 declaration as it applies to PG&E's LRA application. In Section VI, I will address additional concerns that have arisen since then.

## II. EXPERT QUALIFICATIONS

5. I am Professor in Residence at the University of California at Berkeley (UC Berkeley), in the Departments of Nuclear Engineering and Materials Science and Engineering, one of the world's preeminent nuclear engineering programs. I hold a Ph.D. in Chemistry from the University of Calgary in Canada and B.Sc. and M.Sc. degrees also in Chemistry from the University of Auckland in New Zealand. I am a qualified expert in the field of materials science with an emphasis on materials in nuclear power reactors (fission and fusion). My areas of expertise include electrochemistry, thermodynamics, applied fracture mechanics, and corrosion science, with emphasis on the growth and breakdown of passive films, chemistry of high temperature aqueous solutions, electro-catalysis, advanced batteries and fuel cells, stress corrosion cracking and corrosion fatigue, materials for nuclear power reactors, and the deterministic prediction of corrosion damage.
6. My expert qualifications are described in detail in Section II of my 9/14/23 Declaration. And a copy of my curriculum vitae is attached as Appendix A to my 9/14/23 Declaration.

## III. BACKGROUND ON ROLE OF PRESSURE VESSEL AND NRC REGULATORY REQUIREMENTS AND GUIDANCE

7. As discussed in Section IV of my 9/14/23 Declaration, the pressure vessel is a uniquely important and vulnerable component in a nuclear reactor, because it holds water on the highly radioactive reactor core, and because it has no backup if it should crack and lose water during an accident. Therefore, compliance with NRC requirements for monitoring the condition of the plant-specific pressure vessel is essential.

8. For pressure vessels, these regulatory requirements are three-fold and complementary:

    - First, through "Charpy" testing of samples taken from the reactor vessel, the licensee must demonstrate that the change in the "reference" temperature for pressurized thermal shock ($RT_{PTS}$) is below a threshold of 270ºF for axially oriented welds and

300°F for circumferential welds. $RT_{PTS}$ is the temperature at which fracture morphology of the pressure vessel changes from ductile to brittle as its temperature drops from the addition of cooling water during a loss of coolant accident (LOCA). Data for the fracture energy vs. test temperature are determined from Charpy testing of standard specimens (ASTM 185-82) that had been irradiated in capsules located between the reactor core and the inner surface of the RPV and are compared with the unirradiated material and the change from that state is deermined. The capsules are withdrawn at more-or-less equally spaced intervals (typically, every ten calendar years) throughout the reactor life of 32 EFPY (40 calendar years).

- Second, also through Charpy testing, the licensee must demonstrate that the pressure vessel is strong enough to withstand the transient stresses induced by thermal shock of the rapidly changing temperature caused by the addition of cooling water, *i.e.,* that the "upper shelf energy" (USE) will remain above 50 ft-lb.

- Finally, every ten years, the licensee must conduct ultrasound testing (UT) inspections of the most vulnerable part of the reactor vessel, the welds around the beltline, to examine for flaws and cracks. NRC guidance appropriately provides that the schedules for these inspections may be relaxed only upon a verifiable demonstration that safety will not be jeopardized.

9. These three types of tests and inspections are complementary in three significant respects. First, each of the measured phenomena makes a distinct and significant contribution to determining the vulnerability of a pressure vessel to cracking. Second, while the reference temperature and USE calculations are both derived from the same Charpy tests, the method of analysis for each is different; and of course, the UT inspections involve completely different methods of acquiring and analyzing data. Third, each type of test or inspection has a different level of reliability. As discussed in Section V.A.2 of my 9/14/23 Declaration, my calculations show that Charpy tests are not particularly sensitive to the extent of embrittlement. Therefore, their results should not be substituted for UT inspections, nor should they be used to justify an extension of the schedule for UT inspections. The three types of data must be considered in unison because they convey important, complementary information on the safety of the RPV.

10. Adequate monitoring of the condition of the pressure vessel is particularly important in the case of Diablo Canyon Unit 1 because the composition of the welds in the pressure vessel was found to be defective at the time it was installed by having excessive copper and nickel. Not surprisingly, in 2006, the NRC identified the Unit 1 pressure vessel among the most embrittled, with only 14 of 72 PTS reference temperatures as high as or higher than Diablo Canyon Unit 1. U.S. NRC 2007. And today, half of those 14 reactors are closed.

11. To obtain NRC approval of a renewed operating license, NRC regulation 10 C.F.R. 54.21 requires that licensees must demonstrate that that the effects of aging due to embrittlement of the RPV and associated internals will be adequately monitored and managed during the entire license renewal term. Required measures for aging

management include time-limited aging analyses (TLAAs). As defined in the regulations, TLAAs are "calculations and analyses" that "(1) Involve systems, structures, and components within the scope of license renewal, as delineated in § 54.4(a); (2) Consider the effects of aging; (3) Involve time-limited assumptions defined by the current operating term, for example, 40 years; (4) Were determined to be relevant by the licensee in making a safety determination; (5) Involve conclusions or provide the basis for conclusions related to the capability of the system, structure, and component to perform its intended functions, as delineated in § 54.4(b); and (6) Are contained or incorporated by reference in the CLB." Thus, TLAAs depend significantly on the calculations and analyses developed in the initial license renewal term.

## IV. BACKGROUND REGARDING PG&E'S LICENSE RENEWAL APPLICATION

12. In Section 4.2, PG&E's license renewal application provides a "reactor vessel neutron embrittlement analysis." [p. 4.2-1] Here and elsewhere in the LRA, PG&E makes several statements that relate its current RPV monitoring program to the monitoring program in the license renewal period. For instance, at page 4.2-1 the LRA states that current calculations regarding ΔRTNDT and USE values are updated for the license renewal application:

> For DCPP Units 1 and 2, the reactor vessel material ΔRTNDT and USE values, calculated on the basis of 32 effective full power years (EFPY) neutron fluence, are determined as part of the CLB and support safety determinations and TS operating limits. Therefore, these calculations are TLAAs [time-limited aging analyses]. For LR, these must be updated to account for the fluence expected to occur during 60 years of plant operation (54 EFPY).

13. Similarly, on page 4.2-2 the LRA states:

> The current license period reactor vessel embrittlement analyses that evaluate reduction of fracture toughness of the DCPP Units 1 and 2 reactor vessel beltline materials are based on predicted 40-year EOLE fluence values. The fluence analysis and the neutron embrittlement analyses that are based upon the fluence analysis are TLAAs as defined by 10 CFR 54.21(c) that must be evaluated for the increased neutron fluence associated with 60 years of operation. These TLAAs include the analyses for neutron fluence, PTS USE, adjusted reference temperature (ART), and pressure-temperature limits including low temperature over pressure protection analysis.

14. On pages 4.2-2 – 4.2-3 the LRA states:

> The last capsule withdrawn and tested from Unit 1 was Capsule V at the EOC 11 in 2002. At that point, Unit 1 Capsule V had an exposure equivalent to 32.25 EFPY of operation. The results were documented in WCAP-15958 (Reference 4.9.1).

> This exposure is less than that expected at EOLE. Therefore, to obtain capsule data for a neutron fluence of between one and two times the peak reactor vessel wall neutron fluence at EOLE, in PG&E Letter DCL-23-038, dated May 15, 2023 (Reference 4.9.2), PG&E requested NRC approval of the Unit 1 Capsule B withdrawal schedule. NRC approved the requested withdrawal schedule by letter dated July 20, 2023 (Reference 4.9.3).

In the context of this discussion, the LRA also states that PG&E "will submit the results of the tested surveillance capsule to NRC in accordance with 10 CFR 50, Appendix G and H." *Id.* Thus, the previous schedule for withdrawal of Capsule B is relied on by PG&E for its plans. However, I am unable to locate any commitment by PG&E to a deadline for removing and testing Capsule B.

15. On page 4.3-3, the LRA also references the current monitoring program in stating:

> For LR, updated fluence projections based upon 54 EFPY were prepared for use as inputs in the neutron embrittlement analyses for 60 years of operation.
>
> The reactor vessel beltline neutron fluence values for 60 years of operation were calculated for DCPP Units 1 and 2 reactor vessel beltline material in WCAP-17299-NP (Reference 4.9.5). These fluence data tabulations include fuel cycle specific power distributions through the end of Cycle 16 for Units 1 and 2, as well as fluence projections at several intervals out to 54 EFPY. The analysis methods used to calculate the predicted 60-year DCPP Units 1 and 2 vessel fluence values satisfy the requirements set forth in RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," Revision 0. These methodologies have been approved by the NRC and are described in WCAP-14040-NP-A, Revision 4.
>
> In addition, in accordance with 10 CFR 50, Appendix H, any materials exposed to a neutron fluence exceeding $1.0 \times 10^{17}$ n/cm$^2$ (E > 1.0 MeV) must be monitored to evaluate changes in fracture toughness. Reactor vessel materials that are not traditionally thought of as being plant limiting were also evaluated in WCAP-17299-NP (Reference 4.9.5) to determine their cumulative fluence values at 54 EFPY. Fluence calculations were performed for the DCPP Units 1 and 2 reactor vessels to determine if the fluence at specific locations will exceed $1.0 \times 10^{17}$ n/cm$^2$ (E > 1.0 MeV) at 54 EFPY. The materials exposed to fluences that exceed this threshold are referred to as the extended beltline materials.

16. On page B.2-33, the LRA program for monitoring the loss of fracture toughness to the existing program, which provides a baseline:

> The AMP will not directly monitor for loss of fracture toughness that is induced by thermal aging or neutron irradiation embrittlement. Instead, the impact of loss of fracture toughness on component integrity will be indirectly managed by: (1)

using visual or volumetric examination techniques to monitor for cracking in the components, and (2) in cases where cracking is detected in the components and is extensive enough to necessitate a supplemental flaw growth or flaw tolerance evaluation, applying applicable reduced fracture toughness properties, including reductions for thermal aging or neutron embrittlement. The AMP will use physical measurements to monitor for any dimension changes due to void swelling or distortion.

However, I am unable to locate any mention in the LRA of how PG&E's general commitment to conduct ultrasonic testing of beltline welds relates to the schedule for UT inspections that is discussed below in par. 19(e).

17. At page B.2-95, the LRA states:

> The DCPP Reactor Vessel Surveillance AMP provides guidance for removal and testing or storage of material specimen capsules. All capsules that have been withdrawn, tested, and not otherwise donated, were stored in conformance with 10 CFR 50 requirements. For Unit 1, the last capsule is expected to be withdrawn and tested having accumulated 1-2 times the peak reactor vessel neutron fluence at 60 years of operation (NRC approved Unit 1 Capsule B withdrawal at approximately 96.19 - 101.01 EFPY). The remaining four standby capsules in Unit 1 have low lead factors, will remain inside the vessel throughout the vessel lifetime, and will be available for future testing. There are no capsules remaining in the Unit 2 vessel.

18. In Enclosure 3, WCAP-17315-NP, the LRA also asserts that PG&E meets the criteria in 10 C.F.R. § 50.61 and therefore does not need to use alternative "less restrictive" criteria in 10 C.F.R. § 50.61a. *Id.* at 6-1.

## V. SCIENTIFIC ANALYSIS

19. As demonstrated above, the LRA incorporates and depends heavily on previous tests and analyses of RPV embrittlement at DCPP and other reactors for its conclusion that (a) the Unit 1 RPV is entering the period of license renewal in a reasonably safe condition that complies with NRC regulations and that (b) its condition can be adequately managed throughout the license renewal term. In my expert opinion, however, these conclusions are not justified. To summarize my views:

    a. As discussed in Section V.A of my 9/14/23 Declaration, in 2002 PG&E withdrew and tested "coupons" or weld samples from the Unit 1 pressure vessel and conducted Charpy tests for PTS reference temperature and USE. PG&E (2003). In 2003, PG&E reported that it had calculated a limiting $RT_{PTS}$ value of 250ºF for the limiting weld 3-442C. *Id.* Thus, PG&E predicted that in 2021 (the expected retirement date for Unit 1 at that time), the reference temperature for Unit 1 would be slightly more than 10º below the screening limit of 270 ºF. Taking into consideration a reasonable margin of error of about 12 ºF (as estimated by inspection of the Charpy curves), PG&E's test showed that Unit 1 would be approaching the limit at the end of its operating life.

b. Nevertheless, PG&E discounted the data as "not credible." *Id.* However, PG&E may have found that the data *were* credible if it had applied standard scientific and NRC guidance for its evaluation. U.S. NRC (1998). PG&E's failure to apply this well-established and reasonable guidance is both inexplicable and gravely concerning, given that the $RT_{PTS}$ data indicated a serious degree of embrittlement. The NRC Staff's approval of PG&E's disregard of the data is also puzzling, given that PG&E had ignored the agency's guidance.

c. Instead of crediting the data it had gathered from Unit 1, PG&E substituted generic data and data from other reactors. As discussed in Section V.C of my 9/14/23 Declaration, PG&E's reliance on substitute data from other reactors was also unreasonable, especially for a period that stretched across decades. Regardless of their initial similarities, all nuclear reactors soon become individualized by unique operating conditions and histories. At the very least, PG&E should have applied a larger error band to any reference temperature calculations that were based on generic data or data from so-called "sister" reactors. Instead, PG&E is doubling down on its reliance on data from sister reactors.[3]

d. As also discussed in Sections V.C and V.D of my 9/14/23 Declaration, the results of the 2003 evaluation of the Charpy tests should have motivated PG&E to speed up its schedules for obtaining more data to get a better sense of the pressure vessel's condition. At the very least, PG&E should have adhered to its approved schedule for the next capsule extraction and Charpy test in approximately 2009.

e. As also discussed in Sections V.C and V.D of my 9/14/23 Declaration, PG&E should have ensured that the most recent (2005) UT inspection -- which identified "one indication . . . in the beltline region" (PG&E (2014)) -- would be followed on schedule with another beltline inspection in 2015. Yet, PG&E repeatedly sought and obtained extensions of time for these measures: the next Charpy test has now been rescheduled from 2009 to 2023 or 2025, depending on whether PG&E is able to withdraw the capsule in 2023 (U.S. NRC (2023)); and the next UT inspection is scheduled for 2025 (U.S. NRC (2015)).

[3] In 2011, eight years after informing the NRC that the data from Capsules S, Y, and V were "not credible" (PG&E (2003)), PG&E relied on data from another reactor to assert that Unit 1 can be safely operated to the end of a 20-year renewal period. PG&E (2011). *See* Table 4.2-4, showing that the limiting weld 3-442C does not meet or approach the regulatory limit of 270 ºF until 54 EFPY, the equivalent of 60 years of operation. The reference document for this prediction is WCAP-17315-NP (Westinghouse (2011)), which relies in part on data from the Palisades reactor to project $RT_{PTS}$ values for the end of the Unit 1 license term.

f. In both cases, the extensions leave an unacceptable gap of 20 years between the tests or inspections. In my professional opinion, two decades is an unacceptable amount of time, for two reasons. First, there was no reason for PG&E to rely on questionable generic data or data from so-called "sister" reactors for more than a short time after the 2003 evaluation. PG&E could have and should have obtained more plant-specific data by now. Second, the condition of the pressure vessel may change significantly over a single decade. *See* Section V.C. of my 9/14/23 Declaration.

g. In addition, the fact that PG&E's 2005 UT inspection of the pressure vessel were "essentially identical" to an inspection done 10 years earlier and yielded only one "indication" of cracking (PG&E (2014)) should have prompted PG&E to evaluate whether the UT inspection was faulty and needed to be repeated. It is reasonable to expect many more indications of voids and cracks, and that they would increase over time. *See* Section V.B. of my 9/14/23 Declaration.

20. Under these circumstances, it is my expert opinion that the NRC currently lacks an adequate basis to conclude that Diablo Canyon Unit 1 can be operated safely. The NRC Staff's recent decision to allow PG&E to postpone the next Charpy test for Unit 1 until 2025 (U.S. NRC (2023)) is unjustified. As in my 9/14/23 Declaration, I continue to believe that in order to protect the public from the unacceptable risk of a core meltdown accident caused by pressure vessel cracking and fracture during a loss of coolant accident (LOCA), the NRC should (a) order the immediate closure of the reactor by accelerating a maintenance shutdown now scheduled for October, (b) require that the reactor must remain closed pending completion of the next scheduled Charpy tests, (c) ensure that any coupons or capsules that have been withdrawn but were not tested are subject to Charpy tests, (d) account for the data provided by the wedge opening loading (WOL) specimens and the tensile specimens that were scheduled to be contained in the capsules, and (e) ensure that any remedial steps taken by PG&E to address the condition of the Unit 1 reactor pressure vessel are subjected to rigorous review by the NRC Staff, the Advisory Committee on Reactor Safeguards (ACRS), and the general public. *S*ee Section VI.A of my 9/14/23 Declaration.

21. For the same reasons, it is also my expert opinion that the NRC lacks a reasonable basis to approve PG&E's license renewal application. Unless and until the NRC establishes that the Unit 1 pressure vessel can operate with a reasonable degree of safety, it has no basis to permit continued operation in a license renewal term.

22. Finally, in Section V.E of my 9/14/23 Declaration, I have offered "information" that I believe will "improve the accuracy of the $RT_{PTS}$ value significantly." In my professional opinion, the newly developed method of nano-indentation promises to be capable of far more extensive results from a single specimen than the conventional Charpy Impact Test methods prescribed by NRC regulations. The more extensive data will permit rigorous statistical analysis, something that is not possible with Charpy because of the limited number of specimens that can be accommodated in a capsule. Importantly, this method has already been applied by Professor Peter Hosemann of the Department of Nuclear Engineering, University of California, Berkeley and found to be sensitive to the change in physical properties of PWR RPV steels brought about by radiation embrittlement.

Accordingly, in my professional opinion, the technique requires further application in the field to define and quantify its advantages.

## VI. ADDITIONAL CONCERNS

23. The following additional concerns have arisen since I prepared my 9/11/23 Declaration.

24. **Additional concern 1.** PG&E's strategy (and the strategy of all other reactor operators) for formulating the "lead factor" used to estimate the fluence to a certain level of embrittlement has relied on the assumption that embrittlement damage accrues in a non-Markovian manner. A Markovian process is a stochastic model describing a sequence of possible events, in which the probability of each event depends only on the state attained in the previous event; that is, what happens in the present depends on what happened in the immediate past. But this assumption is unproven and likely erroneous. Thus, it is tacitly assumed when carrying out fluence calculations that the reactor is operating at 80 % of the full-power output, thereby fulfilling the condition for a non-Markovian process *a priori*. However, what happens if the initial fluence damage was accrued at a power level that is significantly below the 80 % level, as is the case for the DCPP during low power testing? Furthermore, the accrual of radiation damage is a highly non-linear phenomenon as evidenced by the shape of the ETC, Equation (1), shown schematically in Figure 1.

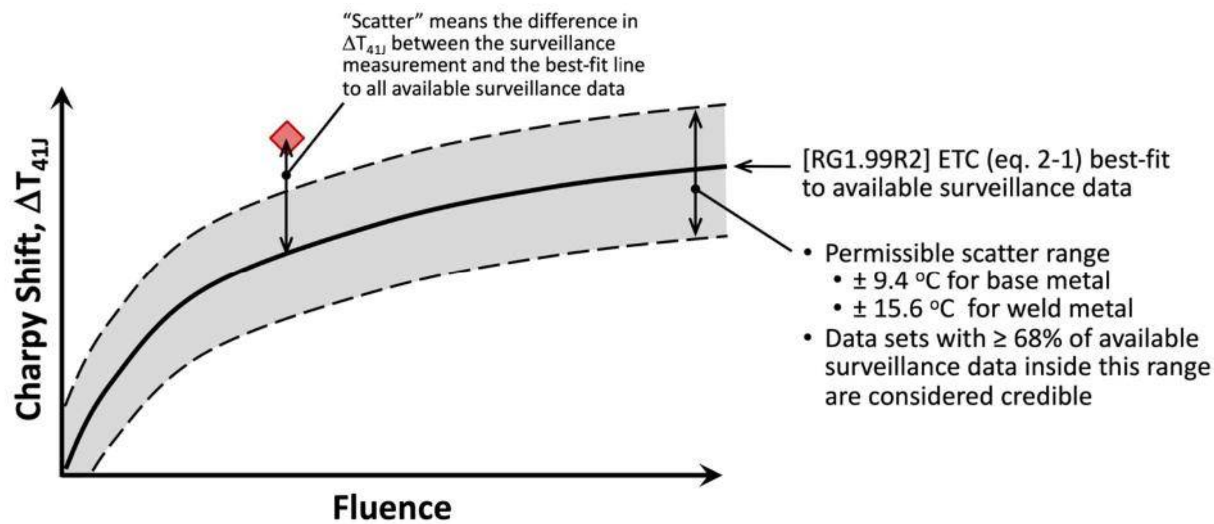


**Figure 1:** Expected variation of $\Delta RT_{NDT,41J}$ ($\Delta T_{41J}$) with fluence for Charpy specimens in a surveillance capsule from DCPP, Unit-1 as specified by the Embrittlement Trend Curve (ETC), Equation (1) according to RG1.99-Rev. 2. The broken lines at any given fluence correspond to the mean value of RTNDT $\pm\sigma$, where $\sigma$ is the standard deviation established from the surveillance data for many PWRs in the US fleet. $\sigma$ = standard deviation of 9.4 °C (17 °F) for base metal and 15.6 °C (28 °F) for weld metal as depicted in the figure.

25. These data are plotted on a single graph, shown schematically in Figure 1 and it is assumed that $\Delta ART_{NDT,41J}$ (equal to $\Delta T_{41J}$) should vary with fluence (*f*) following the RG1.99-R2 ETC (Embrittlement Trend Curve) given by Equation (1)

$$\Delta RT_{NDT} = CFxf^{(0.28-0.1logf)} \quad (1)$$

where CF is the chemistry factor that adjusts $\Delta RT_{NDT}$ $(i.e., ART_{NDT})$ elements in the alloy being irradiated that predispose the substrate to neutron irradiation embrittlement (primarily Cu and Ni).

26. Thus, this ETC shows that for a given increment in fluence, the damage accrued, as measured by the shift in the $ART_{NDT}$ by decreasing amounts from left to right, thereby satisfying the conditions for a non-Markovian process. The capsules are in specific locations in the core between the inner surface of the vessel and the thermal shield, as displayed in Figure 2 for DCPP, Unit 1.

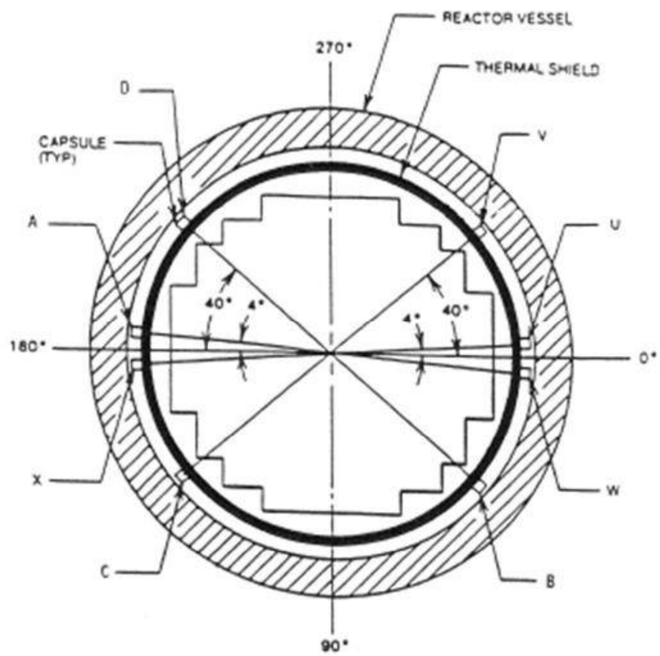


**Figure 1:** Arrangement of surveillance capsules *A, B, C, D, U, V, W,* and *X* in Diablo Canyon Unit 1 NPP. From PG&E, Capsule V Report, WCAP15958.

27. This issue is further complicated by the fact that because of the cylindrical geometry of the core, the neutron flux decreases with $1/r^2$, where $r$ is the radial distance from the core centerline. Thus, the fluence experienced by a capsule depends on its exact radial location with the highest fluence being experienced by the location closest to the core center line. Thus, as the reactor power level changes, not only does the fluence experienced in a capsule also change because of the decrease in the neutron flux (E > 1 MeV),but that at the RPV wall changes by a different amount because of the lead factor. Because of this, a capsule will generally experience a time-varying fluence that is greater than the inner surface of the RPV with the ratio defining the lead factor as $LF = \Phi_{cap}/\Phi_{vw}$, where $\Phi_{cap}$ is the neutron flux at the capsule and $\Phi_{vw}$ is the flux at the RPV inner wall, but one that has a different time-variance than that at the wall. Lead factors

typically range from 2 to 4 in value. This allows reactor operators to "look into the future" by rearranging the above expression to read $\Phi_{vw} = \Phi_{cap}/LF$. Thus, for $LF = 2$, for example, and the end of Life (EoL) is 32 EFPYs (40 calendar years), extraction of the capsule at 16 EFPYs will yield the fluence at the inner RPV wall equivalent to that at the EoL (32 EFPYs). Likewise, if the capsule was extracted at 27 EFPYs, the fluence at the inner RPV wall would be equivalent to that at 48 EFPYs or the end of the first 20-year life extension of the reactor.

28. **Additional Concern 2:** The appearance of Extrema in Capsule V CGraphs and Tables is troubling. Given the discovery of the extrema in $ART_{NDT,30J,}$ and USE data (maximum in the reference temperature shift and a minimum in USE) and recalling the fact that PG&E rushed to deem all of its pre-2004 surveillance data as "not credible" in its interpretation of RG1.99 Criterion 3, this issue should be revisited. The DCPP Unit 1 surveillance data were credible but were summarily dismissed in 2003 when it became clear that the 2003 Capsule V results were not favorable ($RT_{PTS}$=250.9°F). Although these data were also confirmed by the Capsule S Report ($RT_{PTS}$=258°F), PG&E argues that after they developed a new best-fit curve procedure ("CV Graph", a new ETC?), the Capsule Y data that they had previously stated was within RG1.99 predictions was now no longer within the ±1-sigma standard deviation allowed under RG1.99 Criterion 3.

29. In reality, this one data point for the most limiting weld was only 2 °F out of range. Yet this fairly minor "outlier" was used as the basis for discrediting all of their prior data sets through a cascading argument that is entirely contrary to NRC procedures for dealing with outliers and how to deem data not credible as per GL9201. To my knowledge, PG&E never actually conducted a detailed analysis of the outlier datum as required by regulations. One must ask, "Why is it assumed that regulations were precisely followed, on what appears to be a biased supposition?"

30. **Additional Concern 3:** For all the analyses that I have seen for RG1.99-R2 or 10 CFR 50.61/50.61(a), a serious discussion of errors is seldom included. Thus, beltline components that have a regulatory limit of 270 °F, for example, are deemed to have passed even though they display an $\Delta A_{DT,40J}$ of 260 °F. However, given the empirical nature of the field, it is important to determine on a well-founded statistical basis whether the difference of 10 °F is significant. Although error assessment is best done on a statistical basis, particularly if the failure mechanism is poorly defined and the errors cannot be calculated deterministically. A back-of-the-envelope calculation suggests that an error in $\Delta ART_{NDT,40J}$ of ± 30 °F may not be unreasonable. If so, can we state categorically that a range in $\Delta ART_{NDT,40J}$ of 60 °F, with the upper end of this range being at 270 °F, does not indicate that the vessel has not reached the regulatory limit? This issue should be addressed and resolved before license renewal is approved.

31. **Additional Concern 4:** I am concerned that PG&E has not addressed the potentially significant role of hydrogen in the embrittlement/crack propagation process, either in the LRA or previous documents on which PG&E relies. Thermally-embrittled ferritic and martensitic steels are notoriously susceptible to hydrogen embrittlement (HE) and hydrogen-induced cracking (HIC) in many other industrial-scale systems, such as the oil

and gas production, transportation, and storage systems that they are identified as being principal modes of failure of metallic structures. That HIC occurs in reactor internals, such as those components that are manufactured from Alloy 600 and 182, for example, appears to be indisputable given the work of Totsuka and Smialowska.[4] And although it is often claimed that "no evidence exists" for the role of corrosion in the failure of RPVs, great caution must be exercised because that was precisely that position of many concerning the ID cracking of steam-generator tubing until they were proven wrong by the work of Totsuka and Smialowska in 1987.

*Executed in Accordance with 10 C.F.R. § 2.304(d)*
By Digby Macdonald March 4, 2024

[4] N. Totsuka and Z. Szklarska-Smialowska, Effect of Electrode Potential on the Hydrogen-Induced IGSCC of Alloy 600 in an Aqueous Solution at 350 C, *CORROSION* (1987) 43 (12): 734–738.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
BEFORE THE COMMISSION

In the matter of
Pacific Gas and Electric Company — Docket No. 50-275
Diablo Canyon Nuclear Power Plant, Unit 1

**DECLARATION OF DIGBY MACDONALD, Ph.D**
**IN SUPPORT OF HEARING REQUEST AND**
**REQUEST FOR EMERGENCY ORDER**
**BY SAN LUIS OBISPO MOTHERS FOR PEACE**
**AND FRIENDS OF THE EARTH**

September 14, 2023

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APPENDIX A: CURRICULUM VITAE

APPENDIX B: REFERENCE LIST

## GLOSSARY OF ACRONYMS

| | |
|---|---|
| ACRS | Advisory Committee on Reactor Safeguards |
| AECL | Atomic Energy of Canada Ltd |
| ANN | artificial neural network |
| $ART_{NDT}$ | Adjusted Nil Ductility Transition Temperature |
| ASME | American Society of Mechanical Engineers |
| ASTM | American Society of Testing and Materials |
| BWR | Boiling Water Reactor |
| CANDU | CANada Deuterium Uranium |
| CEFM | Coupled Environment Fracture Model |
| CECFM | Coupled Environment Corrosion Fatigue Model |
| CGR | crack growth rate |
| CIT | Charpy Impact Test |
| CRUD | Chalk River Unidentified Deposit |
| ECCS | emergency core cooling system |
| ECP | electrochemical corrosion potential |
| EoE | extent of embrittlement |
| EOL | end of operating life |
| FAVOR | Fracture Analysis of Vessels |
| FoE | Friends of the Earth |
| HAZ | heat affected zone |
| HIC | hydrogen-induced cracking |
| HLNW | high-level nuclear waste |
| IGSCC | inter granular stress corrosion cracking |
| INL | Idaho National Laboratory |
| J | Joules, SI unit of energy |
| MPM | Mixed Potential Model |
| NPP | nuclear power plant |
| NRC | U.S. Nuclear Regulatory Commission |

| | |
|---|---|
| ORNL | Oak Ridge National Laboratory |
| PG&E | Pacific Gas and Electric Company |
| PTS | pressurized thermal shock |
| PWR | pressurized water reactor |
| RFO | refueling outage |
| RRE | rate of radiation embrittlement |
| RoA | reduction of area upon fracture |
| RPV | reactor pressure vessel |
| $RT_{NDT}$ | Nil Ductility Transition Temperature |
| $RT_{PTS}$ | Reference Temperature for Pressurized Thermal Shock |
| SCC | stress corrosion cracking |
| SCK CEN | Belgian Nuclear Research Centre |
| SG | steam generator |
| SLOMFP | San Luis Obispo Mothers for Peace |
| SRM | Standard Reference Material |
| SS | stainless steel |
| SSM | Swedish Radiation Safety Authority |
| TWCF | through-wall cracking frequency |
| STP | standard temperature and pressure |
| USE | upper shelf energy |
| UT | ultrasonic testing |
| VP | vice president |
| VPM | void pressurization model |
| WOL | wedge opening loading |
| YS | yield strength |

## I. INTRODUCTION

1. I have been retained by San Luis Obispo Mothers for Peace (SLOMFP) and Friends of the Earth (FOE) to evaluate changes in Pacific Gas and Electric Company's (PG&E's) program for surveillance of the Diablo Canyon Unit 1 reactor pressure vessel and the adequacy of the justifications provided by the U.S. Nuclear Regulatory Commission (NRC) in support of those changes. My analysis, provided below, supports the Hearing Request and Request for Emergency Action submitted by SLOMFP and FoE to the NRC.

2. The purpose of my declaration is to explain the reasons why, in my professional opinion, 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. Therefore, the reactor should be closed until PG&E obtains and analyzes additional data regarding its condition.

## II. STATEMENT OF PROFESSIONAL QUALIFICATIONS

1. I am Professor in Residence at the University of California at Berkeley (UC Berkeley), in the Departments of Nuclear Engineering and Materials Science and Engineering, one of the world's preeminent nuclear engineering programs. I hold a Ph.D. in Chemistry from the University of Calgary in Canada and B.Sc. and M.Sc. degrees also in Chemistry from the University of Auckland in New Zealand. A copy of my curriculum vitae is attached as Appendix A.

2. I am a qualified expert in the field of materials science with an emphasis on materials in nuclear power reactors (fission and fusion). My areas of expertise include electrochemistry, thermodynamics, applied fracture mechanics, and corrosion science, with emphasis on the growth and breakdown of passive films, chemistry of high temperature aqueous solutions, electro-catalysis, advanced batteries and fuel cells, stress corrosion cracking and corrosion fatigue, materials for nuclear power reactors, and the deterministic prediction of corrosion damage. My experience with the study of corrosion damage includes a wide range of damaging events, including stress corrosion cracking of thermally-embrittled reactor pressure vessel steels and of thermally (weld)-sensitized austenitic stainless steel components in the coolant circuits of water-cooled nuclear power reactors. Radiation embrittlement is often mimicked in the laboratory by using thermal embrittlement to the same physical properties (hardness, yield strength, etc.). That is common practice when access to a nuclear reactor or another high energy neutron ($E > 1$ MeV) source is not available, which is often the case in academia. Since completing my Ph.D. in 1969, I have held multiple positions related to nuclear engineering and materials science, which are listed in my curriculum vitae. Most recently, from 2003 to 2012, I was Distinguished Professor of Material Science and Engineering Director for the Center for Electrochemical Science and Technology at Penn State University, again with an emphasis on materials in nuclear power reactors.

3. I have written over 1,000 papers and four books, and I hold eleven patents. My book *Transient Techniques in Electrochemistry* was the foundational text in the study of electrochemical systems using current and voltage perturbation techniques. These

techniques have been used to study certain corrosion-related phenomena in nuclear materials, such as the hydrogen embrittlement of high strength steels and alloys. In 2003, during my tenure at Penn State, I received the U.R. Evans Award, the highest award in the field of corrosion science and engineering, from the Institute of Corrosion in the United Kingdom. In 2011, I was also nominated for a Nobel Prize in chemistry for my work in the passivity of metals in reactive environments and for explaining how such metals (iron, chromium, nickel, copper, zinc, aluminum, zirconium, titanium, etc.) can form the basis of our reactive metals-based civilization. In fact, I reduced that issue to a single mathematical inequality.

4. Regarding nuclear reactors, I developed the Coupled Environment Fracture Model (CEFM) and the Coupled Environment Corrosion Fatigue Model (CECFM) to deterministically model stress corrosion and corrosion fatigue crack growth rate (CGR) in both boiling water reactor (BWR) and pressurized water reactor (PWR) primary coolant circuits. In the case of BWR coolants, a student and I performed an artificial intelligence analysis (using an artificial neural network) of CGR data from both field and laboratory sources. For the CGR in sensitized Type 304 stainless steel (SS), we showed that the CEFM could predict CGR at least as accurately as it can be measured and a similar result was obtained for the CECFM. To my knowledge, the CEFM and the CECFM are the only deterministic models that are currently available for accurate, first principles calculation of CGR in BWR primary coolant circuits. I have used the CEFM to model the evolution of inter granular stress corrosion cracking (IGSCC) damage in 14 operating BWRs worldwide and where comparison with plant data can be made, the agreement between calculated and observed damage is excellent.

5. For PWR primary coolant circuits, I have concentrated on addressing the Alloy 600 steam generator issues by developing the Void Pressurization Model (VPM), a fully deterministic model, to calculate hydrogen-assisted SCC in Alloy 600 that is in contact with primary coolant. Comparison with experimental CGR data again shows that the VPM is also capable of accurately predicting CGR in mill-annealed Alloy 600 under PWR primary coolant conditions. I and a student then developed a Mixed Potential Model (MPM) and demonstrated that because of (a) the large amount of hydrogen that is added to the coolant [25 cc (STP) $H_2$/kg $H_2O$)] and (b) the pH vs fuel burnup protocol commonly employed (the Coordinated Water Chemistry Protocol), the corrosion potential drops below the critical potential for hydrogen-induced cracking (HIC) in the alloy, thereby rendering crack growth spontaneous with the eventual failure of the component (*e.g.*, steam generator tube). We further demonstrated that to maintain the corrosion potential above the critical cracking potential throughout a fuel cycle and thereby address the problem of primary side cracking in steam generator (SG) tubing, the solution is to tailor the coolant hydrogen concentration and/or to modify the pH vs fuel burnup trajectory (by controlling the Li content of the coolant). The MPM is also applicable to analyzing the embrittlement of highly cold-worked Type 316 SS baffle bolts and high alloy hold-down spring in the core structure, for example. Fracture of these, and other components like them (*e.g.*, radiation embrittled RPVs), might be inhibited by the

judicious tailoring of the primary water chemistry to ensure that the corrosion potential always remains more positive than the critical potential for HIC in these components throughout the fuel cycle. Coolant-side chemical and electrochemical effects to the cracking of embrittled RPVs are all but ignored in the current NUREGs.

6. At the beginning of my career (1971 – 73), I was employed by Atomic Energy of Canada Ltd (AECL) and became heavily involved in resolving the activity transport problem at the Douglas Point CANDU prototype. In this capacity, in 1971 (est.), I proposed a "redox shock" strategy for removing the activated "CRUD" (Chalk River Unidentified Deposit) from the boilers so it could be collected on the filters that are designed to hold activated corrosion products. This resulted in an immediate reduction in the γ-photon radiation field in the boiler room thereby (as expressed to me by a site VP of AECL) "saving the CANDU program". For this accomplishment, I received in 1993 the prestigious W.B. Lewis Memorial Lecture from Atomic Energy of Canada, Ltd., "in recognition of [his] contributions to the development of nuclear power in the service of mankind." I was only the sixth awardee, with four previous winners being Nobel Laureates. To my knowledge, the redox shock strategy was the first example of electrochemical control in an operating nuclear power plant (NPP).

7. I have been heavily involved as an expert consultant on various reactor issues, including hot-shortness cracking in the Perry Unit 1 BWR suppression pool, flow-assisted corrosion at Surry Unit 1, out-of-specification water chemistry at Calvert Cliffs, and others. Additionally, a colleague and I raised a concern with the continued operation of the Doel-3 and Tihannge-2 PWRs in Belgium, which both contain "hydrogen flakes" in the pressure vessels. Bogearts (2022). Ultrasonic testing (UT) examination over the years indicated that both the number density and the sizes of the flakes had increased with time, but it was argued by our opponents (primarily from Electrobel and its subcontractors) that perhaps the change reflected enhanced sensitivity of the UT and that the flakes had been present at the manufacture of the vessels. We raised the concern that embrittlement had reduced the fracture toughness so that even a smaller flake could eventually initiate a crack at a lower stress level than would be the case for a non-embrittled steel. We also found that hydrogen flakes had the potential to grow to a dimension that, if properly orientated with respect to the principal stress axis, would have a stress intensity factor exceeding the fracture toughness of the RPV steel. This phenomenon could result in an unstable crack growth rate and failure of the vessel. Given the large size of some existing flakes (> 1-cm), in our opinion the continued operation of the reactors created "accidents waiting to happen". Nevertheless, our argument was rejected, and the plants have continued operating.[1]

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[1] The NRC, the staff of which are primarily mechanical/nuclear engineers, do not consider hydrogen embrittlement (HE) or hydrogen-induced cracking of radiation-embrittled RPVs in their repertoire of failure mechanisms even though it is considered to be the primary cause of failure of embrittled steels (e.g., of welds in carbon steels) in the oil and gas industry. This

8. During the last ten years, I have striven to introduce determinism into corrosion science to accurately predict the evolution of corrosion damage in nuclear systems. Macdonald (2023). For example, under sponsorship of ONDRAF-NIRAS of Belgium, I predicted the evolution of general corrosion and pitting corrosion to carbon steel canisters for the disposal of high-level nuclear waste (HLNW) in Boom Clay repositories over a 100,000-year disposal period, yielding realistic results. Under sponsorship of the Swedish Radiation Safety Authority (SSM), I performed similar work on copper canisters in granitic rock repositories. Prior to that, I was heavily involved in predicting corrosion damage in canisters for the now-defunct Yucca Mountain program and demonstrated that pitting corrosion might lead to the failure of the Alloy 22 corrosion resistant alloy outer layer of the canister. Using the CEFM, I and a student also calculated the CGR in Alloy 22 under Yucca Mountain environmental conditions where the CGR was so low ($< 10^{-11}$ cm/s) that it cannot be measured experimentally without the imposition of a ripple load (low R-ratio fatigue loading). Our calculations were judged to be realistic and showed that SCC is not a threat to canister integrity.

9. Since the early 1970s, when I was employed by AECL, I have worked to introduce electrochemistry into reactor coolant technology. For that effort, I was recently nominated for the Enrico Fermi Award, perhaps the premier award in nuclear science and engineering.

10. I am familiar with NRC regulations and industry guidance for pressure vessel maintenance and surveillance and the record of PG&E's surveillance program and NRC reviews.

## III. SUMMARY OF EXPERT OPINION

1. As discussed below in Section IV, the pressure vessel is a uniquely important and vulnerable component in a nuclear reactor, because it holds water on the highly radioactive reactor core, and because it has no backup if it should crack and lose water during an accident. Therefore, compliance with NRC requirements for monitoring the condition of the plant-specific pressure vessel is essential.

2. For pressure vessels, these regulatory requirements are three-fold and complementary:

   - First, through "Charpy" testing of samples taken from the reactor vessel, the licensee must demonstrate that the "reference" temperature for pressurized thermal shock ($RT_{PTS}$) is below a threshold of 270ºF for axially oriented welds and 300ºF for circumferential welds. $RT_{PTS}$ is the temperature at which fracture morphology of the pressure vessel changes from ductile to brittle as its temperature drops from the addition of cooling water during a loss of coolant accident (LOCA). Data for the

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oversight is greatly concerning when it is noted that on the solution side of the RPV is a coolant, a solution of boric acid and lithium hydroxide containing 25-35 cc(STP)/kg $H_2O$ of molecular hydrogen. The $\gamma$, $n$, and $\alpha$ radiolysis of the coolant produces a large amount of atomic hydrogen, some of which enters the RPV and further embrittles the steel.

fracture energy vs. test temperature are determined from Charpy testing of standard specimens (ASTM 185-82) that had been irradiated in capsules located between the reactor core and the inner surface of the RPV. The capsules are withdrawn at more-or-less equally spaced intervals (typically, every ten calendar years) throughout the reactor life of 32 EFPY (40 calendar years).

- Second, also through Charpy testing, the licensee must demonstrate that the pressure vessel is strong enough to withstand the transient stresses induced by thermal shock of the rapidly changing temperature caused by the addition of cooling water, *i.e.,* that the "upper shelf energy" (USE) will remain above 50 ft-lb.

- Finally, every ten years, the licensee must conduct ultrasound testing (UT) inspections of the most vulnerable part of the reactor vessel, the welds around the beltline, to examine for flaws and cracks. NRC guidance appropriately provides that the schedules for these inspections may be relaxed only upon a verifiable demonstration that safety will not be jeopardized.

3. These three types of tests and inspections are complementary in three significant respects. First, each of the measured phenomena makes a distinct and significant contribution to determining the vulnerability of a pressure vessel to cracking. Second, while the reference temperature and USE calculations are both derived from the same Charpy tests, the method of analysis for each is different; and of course, the UT inspections involve completely different methods of acquiring and analyzing data. Third, each type of test or inspection has a different level of reliability. As discussed below in Section V.A.2, my calculations show that Charpy tests are not particularly sensitive to the extent of embrittlement. Therefore, their results should not be substituted for UT inspections, nor should they be used to justify an extension of the schedule for UT inspections. The three types of data must be considered in unison because they convey important, complementary information on the safety of the RPV.

4. As discussed below in Section IV.B., adequate monitoring of the condition of the pressure vessel is particularly important in the case of Diablo Canyon Unit 1 because the composition of the welds in the pressure vessel was found to be defective at the time it was installed by having excessive copper and nickel. Not surprisingly, in 2006, the NRC identified the Unit 1 pressure vessel among the most embrittled, with only 14 of 72 PTS reference temperatures as high as or higher than Diablo Canyon Unit 1. U.S. NRC 2007. And today, half of those 14 reactors are closed.

5. As discussed below in Section V.A, in 2002, PG&E withdrew and tested "coupons" or weld samples from the Unit 1 pressure vessel and conducted Charpy tests for PTS reference temperature and USE. PG&E (2003). In 2003, PG&E reported that it had calculated a limiting $RT_{PTS}$ value of 250°F for the limiting weld 3-442C. *Id.* Thus, PG&E predicted that in 2021 (the expected retirement date for Unit 1 at that time), the reference temperature for Unit 1 would be slightly more than 10° below the screening limit of 270 °F. Taking into consideration a reasonable margin of error of about ± 10 °F (as estimated

by inspection of the Charpy curves), PG&E's test showed that Unit 1 would be approaching the limit at the end of its operating life.

6. Nevertheless, PG&E discounted the data as "not credible." *Id.* But PG&E may have found that the data *were* credible if it had applied standard scientific and NRC guidance for its evaluation. U.S. NRC (1998). PG&E's failure to apply this well-established and reasonable guidance is both inexplicable and gravely concerning, given that the $RT_{PTS}$ data indicated a serious degree of embrittlement. The NRC Staff's approval of PG&E's disregard of the data is also puzzling, given that PG&E had ignored the agency's own guidance.

7. Instead of crediting the data it had gathered from Unit 1, PG&E substituted generic data and data from other reactors. As discussed in Section V.C, PG&E's reliance on substitute data from other reactors was also unreasonable, especially for a period that stretched across decades. Regardless of their initial similarities, all nuclear reactors soon because individualized by unique operating conditions and histories. At the very least, PG&E should have applied a larger error band to any reference temperature calculations that were based on generic data or data from so-called "sister" reactors. Instead, PG&E is doubling down on its reliance on data from sister reactors.[2]

8. As also discussed in Sections V.C and V.D, the results of the 2003 evaluation of the Charpy tests should have motivated PG&E to speed up its schedules for obtaining more data in order to get a better sense of the pressure vessel's condition. At the very least, PG&E should have adhered to its approved schedule for the next capsule extraction and Charpy test in approximately 2009. And PG&E should have ensured that the most recent (2005) UT inspection -- which identified "one indication . . . in the beltline region" (PG&E (2014)) -- would be followed on schedule with another beltline inspection in 2015. Yet, PG&E repeatedly sought and obtained extensions of time for these measures: the next Charpy test has now been rescheduled from 2009 to 2023 or 2025, depending on whether PG&E is able to withdraw the capsule in 2023 (U.S. NRC (2023)); and the next UT inspection is scheduled for 2025 (U.S. NRC (2015)).

9. In both cases, the extensions leave an unacceptable gap of 20 years between the tests or inspections. In my professional opinion, two decades is an unacceptable amount of time, for two reasons. First, there was no reason for PG&E to rely on questionable generic data or data from so-called "sister" reactors for more than a short time after the 2003

[2] In 2011, eight years after informing the NRC that the data from Capsules S, Y, and V were "not credible" (PG&E (2003)), PG&E relied on data from another reactor to assert that Unit 1 can be safely operated to the end of a 20-year renewal period. PG&E (2011). *See* Table 4.2-4, showing that the limiting weld 3-442C does not meet or approach the regulatory limit of 270 °F until 54 EFPY, the equivalent of 60 years of operation. The reference document for this prediction is WCAP-17315-NP (Westinghouse (2011)), which relies in part on data from the Palisades reactor to project $RT_{PTS}$ values for the end of the Unit 1 license term.

evaluation. PG&E could have and should have obtained more plant-specific data by now. Second, the condition of the pressure vessel may change significantly over a single decade. *See* Section V.C below.

10. In addition, the fact that PG&E's 2005 UT inspection of the pressure vessel were "essentially identical" to an inspection done 10 years earlier and yielded only one "indication" of cracking (PG&E (2014)) should have prompted PG&E to evaluate whether the UT inspection was faulty and needed to be repeated. It is reasonable to expect many more indications of voids and cracks, and that they would increase over time. *See* Section V.B below.

11. Under these circumstances, it is my expert opinion that the NRC currently lacks an adequate basis to conclude that Diablo Canyon Unit 1 can be operated safely. And the NRC Staff's recent decision to allow PG&E to postpone the next Charpy test for Unit 1 until 2025 (U.S. NRC (2023)) is unjustified. In order to protect the public from the unacceptable risk of a core meltdown accident caused by pressure vessel cracking and fracture during a loss of coolant accident (LOCA), the NRC should (a) order the immediate closure of the reactor by accelerating a maintenance shutdown now scheduled for October, (b) require that the reactor must remain closed pending completion of the next scheduled Charpy tests, (c) ensure that any coupons or capsules that have been withdrawn but were not tested are subject to Charpy tests, (d) account for the data provided by the wedge opening loading (WOL) specimens and the tensile specimens that were scheduled to be contained in the capsules, and (e) ensure that any remedial steps taken by PG&E to address the condition of the Unit 1 reactor pressure vessel are subjected to rigorous review by the NRC Staff, the Advisory Committee on Reactor Safeguards (ACRS), and the general public. *S*ee Section VI.A.

12. Finally, in the spirit of 10 C.F.R. § 50.51(c)(3), I will offer "information" that I believe will "improve the accuracy of the $RT_{PTS}$ value significantly." In my professional opinion, the newly developed method of nano-indentation promises to be capable of far more extensive results from a single specimen than the conventional Charpy Impact Test methods prescribed by NRC regulations. *See* Section V.E. The more extensive data will permit rigorous statistical analysis, something that is not possible with Charpy. Importantly, this method has already been applied by Professor Peter Hosemann of the Department of Nuclear Engineering, University of California, Berkeley and found to be sensitive to the change in physical properties of PWR RPV steels brought about by radiation embrittlement. Accordingly, in my professional opinion, the technique requires further application in the field to define and quantify its advantages.

## IV. BACKGROUND ON PRESSURE VESSEL AND REGULATORY REQUIREMENTS

### A. Importance of pressure vessel integrity in a pressurized water reactor

1. At Diablo Canyon and other pressurized water reactors, the reactor fuel core is contained within the pressure vessel, a massive steel structure approximately 30 feet tall and ten feet in diameter, with a wall thickness of approximately 10 inches. A cut-away view of the RPV of a typical Westinghouse PWR is displayed in Figure1. The pressure vessel is normally completely filled with water to keep the core covered and is kept under pressure to prevent the cooling water from boiling at the high temperatures under which the reactor is operated. During normal operation, the pressure vessel and its contents are heated to approximately 550 $^{o}$F by the nuclear fissioning of $^{235}U_{92}$ and toward the end of the core life by fissioning of various isotopes of plutonium such as $^{239}Pu_{94}$ and $^{241}Pu_{94}$. The region of principal concern in the petition is the beltline region, which is the region of the RPV that is immediately opposite to the core and is depicted in Figure 1 as the “150" active core length”. It is this region that experiences the greatest fast neutron flux (E > 1 MeV) and hence fluence and which becomes the most radiation embrittled. Of principal concern is the embrittlement of “limiting” materials, such as welds and heat-affected zones (HAZ) that are envisioned to be the weakest components when embrittled and hence are those that will likely fail first.

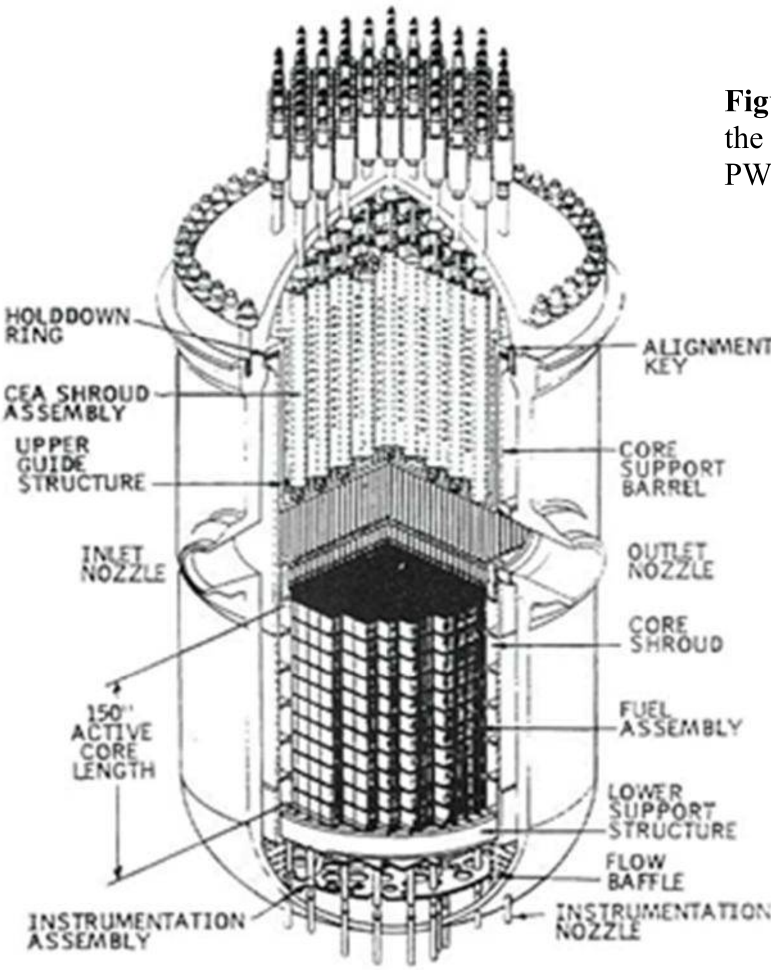


**Figure 1**: Cut-away schematic of the core of a typical Westinghouse PWR.

2. The reactor pressure vessel, together with the reactor coolant piping connected to it, form the reactor coolant pressure boundary which holds the reactor cooling water. Reactor cooling water must be always kept on the core to prevent the core from overheating and possible melting down even during shutdown because of the decay heat from the spontaneous decay of unstable isotopes ("fission products"). The melting of the core, should it occur, could release a large quantity of radioactivity into the reactor's containment. Should the containment building also fail, this would probably result in the release of significant levels of radiation outside the plant, potential causing deaths, illness, environmental damage, and economic injuries. The Chernobyl accident is illustrative of the scale of potential health and environmental effects and costs, although that reactor did not have containment of the type in Western reactors.
3. Unlike most other reactor safety components, the pressure vessel has no redundant and independent backup system that can be called upon if it should crack or fracture and lose essential cooling water. In the event of water loss from the pressure vessel and uncovering of the reactor core, a nuclear meltdown may occur.

4. Pressurized thermal shock ("PTS") is a reactor pressure vessel condition that can occur during an accident when high pressure combines with sudden decrease in temperature. If core cooling water is lost during a break in the pressure boundary, a loss of coolant accident ("LOCA") may occur. In response to such an event, the emergency core cooling system ("ECCS") responds by pumping cold water into the vessel. The rapid decrease in the temperature at the vessel wall compared with that further into the wall generates thermal stresses, which together with the stresses induced by the operating pressure of ca. 2250 psi, may act upon a suitably oriented flaw such that the stress intensity factor ($K_I$) exceeds the fracture toughness, $K_{Ic}$. This may result in the rapid propagation of a through wall crack in the embrittled vessel and in the failure of the vessel.

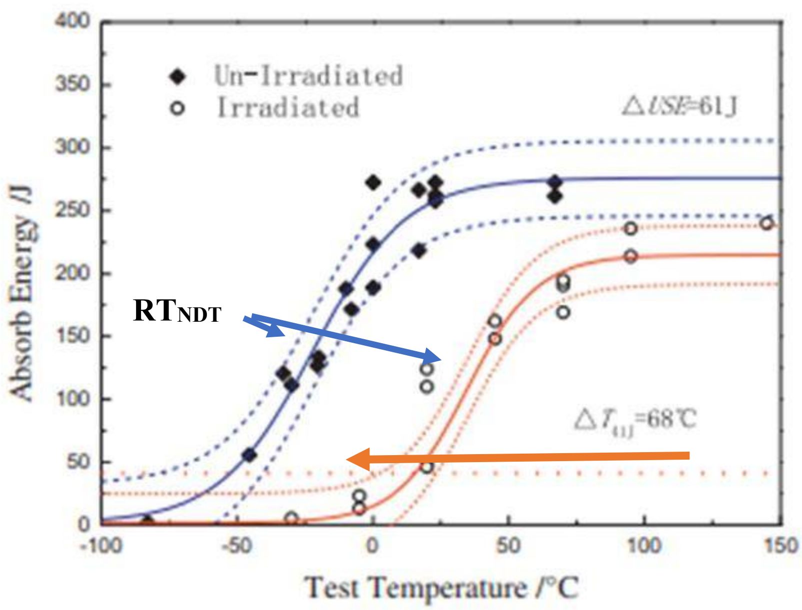


**Figure 2:** Effect of neutron irradiation on the Charpy impact test results for a fluence of $10^{20}$ n/cm$^2$ (E > 1 MeV) for A508-3 RPV steel. After Lin, et.al. Note that irradiation cause the value of $RT_{NDT}$ to shift by about 68 ºC (154 ºF) and the USE to be reduced by 61 J.

5. If the ductile to brittle transition temperature of the embrittled steel, as characterized by the nil ductility transition temperature or "$RT_{NDT}$", is sufficiently high compared with the unirradiated, non-embrittled steel, the vessel may fail by brittle fracture because of the sudden reduction in the fracture toughness as the temperature moves below $RT_{NDT}$. This is indicated in Figure 2 where $RT_{NDT}$ is depicted by the inflection points (indicated by the blue arrows) in the hyperbolic tangent dependence of the fracture ("Absorb") energy on temperature for both the unirradiated steel and the irradiated steel. These values are quite different from the arbitrarily defined values for $RT_{NDT}$ at 41 J (30 ft-lb) recommended by the ASME Pressure Vessel Code and adopted uncritically by the NRC. Both the $RT_{NDT}$ and the USE are used to judge the susceptibility of the RPV to PTS but the NRC defines $RT_{NDT}$ as that temperature corresponding to a fracture energy of 30 ft-lb (41 J), as indicated by the red-dotted line in Figure 2. These values are significantly different from those indicated by the inflection points.

6. Thus, while it is readily understood as to why $RT_{NDT}$ was defined this way by ASME, ASTM, and the NRC in that it yielded a definite metric corresponding to the intersection of two lines, the more fundamental $RT_{NDT}$ corresponding to the inflection point is also readily determined from the hyperbolic tangent function that is used to fit to the Charpy fracture energy (FE) vs. test temperature data with minimal mathematical manipulation.

It is generally good scientific practice to choose the more fundamentally defined metric if they can all be determined with comparable precision.

**B. Importance of reactor-specific surveillance programs to assess and maintain safe operation**

1. NRC standards for the condition of reactor vessels are found in 10 C.F.R. Part 50 Appendix G and 10 C.F.R. § 50.61(b). These standards establish two general sets of requirements: for fracture toughness as demonstrated by "Charpy" upper shelf energy (USE) and the shift in the adjusted nil ductile to brittle transition ($ART_{NDT}$) temperature of the embrittled (neutron irradiated) steel microstructure compared with the un-embrittled (unirradiated) microstructure and the fracture resistance to pressurized thermal shock (PTS). Appendix G sets a limit of 50-ft-lbs for the USE in a pressure vessel. Section 50.61(b)(2) establishes a screening criterion of 270 °F for ($RT_{PTS}$) for axial welds and 300 °F for circumferential welds, where $RT_{PTS}$ is the reference temperature at the end of a reactor's operating life (EOL). If a reactor vessel is predicted to exceed the screening criterion, 10 C.F.R. § 50.61(b)(3) requires that flux reduction measured must be employed. Both sets of requirements must be satisfied.

2. The purpose of a surveillance program is to expose *in situ* samples of limiting materials [*e.g.*, plates, welds, heat-affected zones (HAZ), and standard reference materials (SRM)] in the beltline region in the reactor pressure vessel (RPV) under identical conditions to those experienced by the RPV itself. Because the neutron flux varies with radial distance ($r$) from the core axis roughly as $\frac{1}{(r-r_0)^2}, r > r_o$, where $r_o$ is the radius of the core, the placement of the capsule at a specific radial distance enables the end of life (EOL) fluence to be simulated for an exposure time of less than the design life of the reactor (typically 32 EFPYs or 40 calendar years). This "lead factor", which is the ratio of the neutron flux at the capsule and that at the vessel inner surface, is important in the design of an effective surveillance program because it enables the fluence future to be foretold within certain constraints, provided various factors (e.g., operating conditions) remain the same into the future as they were in the immediate past.

3. Equally important is the capsule withdrawal schedule, which typically specifies that one capsule must be withdrawn every 10 years for a four-capsule surveillance program. This is so because a regular withdrawal schedule allows the evolution of radiation embrittlement to be followed and hence to provide consistency in the EOL radiation damage estimates (from all capsules depending on the lead factors). As discussed below in Section V.D, PG&E has postponed this surveillance to such an extent that it completely skipped the withdrawal and testing of Capsule B as originally scheduled for 2007, and now proposes to withdraw the capsule in 2023 or 2025. As a result, PG&E lacks fundamentally important data regarding the condition of the Unit 1 pressure vessel.

4. The regulations also require tensile and fracture mechanics (WOL, wedge opening loading) to be exposed in each capsule along with the Charpy specimens. The tensile specimens are used to measure *ex situ* the yield stress (YS) and the ultimate tensile stress/strain, both of which are indicative of the state of embrittlement, while the WOL

specimen yields a measure of the true fracture toughness, $K_{IC}$ from the crack length upon removal of the capsule and the compliance of the specimen. This is important, because the "fracture toughness" measured by the Charpy tests is not the same as $K_{IC}$ that is used to determine if a suitably oriented flaw (with respect to the stress axis) in the vessel will grow unstably and possibly initiate a LOCA. Although PG&E appears to have performed the tensile tests, I cannot find any analysis of the WOL specimens. In my opinion, this is an unacceptable omission from the surveillance program for Diablo Canyon Unit 1.

5. Because the strength and fracture resistance of a reactor vessel change over time as the vessel is exposed to radiation and changing temperatures, NRC regulations in Appendix H and 10 C.F.R. § 50.61 Subsection I(2) requires licensees to have a "material surveillance program" with a schedule for removal and testing of surveillance capsules that conforms to industry standard ASTM E 185. NRC regulation 10 C.F.R. § 50.61I(2)(i) further requires all licensees to integrate the results of their plant-specific surveillance programs into the estimate of reference temperature ($RT_{NDT}$) for the reactor vessel material.

6. In my professional opinion, the reactor-specific surveillance data required by the NRC's regulations is key to ensuring that a reactor operates in compliance with NRC safety limits. As contemplated by the regulations, generic data and data from so-called "sister" reactors should not be relied on unless and until the options for obtaining reactor-specific data have been exhausted. In any complex industrial system (nuclear reactor, chemical plant, aircraft, etc.) the judgment that the system is safe to operate must be based on plant-specific data in the same way that a health professional judges the viability of a person to operate successfully in life. That decision cannot be made upon the basis of the health of a sibling, even if that sibling was an identical twin. So it is for a nuclear reactor. It is for that reason that the NRC mandates a plant-specific surveillance program.

7. In the case of Diablo Canyon Unit 1, obtaining surveillance data specific to that pressure vessel is particularly important because the reactor weld chemistry was deemed defective when the pressure vessel was installed, because of excessive copper and nickel content that render it more vulnerable to embrittlement. The excessive copper (approx. 0.2 %) arises from the corrosion protective copper coating on the weld wire employed and the excessive nickel content of approx. 1 % originates from the composition of the weld wire itself. The deleterious impact of both copper and nickel in the radiation embrittlement of welds in ferritic steels has been established by numerous laboratory and field studies. After Diablo Canyon Unit 1 was completed, the error was realized, and Unit 2 did not contain excessive Cu and Ni in the welds.
8. The number of capsules needed for a reactor vessel surveillance program is established with reference to the ASTM standard. In the case of Diablo Canyon, to satisfy the requirements of ASTM E 185-73, PG&E started with a five-capsule program based on the estimated shift in the adjusted nil ductility reference temperature above 200° F. PG&E

(1992).[3] In 2006, for unexplained reasons, the NRC re-characterized the surveillance program as a "four capsule program." U.S. NRC (2006). Whether characterized as a 4 or 5-capsule program, each program was designed for the current license term and included a schedule for removal of Capsule B about midway through the current license term (EFPY 19.2 or EFPY 20.7, RFO 14 and RFO 15 in the period 2007-09).[4]

9. The data collected by a reactor vessel surveillance program is useful both for assessing the current integrity of the reactor vessel and for projecting its condition in the future. Thus, for example, PG&E's surveillance program, as approved by the NRC in a 2006 license amendment for recapture of the low-power testing period, required removal of Capsule B at 20.7 EFPY. U.S. NRC (2006). This timing would allow PG&E to obtain data about the current condition of the vessel. It would allow provide information about the fluence of the vessel at the end of the license renewal term, or "approximately twice the projected limiting inside RV fluence for DCPP-1 [Diablo Canyon Unit 1] at the EOL (i.e., approximately $2 * 1.43 \times 10^{19}$ n/cm$^2$ (E > 1.0 MeV]." U.S. NRC (2006).

10. And while the number of capsules inserted into a pressure vessel cannot be changed (other than by adding more of them for future assessment), the schedule can be adjusted to accommodate the demands of the surveillance program. For instance, if a set of surveillance data from a particular capsule turns out not to be credible, the licensee may remove other capsules if the altered schedule change is consistent with the industry standard.

11. In my professional opinion, the most important reason for changing a surveillance schedule, other than adjusting to new information regarding vessel fluence, would be to provide additional data where available data had proven to be insufficient. It would not be reasonable, however, to change a capsule removal schedule for any other purpose if the change would leave the surveillance program with a gap of ten or more years.

12. The measurement of $RT_{NDT}$ and USE is only part of the story in assessing whether an embrittled RPV is in danger of rupture particularly under "pressurized thermal shock" (PTS) conditions resulting from the injection of cold water to compensate for loss of coolant from the rupture of the pressure boundary elsewhere. While $ART_{NDT}$ and USE are appropriate monitors of the state of embrittlement, the probability of crack nucleation is a question that must be addressed by probabilistic fracture mechanics that requires the assessment of the population, size, and orientation of flaws close to the cladding/steel interface. Therefore, UT is used to evaluate flaw volume density (#/cm$^3$), flaw size, and flaw orientation so as to determine if any flaw is characterized by a stress intensity factor ($K_I$) that exceeds $K_{IC}$ for the embrittled steel. The American Society of Mechanical

[3] PG&E inserted Capsule B into the Unit 1 pressure vessel and the NRC approved a schedule for withdrawing and testing it when the reactor achieved 19.2 EFPY. *Id. See also* Table 4. In 2006, in approving a license amendment for "recapture" of the three years of low-power testing of Unit 1, the NRC approved a change in the withdrawal schedule to 20.7 EFPY. U.S. NRC 2006.

[4] This schedule can be derived from PG&E (1992), Enclosure at 3-4, Table 4; U.S. NRC (2006), Safety Evaluation at 5; and PG&E (2023), Enclosure 2.

Engineers (ASME) code that is incorporated by NRC regulation 10 C.F.R. § 50.55a requires that an UT inspection must be performed every ten years.

## V. DISCUSSION

### A. PG&E failed to consider credible data showing that Unit 1 is now approaching PTS temperature screening criteria.

#### A.1. Unit 1 $RT_{PTS}$ surveillance data obtained in 2003, erroneously characterized by PG&E as "not credible", show that Unit 1 could approach NRC's threshold for remedial action as early as 2024.

1. In my professional opinion, PG&E has incorrectly discredited the data it obtained from Unit 1 in Capsules S, Y and V for the purpose of calculating $RT_{PTS}$ values. PG&E should have been concerned that these data showed that Unit 1 could approach the PTS temperature screening limit by the end of the reactor's initial license term and should have investigated the reasons for anomalies in the data. Yet, in disregard of common scientific practice methods and NRC guidance, PG&E claimed the data were "not credible." PG&E (2003).

2. In 2003, PG&E tested data from recently withdrawn Capsule V. According to PG&E Letter DCL-03-052, at Unit 1's EOL date of 32 EFPY (which at that time was 2021), the limiting $RT_{PTS}$ value calculated by PG&E's contractor, Westinghouse, for the limiting weld 3-442C was 250.9 °F. PG&E (2023), Westinghouse (2003). This calculation should have concerned PG&E because it was approaching the PTS screening criterion of 270 °F for plates, forgings and axial weld materials and within a reasonable margin of error of about ± 10 °F (as estimated by inspection of the Charpy curves), resulting in an overlap of uncertainties in the screening criterion (270 °F) and the Westinghouse estimate (250.9 °F) for weld 3-442C. In addition, as further explained in Section V.A1, the fact that the measured $RT_{NDT}$ for Capsule V (201.07 °F) was lower than the value for Capsule Y that had been removed ten years earlier at 1R5 (232.59 °F) (Westinghouse (2003), Table D-2) indicated a reasonable possibility that one of those tests was erroneous, because it unlikely that continued exposure to radiation would "heal" the metal. If the value of Capsule V was erroneous and the value of Capsule Y was correct, then the limiting $RT_{PTS}$ value Unit likely was even closer to the PTS screening criterion than calculated by PG&E.

3. Despite these concerning results, PG&E discredited *all* of the data it had obtained from Unit 1 in Capsules S, Y and V, based on a determination that the "best fit curve" between the Capsule V data and data from earlier-withdrawn Capsules S and Y contained scatter values for two data points that exceeded the criteria in Regulatory Guide (RG) 1.99, Rev. 2, Criterion 3 (U.S. NRC 1988)). According to RG 1.99, the scatter values for data "normally should be less than 28°F for welds and 17°F for base metal" PG&E (2003), Westinghouse (2003). This is equivalent to ± 1 Sigma. Therefore, PG&E declared that all

the data from Capsules S, Y and V were "not credible" for the purpose of calculating limiting $RT_{PTS}$ values. PG&E (2003).[5]

4. PG&E's methodology for assessing the credibility of the data is inconsistent with NRC's own guidance for performing credibility assessments. U.S. NRC (1998). At page 11, the guidance states as follows:

    A. If there exists an identified and recorded deficiency in a datapoint - a duplicate or untraceable record, a record which identifies an atypical condition or sample location, or

    B. If a datapoint is identified as a statistical outlier <u>and</u> a physical basis exists for believing the datapoint to be atypical -

    - A*ll data not excluded in (A.) should be used as the dataset*
    - *A priori exclusion of some data based on "inconsistency" with expected norms should not be used before analysis for statistical outliers is conducted".*

    (Italics mine). In violation of the NRC guidance, PG&E excluded not just inconsistent data but all of the data "a priori", without conducting "an analysis for statistical outliers."

5. In addition, the rejection of all the data because one datum did not fall within the bounds by a narrow margin does not conform with accepted scientific and engineering practice. In analyzing scattered data, it is common to find points that lie outside of a preconceived scatter band. If the scatter band has been established via the analysis of a significant population of historical data for identical samples from the same system (reactor) and it is established that the data follow a normal distribution, it is possible to define the width of the scatter band in terms of the standard deviation with the next sample having a 68 % probability of falling within the mean ± one standard deviation or a 96 % probability for falling within a ± two standard deviations ,and so forth. However, there is a finite probability that future values of $RT_{NDT}$ and USE will lie outside of these limits (32 % and 4 %, respectively). That is the inherent nature of experimental data.[6] For a system as critical as a beltline weld, for example, a margin of error of the mean ± one standard

[5] As discussed in Section V.A.2 below, separately, PG&E found that the USE data from Capsule V do not indicate excessive embrittlement. USE remains above 50 ft-lbs to the reactor's end of life (EOL) or 32 EFPY, as required by 10 C.F.R. Part 50, Appendix G. My own analysis of the USE data, however, demonstrates that Unit 1 may reach an unacceptable level of embrittlement at 43.8 EFPY or earlier.

[6] If the data from a single reactor are insufficient, it is possible to examine data from another reactor to evaluate whether the distribution is normal. But if the data are not from the same system, a systematic error will likely be introduced, the magnitude of which could vary widely from one data set to another from different reactors. If sufficient data were available from two "sister" reactors it is unlikely that they follow the same standard normal distribution since each reactor is unique because of unique operating conditions and histories. Under these circumstances, defining the uncertainty in terms of a standard deviation becomes problematic.

deviation is too tight and in my professional judgement the probability and consequences of failure are too high.

6. Even if the use of the "standard deviation" is correct and I had established the correlation with three data points (as is the case for Diablo Canyon Unit 1) and found the distribution to be normal, and I added one more datum that was from the same population, there is a 0.32x3 = 0.96 (= 1) probability that the datum will fall outside the mean ± one standard deviation for no obvious reason. Thus, the observation that one point in the Diablo Canyon Unit 1 correlation fell outside the error band is statistically insignificant (bordering on the nonsensical) and calls into serious question the invalidation of the Capsule S, Y, and V data by PG&E.

7. PG&E also departed from standard scientific practice in failing to plot the data it relied on, relying instead on a narrative. Nowhere can I find the actual graphical presentation of the correlation of $\Delta R_{TNDT}$ with fluence so that I can judge for myself the validity of PG&E's non-credibility claim. Given the safety significance of PG&E's rejection of the Unit 1 surveillance data, its failure to fully disclose the quantitative data on which it relies constitutes a serious violation of normal scientific and engineering practice. Furthermore, I can find no attempt by PG&E to establish the assumption that the data follow a standard normal distribution, which must support any analysis and specification of a standard deviation. Many physical phenomena follow a lognormal distribution that could significantly change the conclusions arrived at by PG&E.[7]

8. Accordingly, for any point that does lie outside of the limits, especially far outside the limits, the first course of action should be to ascertain whether there is a valid physicochemical reason for the anomalous result. If a valid reason can be found, such as an experimental error, then that datum is treated as an "outlier" and can be excluded from the analysis of the remaining data. Importantly, where outliers exist, they do not provide a valid reason for discrediting the data that do meet the criteria for credibility.

9. It is also unreasonable to reject otherwise plausible data out of hand when the entire available data set is so small. The only reasonable solution to the problem that the scatter values exceeded the NRC's criteria was to gather more data and compare it to the existing data. Had PG&E collected and tested more data, then the appropriate placement of the "best fit" curve in the correlation would have been more reliably established and it would have been more difficult to throw the data out. Gathering the data from Capsule B and testing those data along with Capsule C is an essential step toward improving the size of the data pool and thereby the quality of the analysis.

10. Had PG&E appropriately credited its own data, it would have had to take remedial measures to ensure the integrity of the pressure vessel, as required by Section 10 C.F.R. 50.61a. Instead, as discussed below in Sections V.C and V.D, PG&E relied for an

[7] Underlying this whole issue is the paucity of data from the Charpy test. *See* Section V.A.2 above.

extended period on data from other reactors to justify continued operation and postponed any further testing or inspection of the reactor vessel.

### A.2 My separate and independent analysis of 2003 Charpy Impact Test data that were deemed credible by PG&E shows that the Unit 1 pressure vessel could reach an unacceptable level of embrittlement at 43.8 ±10 EFPY.

1. The paucity of plant-specific data from 14.27 EFPY (when the Capsule S was withdrawn and tested (PG&E (2023)), to the EOL EFPY of 32 is a problem of the utmost seriousness, particularly when one realizes that data from one or both of Capsules Y and V are suspect for reasons speculated upon elsewhere in this Declaration. Leaving aside for the moment PG&E's unjustified attempt to exclude all plant-specific data, the paucity of data could stretch from 5.87 EFPY or even from 1.25 EFPY to the EOL at 32 EFPY. This is an intolerable situation that essentially means that neither PG&E nor the NRC have a defendable estimate of the time that it will take for the weld to achieve the critical condition of USE = 50 ft-lb. This deficiency is addressed below in my reanalysis of PG&E's Charpy data using completely new methodology for analyzing those data. Using that methodology, I calculate that the critical condition will be reached at 43.8 EFPY with an estimated uncertainty of ± 10 EFPY.

2. Given PG&E's failure in 2003 to present any Unit 1-specific evidence regarding the rate of embrittlement over time, I developed a model that would use the Charpy Impact Test (CIT) data deemed credible by PG&E to determine the Extent of Embrittlement (EoE) over the life of Diablo Canyon Unit 1.

3. USE measurements or CIT data for nuclear reactor pressure vessels provide a direct experimental quantification of the degree of embrittlement over time. For the 2003 USE evaluation, PG&E and Westinghouse determined that the CIT data were credible. PG&E (2003), Westinghouse (2003). For my own review, I have consulted the CIT data for three reasons: first, because PG&E deemed them credible in contrast to the $RT_{NDT}$ data; second, because they are unencumbered with corrections, such as the chemistry factor, margin, and the fluence factor that are required to correct $RT_{NDT}$ to a specific material in a specific plant; and third, because the USE is more directly related to the degree of embrittlement than is the adjusted $RT_{NDT}$.

4. By mathematically deriving an expression for the EoE from coefficients ($A$, $B$, $C$, and $T_0$) obtained for the symmetric hyperbolic tangent function ($FE = A + B.\tanh[(T - T_0)/C)$ that is used by PG&E to optimize on the fracture energy (FE) vs test temperature CIT data, I have calculated $EoE = \left[1 + \frac{e^x - e^{-x}}{e^x + e^{-x}}\right]/2$ and $x = (RT_{NDT,30} - T_0)/C$ where $RT_{NDT,30}$ is the transition temperature that is defined for a fracture energy of 30 ft-lb (41 J). The EoE are plotted as a function of fluence in Figure 3. The expression for EoE tacitly assumes that the EoE also follows the hyperbolic tangent function given above where the point of inflection $RT_{NDT,PoI} = T_0$. By my reasoning, $RT_{NDT,PoI}$ is a much better definition of the nil-ductility transition temperature than is the arbitrarily defined $RT_{NDT,30}$, as noted above. Note that at the point of inflection (PoI), the EoE = 0.5 indicating that the fracture is 50 % brittle and 50 % ductile. As we will see below, this

ratio of brittle vs. ductile fracture is close to the ratio (= 1.1) at the critical condition defined by the NRC of 50 ft-lb.

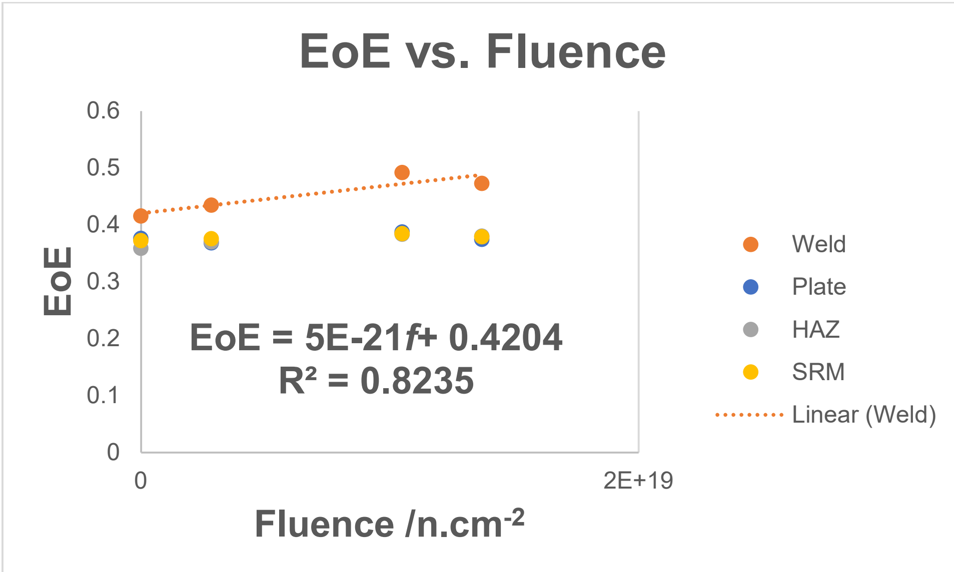


**Figure 3:** Values for EoE derived from the CIT data of PG&E for metal specimens from Capsules S, Y, and V that were exposed in Diablo Canyon Unit 1.

5. As we see from Figure 3, the EoE for the weld metal is significantly greater than that of the plate, HAZ, and SRM samples showing that the weld is the most susceptible of the samples contained in Capsules S, Y, and V that were exposed in Diablo Canyon Unit 1.

6. This difference is addressed as follows. When choosing a technique to monitor a selected phenomenon in a well-designed experiment, it is essential that the dependent variable (the measure of the phenomenon, e.g., the EoE) have a high sensitivity to the principal independent variable, in this case, the fluence. Figure 3 reveals that the CIT has different levels of sensitivity for different materials. For the plate, HAZ, and SRM, the CIT is not very sensitive to the extent of embrittlement, with EoE changing by no more than 3 % over the first 14.27 EFPY operating life of the reactor. In contrast, for the weld metal, the EoE changes by about 8 %. Of course, the lack of sensitivity may also reflect that the plate, HAZ, and SRM do not embrittle rapidly, at least up to a fluence of $1.37 \times 10^{19}$ n/cm$^2$. Fortunately, the CIT does effectively detect the embrittlement of the limiting weld material.[8]

[8] In my opinion, the CIT should be replaced, or at least complemented by another technique that does meet that standard of high sensitivity of the dependent variable on the principal independent variable. Such a technique might be nano indentation that is recognized by the NRC (U.S. NRC (1988) and currently being further developed by Prof. Peter Hosemann in the Department of Nuclear Engineering at the University of California at Berkeley (see below). While indentation is recommended by the NRC as an optional technique, in my opinion it should be made mandatory in reactor surveillance programs.

7. As demonstrated by my methodology, the EoE for the plate, HAZ, and SRM changes by no more than 3 % over the entire 14.27 EFPY at the withdrawal of Capsule V from the reactor while that for the weld metal changes by about 8 %; and (b) The final issue of the time that it will take to achieve the critical condition of the USE being reduced to 50 ft-lb has not so much to do with the CIT, itself, as it has to do with PG&E's analysis of the data obtained using the CIT.

8. It is also important to note that my methodology differs from the traditional approach of assessing USE changes over time. I have observed that most, if not all engineers and scientists skilled in the science of radiation embrittlement accept the view that whatever metric is adopted for monitoring the progression of radiation embrittlement ($RT_{NDT,30}$, $RT_{NDT,PoI}$, USE) the metric should change monotonically with increasing fluence and approach a plateau asymptotically at very high fluence. However, by all metrics examined by me, the extent of embrittlement as determined from PG&E's Charpy data passes through a maximum ($RT_{NDT,30}$, $RT_{NDT,PoI}$) or a minimum (USE) with increasing fluence, which is at odds with theoretical expectation. The rationale for my expectation of monotonic change is that the metal displacement reaction can be written as $n + m \rightleftarrows m_i + v_m$ where $n$ is the concentration of high energy neutron in 1 cm$^3$ of the metal in their transit from the entrance to the exit face of the metal cube and $m$, $m_i$, and $v_m$ are the concentrations of metal atoms, metal interstitials, and metal vacancies, respectively in the same volume. The rate of formation of displaced atoms (i.e., interstitials) can be written from chemical rate theory as: $\frac{d[m_i/m_i^0]}{dt} = k_1 f(1 - e^{-\alpha})\left[\frac{m}{m^0}\right] - k_{-1}[\frac{m_i}{m_i^0}]$ where $[m_i]$ is the concentration of displaced metal atoms (#/cm$^3$), $f$ is the fluence at the 1 cm$^2$ input face of the metal cube, and $\alpha$ is the neutron absorption coefficient in the metal. Note that the thickness of the cube of metal is 1 cm. At steady state and at limitingly high fluence $\frac{d[m_i/m_i^0]}{dt} = 0$ and we obtain $m_i = \left(\frac{k_1}{k_{-1}}\right) f(1 - e^{-\alpha})m$. This corresponds to the steady state initiation of damage as measured by the concentration of displaced metal atoms alone.[9]

9. Using the assumptions and methods set forth above, I now proceed with calculating when the beltline weld material will become unacceptably embrittled as reflected by the USE dropping below 50 ft-lb (41 Joules (J)). Thus, a plot of USE vs. EoE for all materials in Capsules S, Y, and V is displayed in Figure 4.[10] All the data are found to follow a single

[9] This simple model is incomplete in that it does not consider cascading, in which the displaced atom moves through the lattice and induces further displacements. But the model provides a reasonable physical account of the initial events in the embrittlement phenomenon. In addition, the equation is first order in fluence and cannot predict an extremum (maximum or minimum). That would require at least a second order dependence on fluence, *i.e.,* of the form $m_i = Af^2 + Bf + C$, where A, B, and C are constants.

[10] I note here that the measured USE data passes through a minimum, indicating that, somehow, the damage heals with increasing fluence from Capsules Y to V. This seems unlikely if not

locus that is represented by the equation $USE = 9.4378EoE^{2.59}$ with the plot being characterized by $R^2 = 0.9976$, indicating a high "goodness of fit". Substitution of USE = 50 ft-lb yields the critical extent of embrittlement ($EoE_{crit}$) of 0.525; that is, the fracture is predicted to comprises 52.5 % of brittle fracture (47.5 % ductile fracture) when the USE is reduced to the NRC-imposed lower limit of 50 ft-lb (41 J). This critical condition is shown as the orange data point in Figure 4. From the correlation shown in Figure 3, the critical EoE will be reached at a fluence of 2.09e19 n/cn$^2$, E > 1 MeV. Note that the ratio of brittle vs. ductile facets on the fracture surface (ratio = 1.1) is close to that defined by $RT_{NDT,PoI}$ (ratio = 1) thereby supporting my conclusion that $RT_{NDT,PoI}$ is a more fundamentally-based and hence superior metric for defining the state of embrittlement than is $RT_{NDT,30}$.

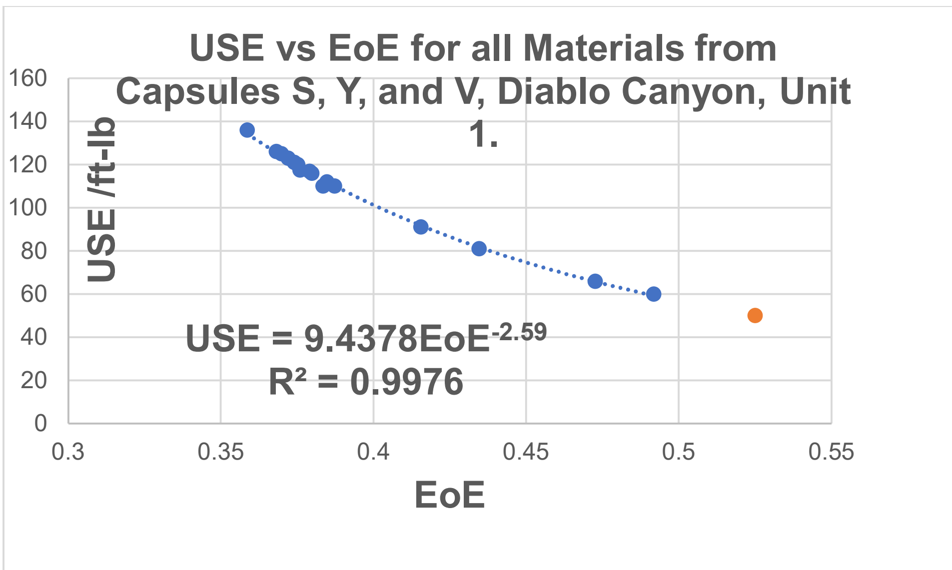


**Figure 4:** Plot of USE vs. EoE for all materials from Capsules S, Y, and V, Diablo-Canyon, Unit 1 NPP.

10. In Figure 5, I plot the fluence vs the EFPYs when Capsules S, Y, and V were withdrawn from the reactor. The data, although of significant paucity, are adequately represented by the equation given in the figure as shown by the high "goodness of fit" ($R^2 = 0.9939$). Extrapolation of the data to the critical fluence of 2.09e19 yield the time at which the USE of the weld (24702) in the beltline equals the 50-ft-lb limit. That time is calculated as 43.8 EFPYs and is represented by the last datum on the right side of Figure 5. Inclusion of this point in the fitting yields the same equation but with $R^2 = 0.9911$. Thus, the weld is predicted to meet the regulatory minimum USE in about 55 calendar years after the original, adjusted startup date or 2039. Upon consideration of these various

impossible based on current knowledge, and may have resulted from discrepancies in the testing methods over time – or possibly by transposing the results from Capsules Y and V. This issue should be carefully examined by PG&E. Nevertheless, PG&E initially accepted the data as being credible.

contributions to the total uncertainty, I estimate that the uncertainty in the time taken for the weld to reach fracture criticality is about ± 10 EFPY. The uncertainty band appears to be dominated by the asymptotic nature of the curves (blue points) USE vs. EoE and Fluence vs. EFPY, as plotted in Figures 4 and 5, respectively. As a result, fracture criticality could be reached as soon as 33.8 EFPY, which is soon after the EOL of 32 EFPY, or as long as 53.8 EFPY, but safety prudence dictates that the lower number of 33.8 EFPY should be adopted. In my opinion, the uncertainty could have been reduced significantly had PG&E adhered to the capsule withdrawal schedule that was initially accepted from the NRC and had they followed the accepted scientific analytical method, as sanctioned by the NRC for the exclusion of identified problematic data.

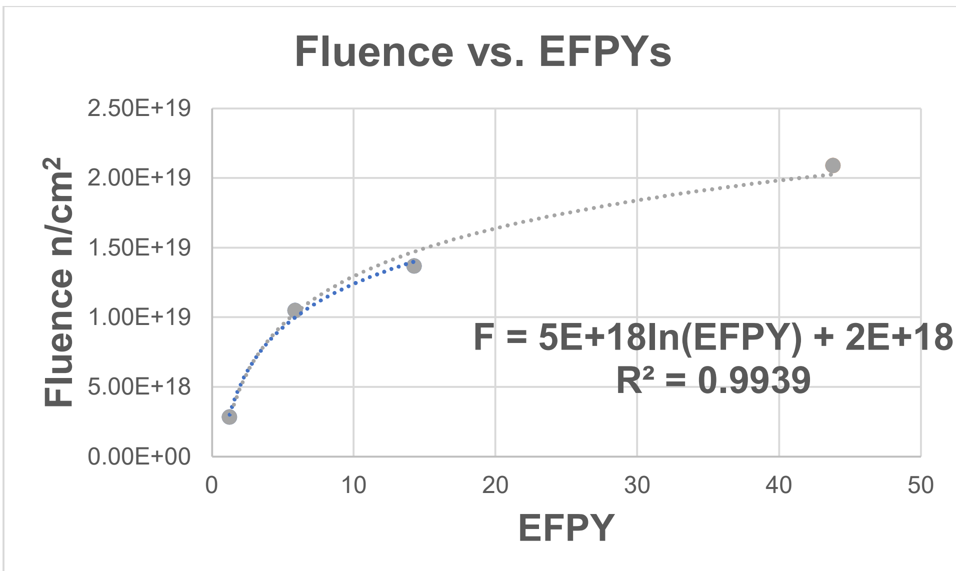


**Figure 5:** Plot of fluence vs the EFPYs when Capsules S, Y, and V were withdrawn from the reactor.

11. There is uncertainty in this projection, arising from four sources: (a) the inherent uncertainty in the data themselves; (b) the lack of any capsule surveillance data after 14.27 EFPYs; (c) the shape of the curves, particularly those in Figures 4 and 5, and (d) The length of the extrapolation, which is really a consequence of (b) above. Regarding the accuracy of USE, examination of the Charpy Impact Test data in WCAP-15958 suggests that the data are accurate to about ± 5 ft-lb. This number is important is determining the time at which the weld reaches the critical condition because, as shown in Figure 4, the USE vs. EoE plot approaches a limit asymptotically indicating that any uncertainty in USE becomes an increasingly larger uncertainty in EoE as the fluence increases. Thus, from Figure 5, this error is propagated into a corresponding uncertainty in the critical fluence that, in turn, is transferred to an uncertainty in the EFPY at which the critical condition is reached.

12. This analysis does not predict that the radiation embrittlement damage passes through an extremum (maximum or minimum) as is shown by PG&E's data (see, for example, the two highest fluence points in Figure 3), as that would require the expression for $m_i$

(given immediately above) to be a quadratic in the Fluence at the least. It seems more likely that the extrema simply reflect erroneous experimental technique and/or data analysis or that the data from Capsules Y and V were somehow transposed. Regardless of the speculated reason, if PG&E followed accepted scientific practice, they should have immediately inquired as to the reason for this anomalous result, but I can find no evidence that this was ever done. It is likely that this apparent sloppiness is responsible for the outliers that caused PG&E to reject all the data from Capsules S, Y, and V and leave them with no plant-specific data for Diablo Canyon Unit 1. Had they found the cause and identified the specific points in error, normal scientific practice would have justified rejection of those data while retaining the rest. As discussed in Section V.C, PG&E should have obtained more data by withdrawing and testing Capsule B, by testing other capsules that had already been withdrawn, by adding tensile strength testing, and by conducting a thorough ultrasound inspection.[11]

**B. The most recent ultrasound inspection of reactor vessel beltline welds in 2005 does not have credible results and therefore does not support a finding that Unit 1 is safe to operate.**

1. I am concerned by PG&E's 2014 statement that the results of its 2005 UT inspection of the pressure vessel were "essentially identical" to an inspection done 10 years earlier and yielded only one "indication" of voiding/cracking. PG&E (2014). It is reasonable to expect many more indications of voids and cracks, and that they would increase over time. For instance, in UT examinations of the Doel-3 and Tihannge-2 PWRs in Belgium conducted in 2012, up to 40 indications per $cm^3$ were detected in the Doel-3 reactor for a total of 7,776. Bogaerts et.al. (2022). Additional tests conducted in 2014 with adapted equipment detection parameters, revealed 13,047 voids and cracks in Doel-3 and 3,149 voids and cracks in Tihannge-2. Indications were found at depths ranging from 30 to 120 mm measured from the primary water side. Note that the thickness of the stainless-steel cladding is 7 mm, so that the indications occurred at 23 to 113 mm from the cladding/RPV steel interface. The indications were concentrated in the bottommost and upper core shell and were located in base metal, outside of the weld regions. These features can be correlated to steel microstructure and thermo-mechanical history (theoretical modeling) according to SCK-CEN, the Belgian Nuclear Research Centre. These indications were identified as "hydrogen flakes" and were postulated by Electrobel as having formed via excess humidity at the time of casting of the steel. However, the number of indications appear to be increasing with time which indicates that atomic hydrogen is entering from the primary side via the radiolysis of the $H_2$-rich primary side coolant (the PSC contains about 25 ccSTP) of hydrogen per kg of water), diffusing to and recombining in voids (e.g., clusters of metal vacancies), so as to pressurize the voids and causing the voids to grow on number and in size with some eventually transitioning into cracks.

[11] While we are aware that Capsule B apparently did not contain and beltline weld specimens, testing nevertheless would provide useful data.

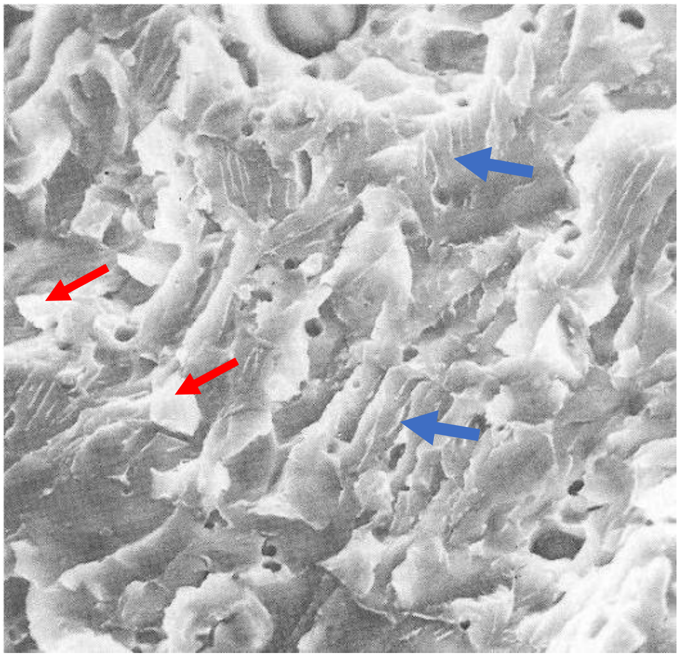

**Figure 6:** Typical "hydrogen flake" cracking in carbon or low-alloy steel. Typical features of hydrogen-induced brittle fracture are: micro-quasi-cleavage fracture, pores and fine hair-lines (indicating ductile fracture on a micro-scale). After Bogaerts et.al. (2015)

2. As shown by Bogaerts et.al. (2015), the microstructure contains both brittle (red arrows) and ductile (blue arrows) features, Figure 1, indicating mixed mode cracking not unlike that observed in other RPVs. Spencer and coworkers at INL have modeled RPV embrittlement within the Grizzly and FAVOR [Fracture Analysis of Vessels] codes. Spencer et.al. (2015, 2016). These are computer algorithms that were developed at Idaho National Laboratory (INL) and Oak Ridge National Laboratory (ORNL), respectively, for modeling the embrittlement and physical changes to RPVs under neutron irradiation. Typical distributions of the number of flaws in a RPV with respect to $RT_{NDT}$ as predicted by FAVOR and Grizzly are shown in Figure 7. FAVOR, which was developed at the ORNL, is acknowledged as providing an accurate prediction of the number and distribution of flaws in a PWR RPV and Grizzly are found to be in excellent agreement except for at the tail for $RT_{NDT} < 120$ °F.

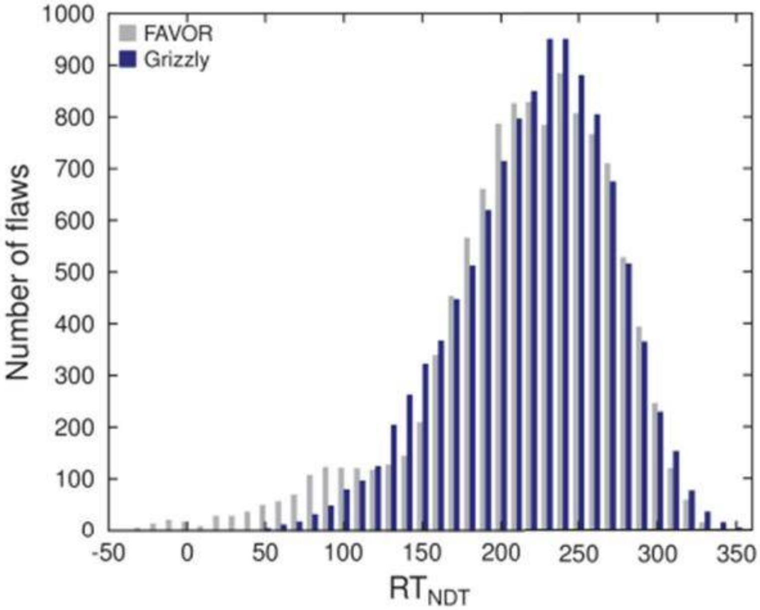


**Figure 7:** Comparison of $RT_{NDT}$ distributions in the same plate analysis in Grizzly and FAVOR. After Spencer et.al. (2016).

3. Accordingly, it is difficult to accept and understand PG&E's claim of detecting only one indication in the 2005 UT examination of beltline materials at Diablo-Canyon, Unit 1, when Figure 7 indicates thousands as determined by summing the number of indications for each bar. In my professional opinion, therefore, the anomalous results of the 2005 UT inspection should have prompted PG&E to evaluate whether the UT inspection was faulty and needed to be repeated. Instead, PG&E sought and obtained a ten-year extension of the 2015 deadline for the next UT inspection, until 2025. PG&E (2014), U.S. NRC (2015). *See also* Section V.D. below.

**C. PG&E has obtained no embrittlement data for Unit 1 for 18-20 years, at a significant risk to public health and safety.**

1. In my opinion, PG&E's failure to obtain embrittlement data since 2003 (Charpy test) and 2005 (UT inspections), plus the questionable quality of those tests and inspection, and on top of indications that embrittlement was occurring at a significant rate, raises serious questions that should be addressed immediately.

2. My concern stems in part from the complex nature of radiation embrittlement, which is idiosyncratic to individual reactors and may change unexpectedly over time, including periods of time less than a decade. Radiation embrittlement is a progressive phenomenon that increases with fluence, but which also depends on temperature. Thus, as the metal component of interest, is irradiated with high energy neutrons ($E > 1$ MeV), the fluence increases monotonically. The fluence, which is the neutron flux multiplied by the time of irradiation is, itself, independent of temperature but the rate of accumulation of damage in the metal is temperature dependent. This is because the various processes that contribute to the accumulation of damage, including the displacement of atoms into interstitial positions, the diffusion of the vacancies and interstitials through the lattice, the multiplication of the interstitial/vacancy pairs through cascading, the condensation of vacancies into clusters at impurities in the lattice that may grow into microscopic voids

and eventually form the macroscopic defects at which unstable cracks may nucleate under PTS conditions, and the recombination of interstitial/vacancy pairs, are thermally activated processes whose rates are temperature dependent.

3. Thus, while the fluence may be determined from the flux and the irradiation time regardless of the temperature, that is not the case for the irradiation damage. Westinghouse/PG&E calculate the fluence as though the reactor operates at full power for 80 % of the calendar years with the remaining 20 % accounting for downtime such as refueling. The resulting "effective full power years (EFPYs)" is therefore independent of whether the reactor operated at reduced power for periods (and hence reduced temperature) throughout the cycle or whether it operated at full power provided the end fluence was the same. However, this is not the case for the accumulated damage because the processes that contribute to the net damage are all thermally activated whose rates are temperature dependent. Because of this, the accumulation of damage depends upon the temperature history of the component, i.e., on the power level history. Thus, the case can be made that specifying $RT_{PTS}$ at a critical fluence would be better recast as $RT_{PTS}$ at a critical level of accumulated damage as measured by hardness, for example. This would appear, then, to fairly consider the effects of both temperature and fluence on the EFPYs required to achieve critical conditions.

4. I am also concerned by PG&E's reliance on data from so-called "sister" reactors that supposedly have similar characteristics. While this may be permissible as a stop-gap measure, PG&E has relied on data from other reactors for decades, instead of obtaining more data from Unit 1. As I have discussed above, complex industrial systems begin to differ in their characteristics almost as soon as they begin to operate. As has been noted by me and others, even if two nuclear plants are identical in every respect (and "sister" nuclear reactors never are), each soon becomes individualized by unique operating conditions and histories. Accordingly, in establishing correlations between accumulated damage (e.g., as measured by USE and/or $\Delta RT_{NDT}$) and fluence or EFPYs from many sister plants, this uniqueness must be recognized and built into the correlation.

5. Thus, if the sister plants were identical even after unique operating histories and the damage was normally distributed with respect to EFPY (a significant and poorly established assumption), a 1 sigma "scatter band" would yield a probability of only 68.2% that an additional datum added to the correlation would fall within that band (Figure 3). In my professional opinion as a scientist and an engineer, that probability is too low to be used for judging the probability of embrittlement in the Diablo Canyon Unit 1 vessel. However, because the sister plants and Diablo Canyon Unit 1 *do have unique operating histories* a larger uncertainty ("standard deviation") should be assigned that would significantly increase the width of the scatter band. Given the above, it is my opinion, that the 2-sigma scatter band, corresponding to a roughly 95.4 % probability that an additional plant (*e.g.*, Diablo Canyon Unit 1), and as specified in RG1.99, would fall within that band and would be more appropriate. By that standard, any legitimacy to

PG&E's decision to discredit the results from Capsules S, Y, and V collapses.

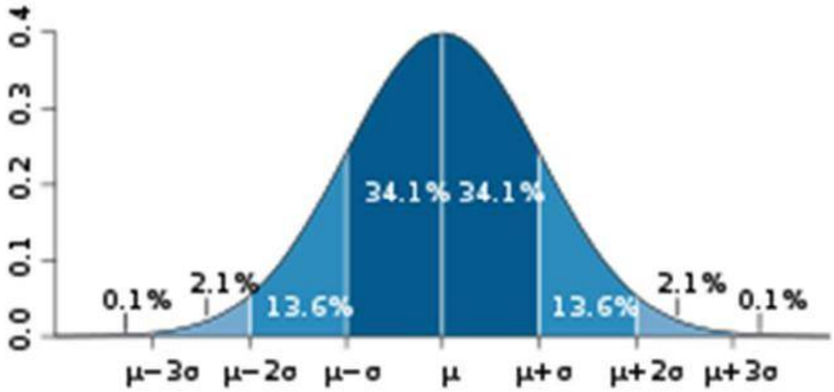


**Figure 8.** The normal distribution function displaying the probability of an additional observation falling within $\mu \mp n\sigma$, where n = 1,2,3,….∞.

6. Many uncertainties, including the memory effect arising from different operating histories arise in describing the evolution of radiation embrittlement damage that are not explicitly accounted for in the evaluation of correlation between $\Delta RT_{NDT}$ and fluence. Thus, numerous studies on the rupture of pipes in NPPs have established that the underlying statistics are Markovian, which specifies that what happens now depends on what happened in the past. I refer to this as the "memory effect" and, when applied to radiation embrittlement of NPP RPVs indicates that the rate of radiation embrittlement (RRE) in the present depends on the factors that controlled the RRE at some past time. For example, it is well established that the RRE is a function of temperature because the recombination of displaced (interstitial) atoms and vacancies, among other factors, is a thermally activated process and hence depends on the temperature.

7. Thus, the vessel, with respect to RRE, "remembers" past excursions in temperature, such as those associated with past shutdowns and restarts, and this factor contributes to the "individualization" of each plant. This also negates the application of strictly stochastic statistical methods in which the distribution can be defined in terms of a completely random distribution function such as the standard normal distribution. This is important, because in their fluence calculation, PG&E assumes that the neutron flux at the source (the core) is a constant when, in fact, the flux changes with the power level of the reactor and that may induce a "memory effect" that is not captured by defining operation in terms of EFPYs.

**D. The NRC's extension of the deadline for beltline ultrasound inspections is not supported by adequate data**

1. In my professional opinion, both PG&E and the NRC Staff have created an unacceptable safety risk by extending the deadline for removing and testing Capsule B a number of times from its originally scheduled removal in 2007 or 2009, to the point that PG&E does not plan to remove the capsule until the fall of 2023 or as late as the spring of 2025. As a result, PG&E has operated Unit 1 for two decades without essential information on the condition of the pressure vessel. And the gap is all the more concerning given the

indications of embrittlement in 2003 and further indications that some of the data were erroneous. Instead of postponing the next scheduled withdrawal and testing of a capsule, the Staff should have required PG&E to hasten the removal of Capsule B, and also to test whatever other capsules had been removed, using all available testing protocols, such as tensile (WOL) testing. Using all available protocols is especially important in light of the fact that Capsule B does not contain the limiting weld material that was in Capsules S, Y and V.

2. For several reasons, it is also my professional opinion that PG&E should conduct a UT inspection of beltline welds as soon as possible, preferably in the next refueling outage, rather than postponing it until 2025. First, as previously discussed, the UT inspection is both different and more reliable than the Charpy tests in that it detects and characterizes flaws that potentially could initiate unstable crack growth in the RPV under PTS conditions. Because it detects events that occur after the initial radiation embrittlement phenomenon, it has an independent value. Second, once PG&E had declared the Charpy data from Capsules S, Y, and V showed that Unit 1 was approaching regulatory limits and yet found the data not to be credible, it was incumbent on PG&E to acquire and evaluate as much additional data as possible, not to postpone obtaining it. Finally, PG&E inappropriately relied on reference temperature data from a sister reactor as input to the calculation of through-wall cracking frequency (TWCF). PG&E (2014), Enclosure at 6. As discussed above, reference temperature data from generic data bases or "sister" reactors should not have been relied on more than ten years after the 2003 Charpy tests for any purpose. Certainly, they should not be relied on to evade a UT inspection of the Unit 1 reactor vessel. The data is suspect and the reasoning is circular.

### E. Alternative testing methods would provide far more accurate results.

1. 10 C.F.R. § 50.51(c)(3) requires licensees to offer "information" that will "improve the accuracy of the $RT_{PTS}$ value significantly." The regulation doesn't apply only to CIT, which obtains one result per sample, and hence yields too few data to be statistically significant for a reasonable confidence level, but I am aware of the newly developed method of nano-indentation that is capable of obtaining many more replicate data than the conventional fracture mechanics methods prescribed by NRC regulations. The nano-indentation technique has been used for many years to assess embrittlement in steels and other alloys as reflected in a change in hardness. Briefly, a sharp point is pressed into a material under a known load and the dimensions of the indentation (width and depth) are measured. Thus, with increasing hardness, the depth and width of the indent become smaller. However, the relationship between hardness and $RT_{NDT}$ and USE still need to be established for this technique to replace the Charpy Impact Test. Nevertheless, I believe that can be done by using an Artificial Neural Network (ANN) to analyze the large body of information on $RT_{NDT}$ and USE vs. degree of embrittlement that is available from PWRs operating within the US and abroad.

2. I note that ASTM185-82 recommends indentation as an optional method for assessing the extent of embrittlement but it appears that too few plants have exercised that option to judge the viability of the method. However, the failed Charpy specimens are archived so

that the NRC could require each operator to measure the hardness using a suitable indenter and compile the results with as many independent variables (IVs) as possible.

3. The variables should include indentation width ($p_w$), indentation depth ($p_d$), fluence ($f$), temperature of irradiation ($T_{irr}$), copper content [Cu], nickel content [Ni], unirradiated yield strength (YS), unirradiated ultimate tensile strength ($UTS_{unirr}$), reduction of area upon fracture (RoA) and possibly others. The data should then be analyzed using artificial intelligence in the form of an artificial neural network (ANN) as presented in Figure 6. The independent variables would make up the input vector in the ANN as shown in the figure. This is the same ANN that I used to analyze the very large body of data from both the field and the laboratory on IGSCC in sensitized Type 304 SS in developing the CEFM. Shi, Wang, and Macdonald (2015). The net comprised one input layer, one output layer, and three "hidden layers", each containing as many neurons as the data contained in each input layer. All of the neurons in any given "hidden" layer are connected to all of the neurons in the preceding and following layers by interconnections of specific weights recognizing the bias associated with them. Establishment of the weights essentially imbues the net with ''memory'' and enables the relationships between the output and input layers to be established. The data collected from both laboratory and field studies are divided randomly into two groups; a training set and an evaluation set. The first set is used to train the net in a supervised, back propagation manner by incrementally adjusting the weights until the difference between the ANN predicted output and the known outputs satisfies some criterion such as the sum of the squares of that difference being minimal. Typically, this occurs after a few thousand to a few tens of thousands of iterations or about a few seconds of execution time on a laptop computer.

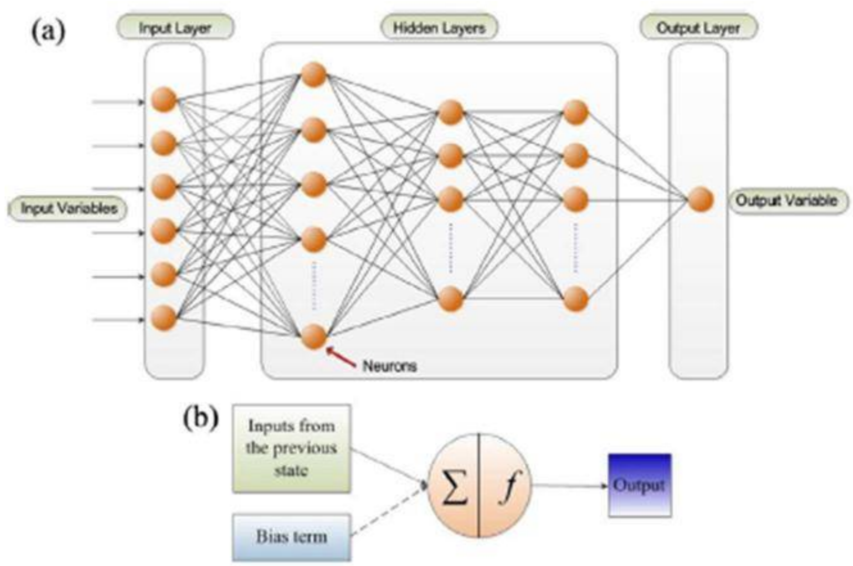


**Figure 9:** Artificial Neural Network for establishing relationships between the dependent variables ($RT_{NDT}$ and USE) and the vector of the Input Variables ($p_w$, $p_d$, $f$, $T_{irr}$, [Cu], [Ni], $YS_{unirr}$, $UTS_{unirr}$, RoA). Note that the neuron sums the values of the inputs from all preceding neurons and then applies a transfer function that determines

how the information is passed on to each of the neurons in the following layer with the amount of the information passed being determined by the weight of the connection between the two neurons.

4. It is important to note that no preconceived relationship between the output and the input is employed and the net has no physical theoretical basis. This extraordinarily powerful technique will define those relationships for us, with the result that we do not need to develop a theoretical physical model for the system. Once the ANN is trained and evaluated for accuracy using the evaluation data set, the net can be used to predict $RT_{NDT}$ and USE or some other parameter that measures the state of embrittlement of the RPV steel for any given indentation parameters. Because nano-indentation (or even classical indentation for that matter) requires very little material (< 2 $mm^2$), many sets of parameters can be obtained from each broken Charpy specimen (for example) thereby allowing the statistical basis of the $RT_{NDT}$ and USE to be explored in a manner that is not possible with the Charpy Impact Test method. The indentation method is quick (a few minutes per measurement) so that large databases of $RT_{NDT}$ and USE vs. the IVs can be developed without interfering with reactor operation. Furthermore, the addition of new data to the net represents continual retraining and refinement of the uncovered relationships between the dependent variables ($RT_{NDT}$ and USE) and the IVs. I suggest that this technology be developed and employed in a complementary manner until its advantages over the CIT have been established.

5. Professor Peter Hosemann, the developer of the nano indentation method at UC Berkeley and my fellow faculty in the Department of Nuclear Engineering kindly contributed the following material that describes the method in greater depth that my account given above and outlines some of his work on using it to characterize the radiation embrittlement of RPV steels. Any additions/clarifications other than correcting grammatical errors, such as missing articles, etc. that I have made to Prof. Hosemann's account are identified in italics.

6. In many nuclear applications there is simply not sufficient sample material available to provide a statistically sound and comprehensive dataset assessing a material mechanical property. In most instances, only a limited number of samples can be tested due to limited reactor space or the hazardous nature of the material. Nanoindentation is a technique assessing a material's hardness using an indenter that quantifies the force and the depth as a load is applied. Both force and displacement-controlled tools are available today. Assessing the force and displacement *in-situ* allows for a fully instrumentalized hardness measurement. Traditionally, a three-sided pyramid indenter (Berkovich) is used to perform the measurement that is calibrated against fused silica. The Oliver and Pharr method allows one to establish hardness and elastic property values. Other approaches utilize spherical indenters that are not self-similar but have the advantage of generating flow curves more directly.

7. Dynamic measurements (CSM, DMA, etc.) allow one to assess hardness as a function of indentation depth. Of course, hardness by itself is not a measure of yield strength or ductility at all but the properties measured using an instrumented hardness test or

nanoindentation allows them to be strongly correlated with these more engineering approaches. The real strength of nanoindentation originates with the fact that no elaborate sample preparation and shaping is required but only a nicely polished surface is needed. Furthermore, many datapoints can be collected within a matter of minutes and hours on a sample allowing one to assess local microstructures and provide statistics.

8. In recent years, scientists have spent significant effort to correlate and calculate more relevant engineering data from simple nano hardness measurements and utilize the benefits of large data numbers from indentation experiments. Several approaches emerged from these efforts allowing one to quantify yield strength as a function of irradiation conditions. Figure 10 shows one approach originally developed by Hosemann et al. and adopted and modified by Zinkle and others. In this approach, the nano hardness is used to calculate a macro hardness (corrected for pile up) which then in turn is used to calculate yield strength [Figure 10 (a)]. A blind test conducted over different reactor irradiated materials compares tensile test and shear punch test generated data to data obtained from nano hardness. As one can see there is a clear agreement between these very different measurements [Figure 10 (b)] again with the benefit that no elaborate sample preparation is needed while always collecting more than 15 datapoints per sample. Therefore, each datapoint is an average of 15 measured datapoints. *The large number of datapoints allows the distribution function to be determined and the appropriate error to be specified (e.g., the standard deviation) with an accuracy that is not possible using Charpy analysis.*

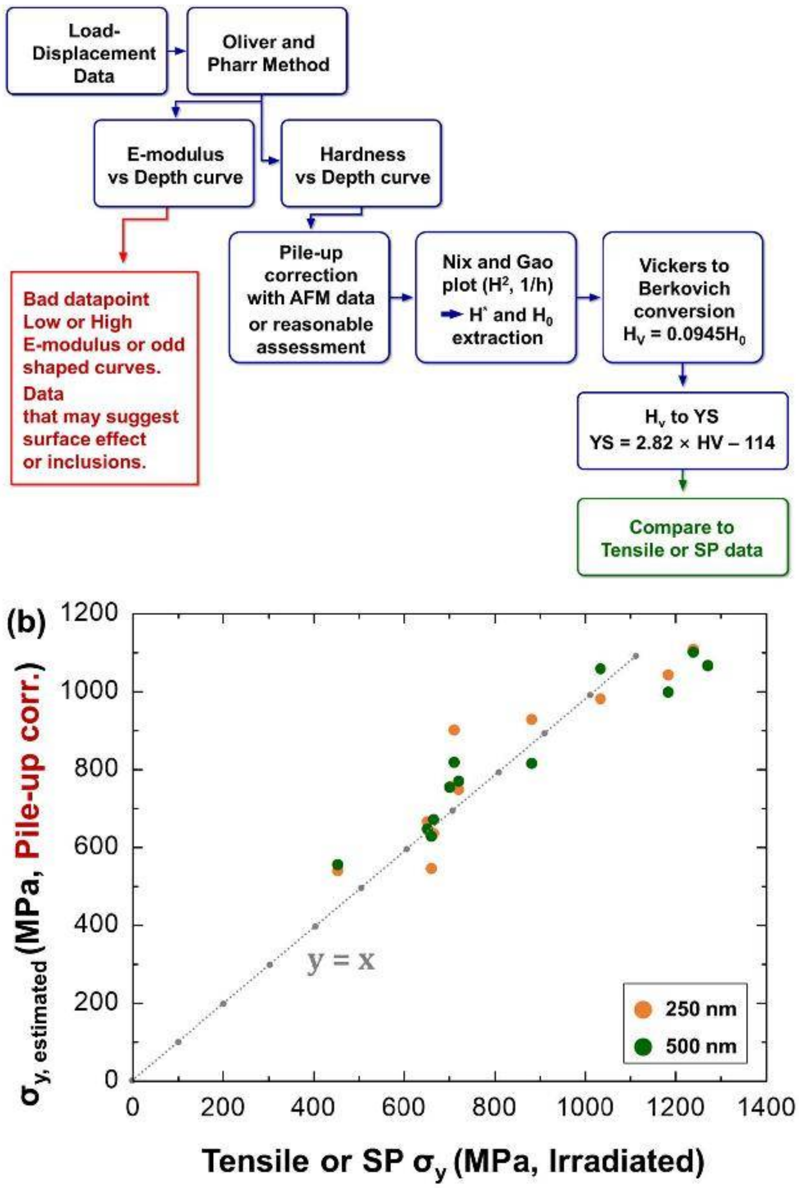


**Figure 10:** (a) Roadmap of nano indentation techniques. (b) Correlation between tensile test and shear punch test generated data to data obtained from nano hardness.

9. Of course, neither the yield strength nor the nanoindentation-obtained yield strength can make a direct statement about the strain to failure or embrittlement. However, the correlation investigating the temperature shift obtained by tensile testing with other more conventional methods such as Charpy or fracture toughness allows a comparison to be made. However, elevated temperature nanoindentation experiments are rare and not very common today but will need to be carried out in the future.

10. Other techniques such as spherical indentation have taken a slightly different approach. There the indentation can generate a direct measurement of yield strength from a single experiment. A direct comparison between different mechanical test techniques was made in the literature (Figure 11)

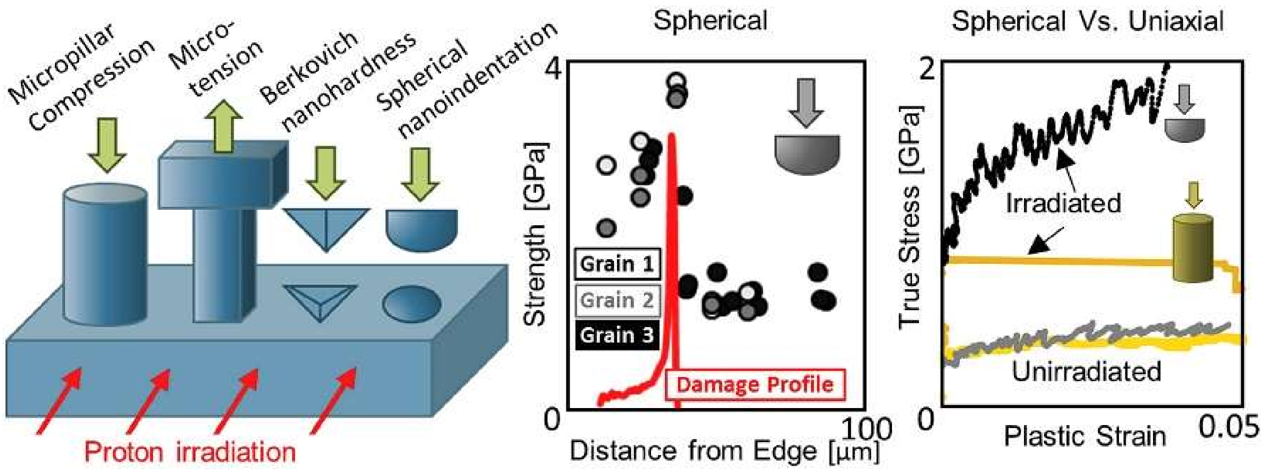


**Figure 11:** (a) Different micromechanical measurement tools; (b) Yield strength as a function of distance from a weld fusion line; and (c) True fracture stress vs plastic strain for irradiated and unirradiated RPV steel as measured using to micromechanical techniques depicted in (a).

11. Again, the key advantage of performing indentation in addition to other more conventional tests is the fact that one can conduct a near limitless number of measurements on the sample since the material is rather small not needing to cut specific sample geometries.

12. As matters currently stand, PG&E has no credible, plant specific data except for the 2005 UT examination, which PG&E claims (improbably) shows only one indication, to assess the state of embrittlement of the RPV of Diablo Canyon Unit 1 with which to assure the public of the reactor's safety. Given this, PG&E should be required to measure the hardness of the fractured Charpy specimens using the indentation method. These measurements should be performed of the actual weld metal, the HAZ, and the plate and be assessed against the unirradiated material. The method of analysis can follow that specified in RG1.99 and the critical hardness may be defined by plotting hardness vs, $ART_{NDT}$ and extrapolating the plot to the critical value of $ART_{NDT}$ for the weld dependent upon its orientation.

## VI. CONCLUSION AND RECOMMENDATIONS

1. For the reasons stated above, it is my professional opinion that the continued operation of Diablo Canyon Unit 1 poses an unreasonable risk to public health and safety and the environment.

2. Therefore, I recommend that the NRC Commissioners order the immediate closure of the reactor and that it must remain closed pending the completion of the following measures:

   a) Withdrawal and analysis of the contents of Capsule B as well as Capsules C and D (previously withdrawn but not analyzed);
   b) Evaluation and analysis of the WOL specimens contained in Capsules B, C and D and the archived capsules;
   c) Performance of nano indentation studies on the fractured remnants of the Charpy specimens from Capsules S, Y, and V;
   d) A comprehensive UT inspection of reactor vessel beltline welds;
   e) publication of the data from the 2015 UT inspection of reactor vessel beltline welds;
   f) A robust re-evaluation of the credibility of data from Capsules S, Y, and V that fully complies with NRC guidance and scientific principles:
   g) Any follow-up steps that may be appropriate for a finding of credibility of the data from Capsules S, Y, and V, including compliance with 10 C.F.R. 50.61a;
   h) Provision to the NRC, the ACRS, and the general public of all data and analyses that are 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; and
   i) A decision by the NRC Commissioners regarding the safety of continued operation that is informed by the outcome of a proceeding for public participation in the decision-making process.

3. In my professional opinion, nothing short of these steps can provide a reasonable level of assurance that Diablo Canyon Unit 1 is safe to operate – either currently or in a license renewal term.

Under penalty of perjury, I declare that the foregoing facts are true and correct to the best of my knowledge and that the opinions expressed herein are based on my best professional judgment.

*Executed in Accord with 10 CFR 2.304(d) by*
*Digby Macdonald*
Digby Macdonald September 14, 2023

# APPENDIX A: Curriculum Vitae

**DIGBY D. MACDONALD**
Professor in Residence, Departments of Nuclear Engineering
and Materials Science and Engineering
University of California at Berkeley
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Berkeley, CA 94720
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## EDUCATIONAL BACKGROUND

B.Sc. (1965) and M.Sc. (1966) in Chemistry, University of Auckland (New Zealand);
Ph.D. in Chemistry (1969), University of Calgary (Canada).

## PROFESSIONAL EXPERIENCE (past 52 years)

- Professor in Residence, Departments of Nuclear Engineering and Materials Science and Engineering, University of California at Berkeley, 1/2013 – present.
- Distinguished Professor of Materials Science and Engineering, Penn. State Univ.,6/2003 – 12/2012.
- Chair, Metals Program, Penn. State Univ., 6/2001 – 6/2003
- Director, Center for Electrochemical Sci. & Tech., Penn. State Univ., 7/99 – 12/2012.
- Vice President, Physical Sciences Division, SRI International, Menlo Park, CA, 1/98 – 7/99
- Director, Center for Advanced Materials, Penn. State Univ., 7/91-3/2000
- Professor, Materials Science and Engineering, Penn. State Univ., 7/91 – 6/03.
- Deputy Director, Physical Sciences Division, SRI International, Menlo Park, CA, 4/87 - 7/91
- Laboratory Director, Mat. Research Lab., SRI International, Menlo Park, CA, 4/87 – 7/91
- Laboratory Director, Chemistry Laboratory, SRI International, Menlo Park, CA, 3/84 – 4/87
- Director and Professor, Fontana Corrosion Center, Ohio State University, 3/79 – 3/84
- Sr. Metallurgist, SRI International, Menlo Park, CA, 3/77 – 3/79.
- Sr. Research Associate, Alberta Research Ltd/University of Calgary, Canada, 3/75 – 3/77.
- Lecturer in Chemistry, Victoria University of Wellington, New Zealand, 4/72 – 3/75.
- Assist. Research Officer, Whiteshell Nuclear Research Establishment, Atomic Energy of Canada Ltd., Pinawa, Manitoba, Canada, 9/69 – 4/72.

## CONSULTING ACTIVITIES (Partial list for the last twenty years).

OLI Systems
Electric Power Research Institute
SRI International
Stone & Webster Engineering Co.
Canadian Auto Preservation, Inc.
Numerous oil and gas companies.
SSM, Sweden.

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10. D. D. Macdonald, et.al., “Silicon Air Battery”, Int. Patent WO2011/061728A1, May 26, 2011.
11. D. D. Macdonald, et.al., “Silicon Air Battery”, US Patent, 8,835,060 B2, Sept. 16, 2014.

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2. J. Qiu, A. Wu, J. Yao, Y. Xu, Y. Li, R. Scarlat, D.D. Macdonald, “Kinetic study of hydrogen transport in graphite under molten fluoride salt environment”. Electrochim. Acta, 2020, 136459 (2020).
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## PROFESSIONAL ASSOCIATIONS AND HONORS

Research Award, College of Engineering, Ohio State University, 1983.
Selector of the Kuwait Prize for Applied Sciences, 1985.
The 1991 Carl Wagner Memorial Award from The Electrochemical Society.
The 1992 Willis Rodney Whitney Award from The National Association of Corrosion Engineers.
Chair, Gordon Research Conference on Corrosion, New Hampshire, 1992.
W.B. Lewis Memorial Lecture by Atomic Energy of Canada, Ltd., 1993, "in recognition of [his] contributions to the development of nuclear power in the service of mankind".
Elected Fellow, NACE-International, 1994.
Member, USAF Scientific Advisory Board, Protocol Rank: DE-4 (Lieutenant General equivalent), 1993-1997
Elected Fellow, The Electrochemical Society, 1995.
Elected Fellow, Royal Society of Canada, 1996. ("National Academy" of Canada).
Wilson Research Award, College of Earth and Minerals Sciences, Pennsylvania State University, 1996.
Elected Fellow, Royal Society of New Zealand, 1997. ("National Academy" of New Zealand).
H. H. Uhlig Award, Electrochemical Society, 2001.
U. R. Evans Award, British Corrosion Institute, 2003.
Elected Fellow, Institute of Corrosion (UK), 2003.
Appointed Adjunct Professor, Massey University, New Zealand, 2003.
Appointed Adjunct Professor, University of Nevada at Reno, 2003.
Elected Fellow, World Innovation Foundation, 2004.
Elected Fellow, ASM International, 2005.
Elected Fellow, International Society of Electrochemistry, 2006.
Khwarizmi International Award Laureate in Fundamental Science, Feb. 2007.
Trustee, ASM International, 2007-2010.
Appointed SABIC Visiting Chair Professor, King Fahd University of Petroleum and Minerals, Dhahran, Saudi Arabia, 2010.
Recipient, Lee Hsun Research Award, Chinese Academy of Sciences, China, 2010.
Inducted Doctuer Honoris Causa by INSA-Lyon, Lyon, France, 2011.
Nominated for the 2011 Nobel Prize in Chemistry for work on passivity.
Awarded the Faraday Memorial Trust Gold Medal, 2012.
Awarded the Gibbs Award in Thermodynamics by IAPWS, 2013
Awarded Frumkin Medal, ISE, 2014.
Awarded the OLIN Palladium Medal by the Electrochemical Society, 2015.
Received the Ad Augusta Award from Auckland Grammar School, 2016.

Plenary Lecturer, Corrosion2019, Nashville, TN, 2019.
Plenary Lecturer, Mexican Electrochemical Society, 2019.
Elected Member of the EU Academy of Science, 2019.
FLOGEN Fray International Sustainability Award for distinguished work in corrosion science.

Signed. Digby D. Macdonald.
September 13,2023.

# APPENDIX B: Reference List

## APPENDIX B: REFERENCE DOCUMENTS


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