Docket No.: A.24-03-018
Date: July 27, 2024
Commissioner: Douglas ALJ: Atamturk
Witness: Dr. Digby Macdonald

**BEFORE THE PUBLIC UTILITIES COMMISSION OF**

**THE STATE OF CALIFORNIA**

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| Application of Pacific Gas and Electric Company to Recover in Customer Rates the Costs to Support Extended Operation of Diablo Canyon Power Plant from September 1, 2023 through December 31, 2025 and for Approval of Planned Expenditure of 2025 Volumetric Performance Fees (U 39 E) | Application 24-03-018<br>(Filed March 29, 2024) |

# OPENING TESTIMONY OF DR. DIGBY MACDONALD ON BEHALF OF SAN LUIS OBISPO MOTHERS FOR PEACE

Dated: July 27, 2024

Dr. Digby Macdonald for SLOMFP
c/o
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**VERIFICATION**

The statements in the foregoing document are true and correct to the best of my knowledge. The facts presented in the forgoing document are true and correct to the best of my knowledge, and the opinions expressed therein are based on my best professional judgment. I declare under penalty of perjury under the laws of the state of California that the foregoing is true and correct.

Executed on 27/07/2024 in Fairfield, CA,

Digby Macdonald
Digby Macdonald (Jul 27, 2024 11:04 PDT)

Digby Macdonald

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## I. STATEMENT OF QUALIFICATIONS

My credentials as an expert in this matter are as follows: 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. My first position after completing my Ph.D. in 1969 was with Atomic Energy of Canada Ltd at their Whiteshell Nuclear Research Establishment in Pinawa, Manitoba, Canada, where I worked on solving the activity transport problem in the primary coolant circuit of the prototype CANDU (CANadian Deuterium moderated, natural Uranium) reactor at Douglass Point, Ontario. I was instrumental in solving that problem by proposing a redox shock strategy that removed the activated CRUD (Chalk River Unidentified Deposits) and in 1995 when the matter finally became public knowledge, I was honored to give the 6th W. B. Lewis Memorial Lecture in Fredrickton, NB. Four of the previous awardees were Nobel Laureates.

After a hiatus from the nuclear industry from 1971 to 1977, while I was a lecturer in Chemistry at Victoria University in Wellington, NZ, and a Research Associate at the Alberta Sulfur Research laboratories in Calgary, Canada, I moved to the US in 1977 and was employed as a Metallurgist at SRI International (Stanford Research Institute) until 1979. During that time, I began a long relationship with the Electric Power Research Institute (EPRI) in Palo Alto, CA working on nuclear reactor coolant and materials issues. In 1979, I accepted a Professorship in the Department of Metallurgical Engineering and Directorship of the Fontana Corrosion Center at the Ohio State University and managed a large EPRI-sponsored research program on stress corrosion cracking and corrosion fatigue in reactor materials. In 1984, I returned to SRI as Director of the Chemistry Laboratory and later as Director of the Materials Laboratory. In1991, I accepted a Professorship in Materials Science and Engineering at the Pennsylvania State University (PSU) to direct the Advanced Materials Center. I was later (1998) promoted to Distinguished Professor but in 1997 I took a leave of absence to become the Vice President of the Physical Sciences Division at SRI International. I returned to PSU in 1998 and continued my work on nuclear materials and coolant issues until I retired from PSU in December 2012. Shortly thereafter, UC Berkeley (UCB) invited me to join the Departments of Nuclear Engineering and Materials Science and Engineering as a Professor in Residence.

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. For example, under sponsorship of ONDRAF-NIRAS of Belgium, I have been able to predict 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 and under sponsorship of SSM, I have performed similar work on copper canisters in granitic rock repositories. 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 BWR and PWR primary coolant circuits.

In the case of BWR coolants, an artificial intelligence analysis (an artificial neural network) of CGR data from both field and laboratory sources for sensitized Type 304 SS 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. 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 SCC damage to 14 operating BWRs worldwide and where comparison with plant data can be made, the agreement between calculated and observed damage is excellent. For PWR primary coolant circuits, I have concentrated on addressing the Alloy 600 steam generator issues by first developing a fully deterministic model to calculate hydrogen-assisted SCC in that alloy in contact with primary coolant. Comparison with experimental CGRs shows that the Void Pressurization Model (VPM) is also capable of accurately predicting CGR in mill-annealed Alloy 600 under PWR primary coolant Conditions.

I and a student demonstrated that because of the large amount of hydrogen that is added to the coolant [25 cc(STP) $H_2$/kg $H_2$)] and because of the pH vs fuel burnup protocol employed (the Coordinated Water Chemistry Protocol), the corrosion potential drops below the critical potential for hydrogen induced cracking in the alloy. The solution to the primary side cracking in SG tubing problem 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) so as to maintain the corrosion potential above the critical cracking potential throughout a fuel cycle.

## II. PURPOSE AND BASIS OF TESTIMONY

I have been retained by San Luis Obispo Mothers for Peace (SLOMFP) to provide testimony relevant to the currently ongoing deliberations of the California Public Utilities Commission (CPUC) with respect to the conditional approval of extended operations of the Diablo Canyon Power Plant (DCPP). Specifically, my testimony pertains to Issue 1 in the Assigned Commissioner's Scoping Memo and Ruling for this proceeding, dated June 18, 2024.

I previously provided testimony in Phase 1 of the CPUC Rulemaking Proceeding R.23-01-007. In the Rulemaking Proceeding, my testimony discussed technical issues and challenges of extended operations pertaining to embrittlement at DCPP Unit 1. I incorporate by reference the entirety of my July 11, 2023 Opening Testimony in Phase 1 of R.23-01-007, including all analysis, attachments and reference materials as if fully set forth herein.[1]

Since that time, I have prepared subsequent and further analysis of the Unit 1 reactor vessel embrittlement issues in a variety of mediums (i.e., declarations, letters and technical papers) which have been submitted either to the Diablo Canyon Independent Safety Committee ("DCISC") or the Nuclear Regulatory Commission ("NRC"). While my July 11, 2023 Supplemental Opening Testimony remains unchanged, my new analysis underscores the gravity and seriousness of the embrittlement issue plaguing the reactor vessel in Unit 1, which PG&E still ignores. Particularly in the light of no new surveillance data having been obtained over the past twenty years that would indicate otherwise.

## III. SUMMARY OF MY RECENT ANALYSIS OF EMBRITTLEMENT OF THE DCPP UNIT 1 REACTOR VESSEL

*September 14, 2023 Declaration*

On September 14, 2023, SLOMFP, Friends of the Earth and the Environmental Working Group filed a Hearing Request and Request for Emergency Order with the United States Nuclear Regulatory Commission ("Hearing Request"). In support of the Hearing Request, I prepared a September 14, 2023 Declaration (hereinafter "September 2023 Declaration" or "Declaration").[2] The Declaration explained the reasons why the DCPP Unit 1 reactor vessel 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. I described the compliance with the complementary three part NRC regulatory framework for monitoring the condition of the plant-specific pressure vessel and how it is essential that these must be considered in unison.[3] I went into further detail about: 1) the importance of pressure vessel integrity; 2) PG&E's failure to consider credible data that Unit 1 is nearing PTS temperature screening criteria; 3) how the most recent ultrasound inspection of reactor vessel beltline welds in 2005 does not have credible results; 4) how PG&E has obtained no embrittlement data for

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[1] Exh. SLOMFP_07 Supplemental Opening Testimony of Dr. Digby Macdonald on Phase 1 Track 2 Issues [https://docs.cpuc.ca.gov/PublishedDocs/SupDoc/R2301007/6448/513343374.pdf]
[2] Attachment A [September 14, 2023 Declaration, pp. 1-46]
[3] *Ibid*.

Unit 1 for 18-20 years, at a significant risk to public health and safety; 5) how The NRC's extension of the deadline for beltline ultrasound inspections is not supported by adequate data; and 6) how alternative testing methods would provide far more accurate results.[4]

*DCISC Technical Papers*

I prepared a June 14, 2024 Technical Evaluation of Mark Kirk's reporting[5] on the conditions of the DCPP Unit 1 Pressure Vessel and described in detail the fundamental scientific flaws and errors in the reporting.[6] This evaluation was submitted to the DCISC. In response to DCSIC questions regarding my evaluation, I then prepared a July 19, 2024 Letter responding to the DCSIC's questions in detail.[7] In the process of preparing this material, I reviewed the most recent DCISC Fact-Findings that discussed embrittlement at DCPP Unit 1.[8] To the best of my current knowledge, the DCSIC is still considering my responses at this time.

**IV. CONCLUSION**

In summary, I reiterate the conclusions I made in my July 11, 2023 Supplemental Opening Testimony on Phase 1 Track 2 in the CPUC Rulemaking Proceeding R.23-01-007. My recent analysis underscores the glaring and fundamental scientific flaws in Mark Kirk's reporting and in PG&E's insistence that embrittlement is a non-issue unworthy of consideration and further evaluation. Expensive repair and/or replacement projects at DCPP to address embrittlement would cause PG&E's projected costs for extended operations to skyrocket. I continue to stand by my conservative estimate of at least $250,000,000 to $500,000,000 for this work.

This concludes my testimony.

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[4] Attachment A [September 14, 2023 Declaration pp. 1-46].
[5] Attachment B [Mark Kirk Reporting]
[6] Attachment C [June 14, 2023 Technical Evaluation pp. 1-54].
[7] Attachment D [ July 19, 2024 Letter Responding to DCSIC Questions, pp. 1-76].
[8] Attachment E [DCISC February 29, 2024 Letter and and June 20, 2024 DCISC Agenda Transmittal Form].

# ATTACHMENT A

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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## 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 270°F 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

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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.

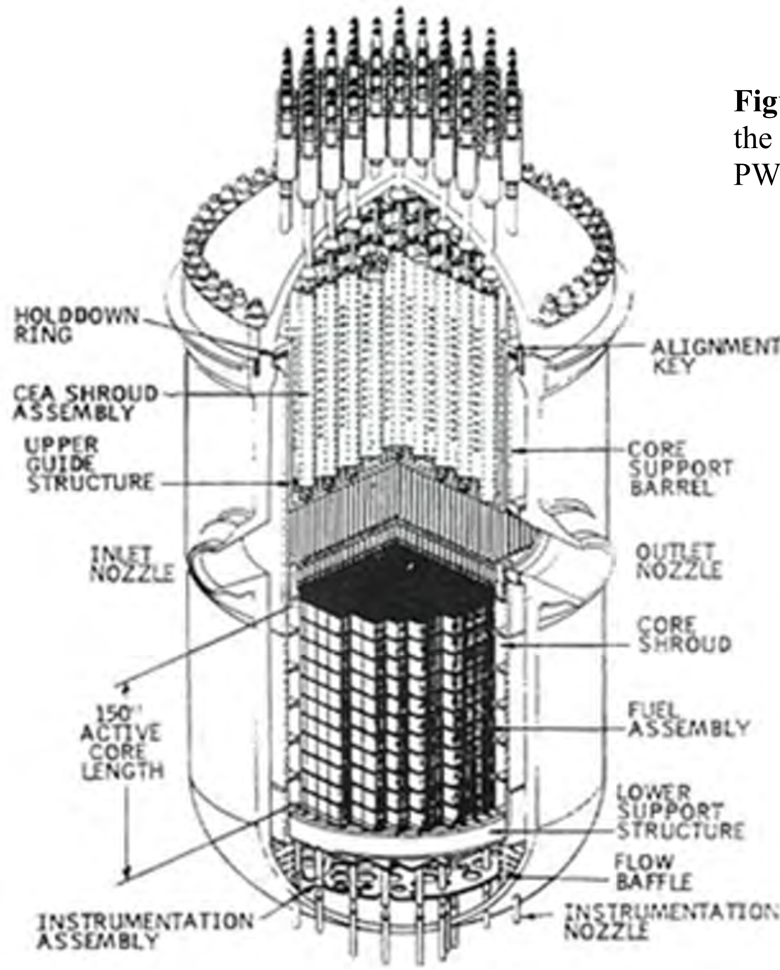


**Figure 1**: Cut-away schematic of the core of a typical Westinghouse PWR.

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.

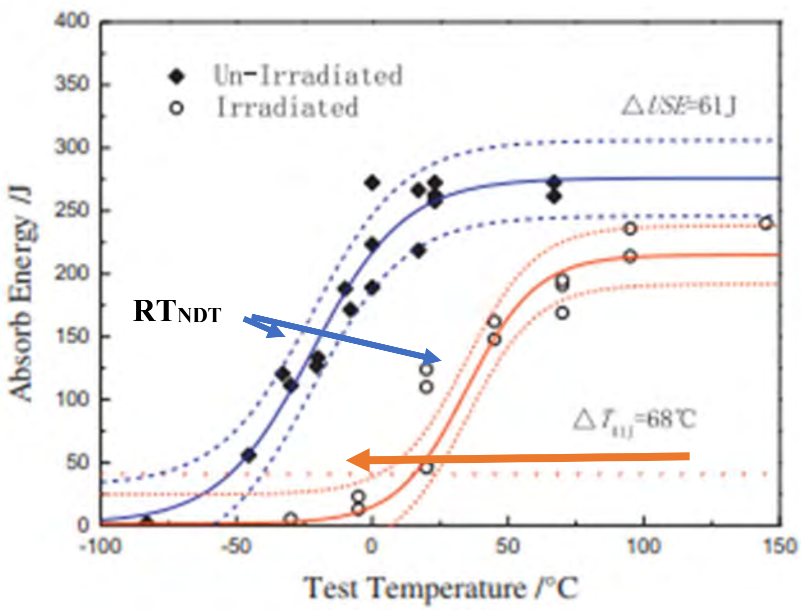


**Figure 2:** Effect of neutron irradiation on the Charpy impact test results for a fluence of $10^{20}$ n/cm$^2$ (E > 1 MeV) for A508-3 RPV steel. After Lin, et.al. Note that irradiation cause the value of $RT_{NDT}$ to shift by about 68 ºC (154 ºF) and the USE to be reduced by 61 J.

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 x $10^{19}$ 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 ºF. PG&E (2023), Westinghouse (2003). This calculation should have concerned PG&E because it was approaching the PTS screening criterion of 270 ºF for plates, forgings and axial weld materials and within a reasonable margin of error of about ± 10 ºF (as estimated by inspection of the Charpy curves), resulting in an overlap of uncertainties in the screening criterion (270 ºF) and the Westinghouse estimate (250.9 ºF) 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 ºF) was lower than the value for Capsule Y that had been removed ten years earlier at 1R5 (232.59 ºF) (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 28ºF for welds and 17ºF 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

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[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

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[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.

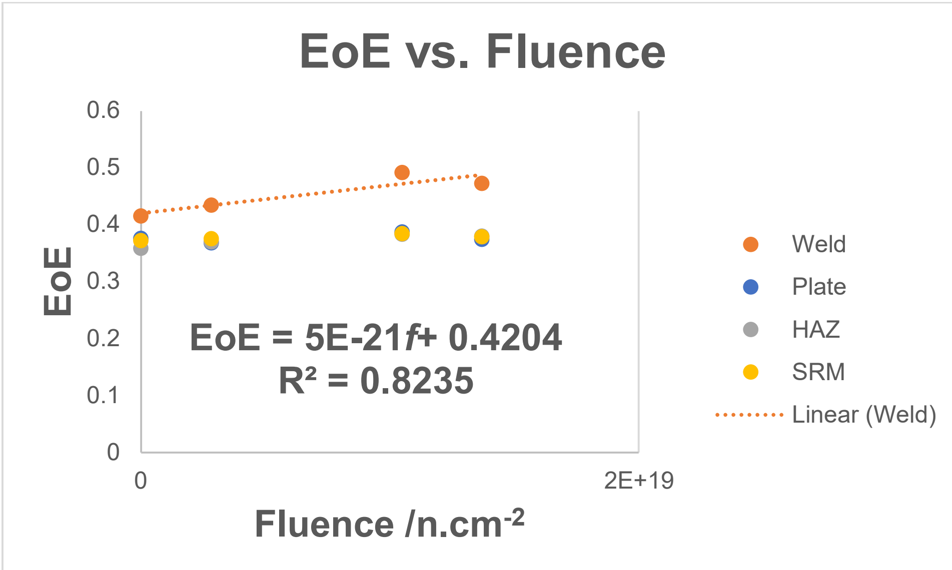


**Figure 3:** Values for EoE derived from the CIT data of PG&E for metal specimens from Capsules S, Y, and V that were exposed in Diablo Canyon Unit 1.

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

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[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 $RT_{NDT,30}$.

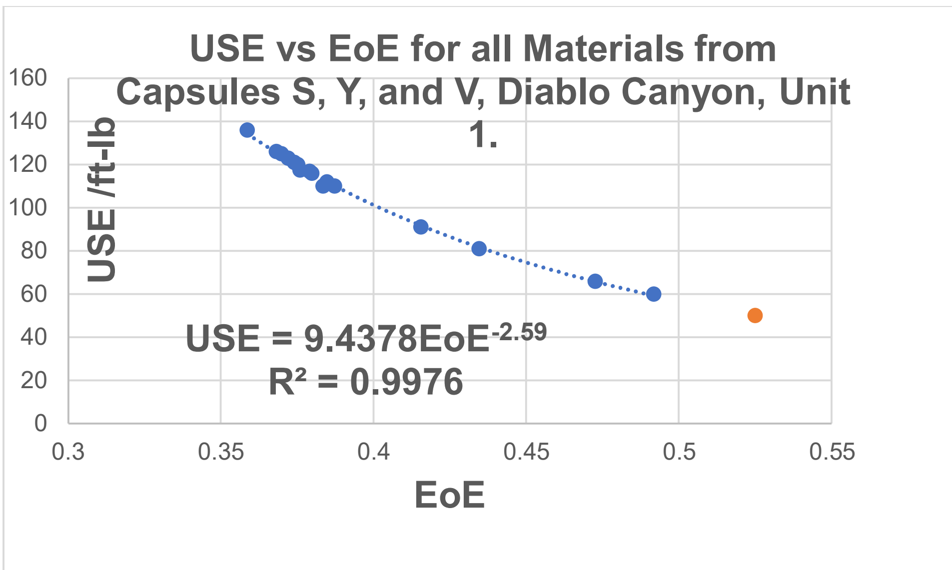


**Figure 4:** Plot of USE vs. EoE for all materials from Capsules S, Y, and V, Diablo-Canyon, Unit 1 NPP.

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.

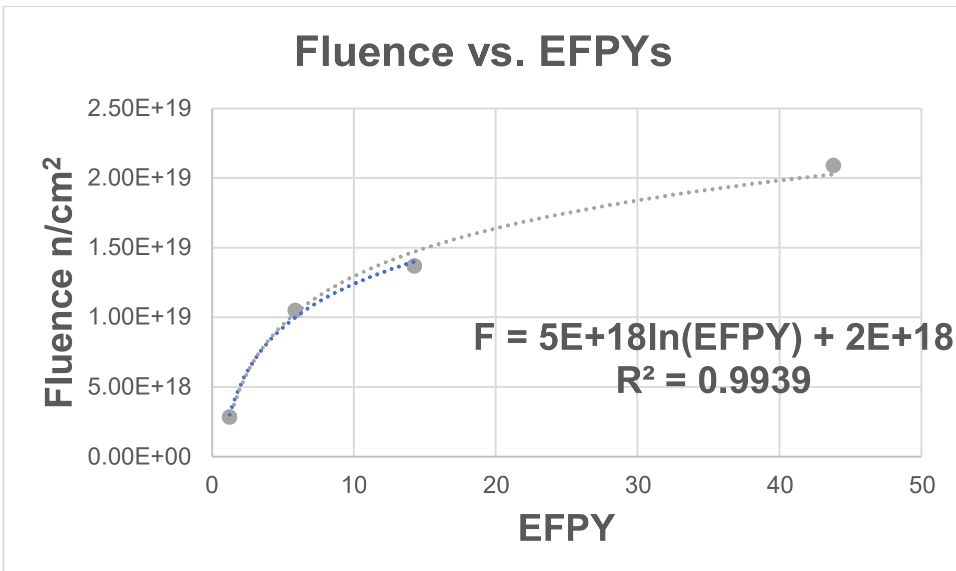


**Figure 5:** Plot of fluence vs the EFPYs when Capsules S, Y, and V were withdrawn from the reactor.

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.

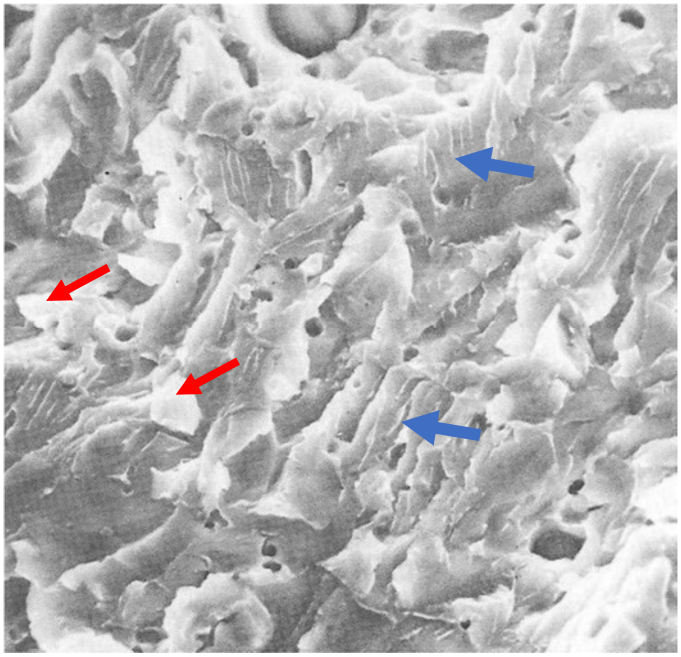

**Figure 6:** Typical "hydrogen flake" cracking in carbon or low-alloy steel. Typical features of hydrogen-induced brittle fracture are: micro-quasi-cleavage fracture, pores and fine hair-lines (indicating ductile fracture on a micro-scale). After Bogaerts et.al. (2015)

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.

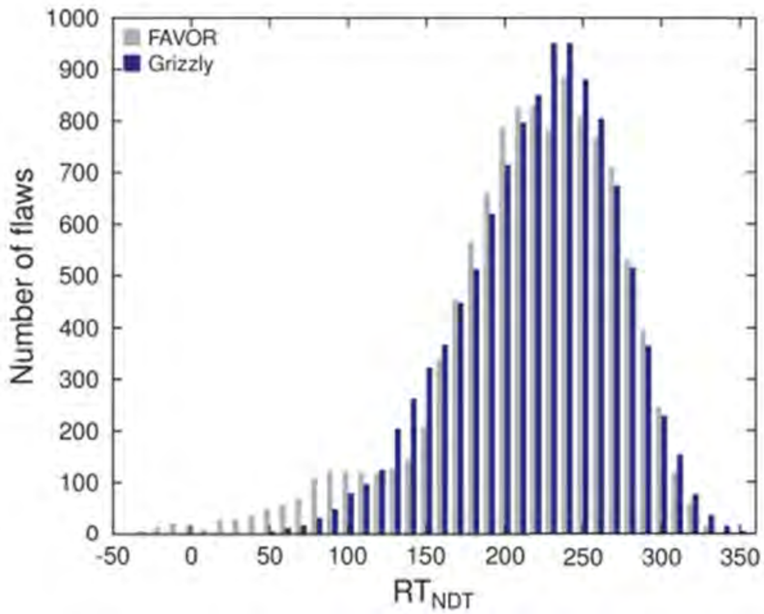


**Figure 7:** Comparison of $RT_{NDT}$ distributions in the same plate analysis in Grizzly and FAVOR. After Spencer et.al. (2016).

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.

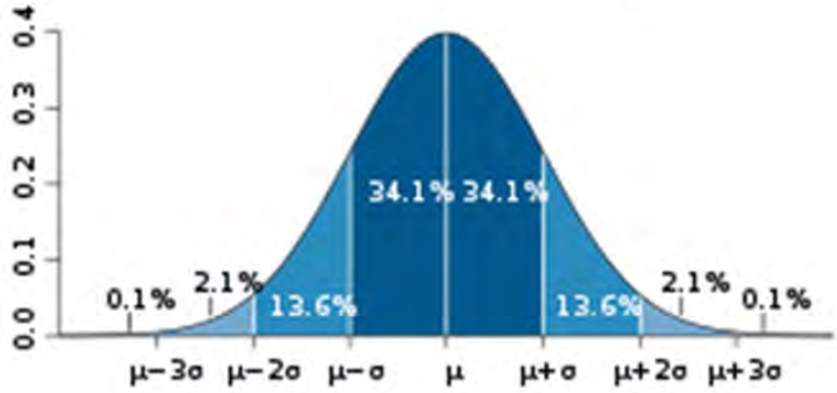


**Figure 8.** The normal distribution function displaying the probability of an additional observation falling within $\mu \mp n\sigma$, where n = 1,2,3,….∞.

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.

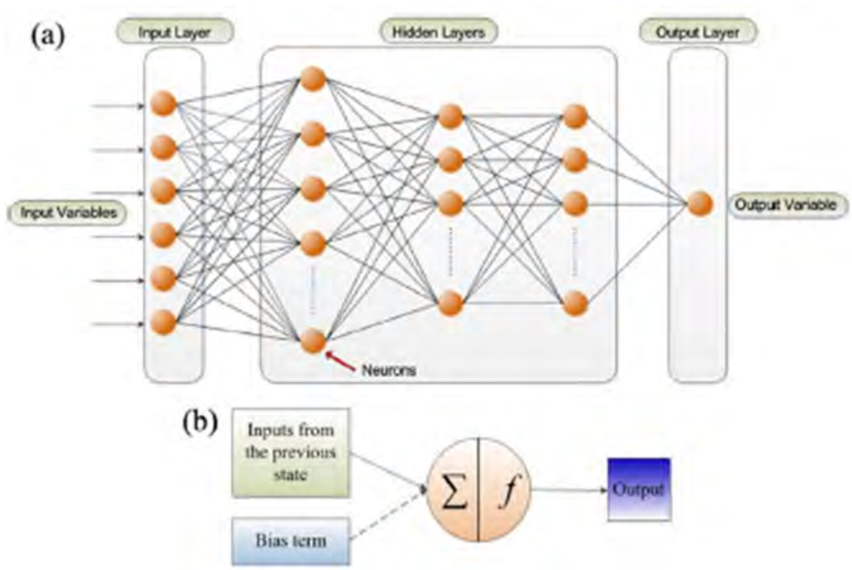


**Figure 9:** Artificial Neural Network for establishing relationships between the dependent variables ($RT_{NDT}$ and USE) and the vector of the Input Variables ($p_w$, $p_d$, $f$, $T_{irr}$, [Cu], [Ni], $YS_{unirr}$, $UTS_{unirr}$, RoA). Note that the neuron sums the values of the inputs from all preceding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

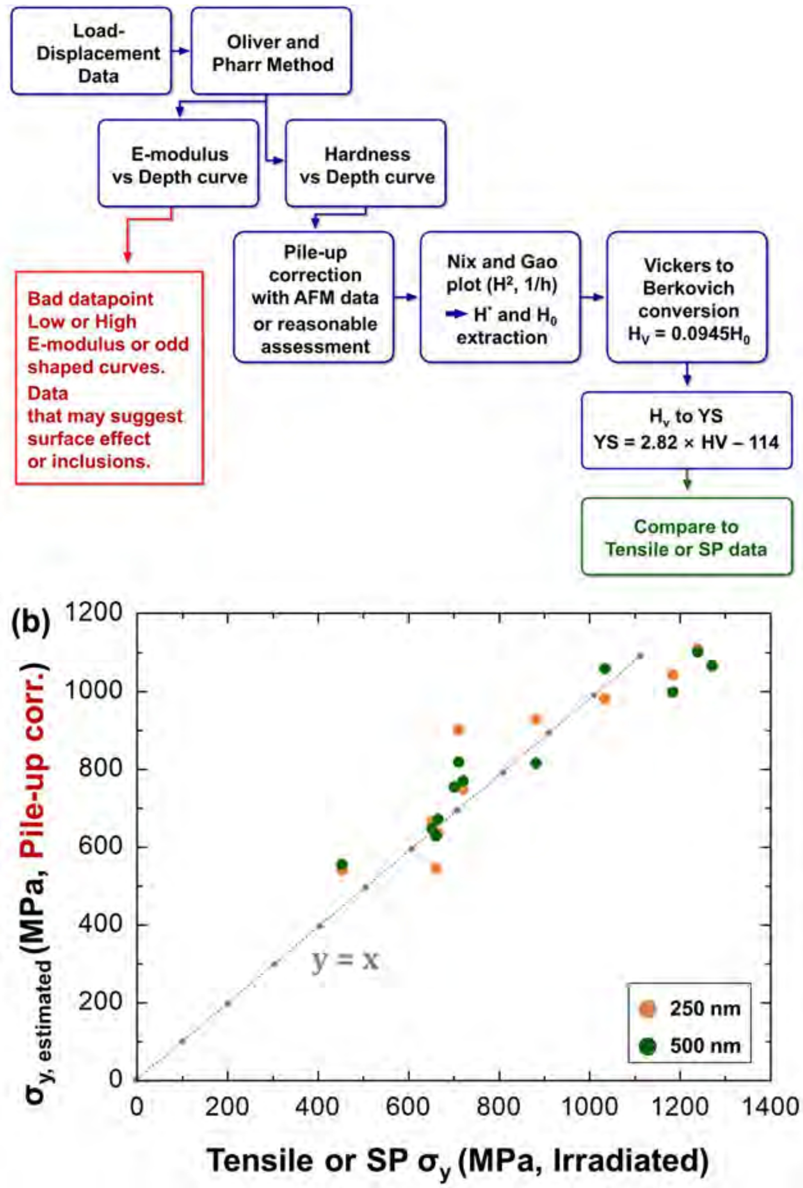


**Figure 10:** (a) Roadmap of nano indentation techniques. (b) Correlation between tensile test and shear punch test generated data to data obtained from nano hardness.

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)

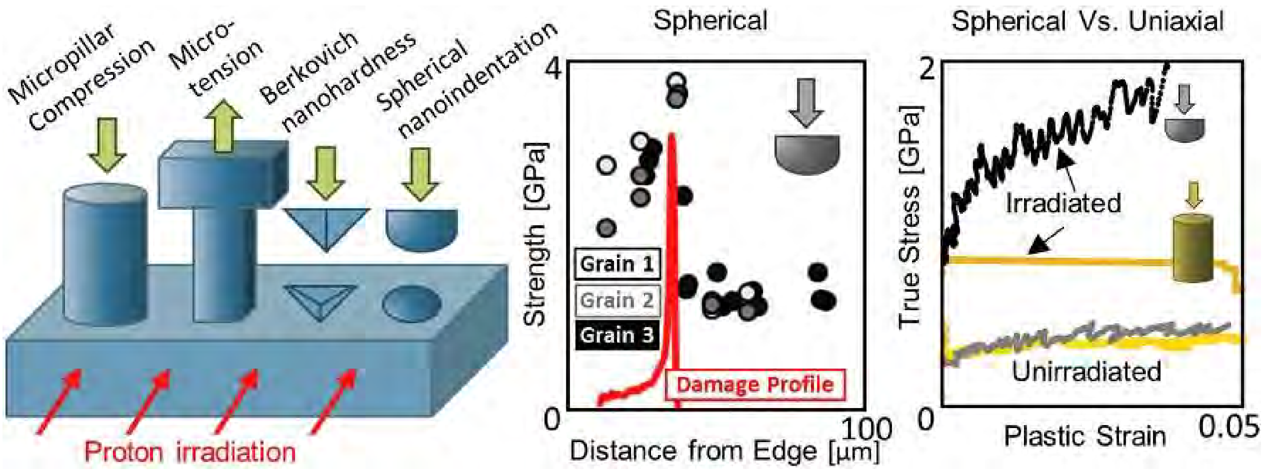


**Figure 11:** (a) Different micromechanical measurement tools; (b) Yield strength as a function of distance from a weld fusion line; and (c) True fracture stress vs plastic strain for irradiated and unirradiated RPV steel as measured using to micromechanical techniques depicted in (a).

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
4151 Etcheverry Hall
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.

## PATENTS

1. D. D. Macdonald and A. C. Scott, "Pressure Balanced External Reference Electrode Assembly and Method", US Patent 4,273,637 (1981).
2. D. D. Macdonald, "Apparatus for Measuring the pH of a Liquid", US Patent 4,406,766 (1983).
3. S. C. Narang and D. D. Macdonald, Novel Solid Polymer Electrolytes", US Patent 5,061,581 (1991).
4. S. Hettiarachchi, S. C. Narang, and D. D. Macdonald, "Synergistic Corrosion Inhibitors Based on Substituted Pyridinium Compounds", US Patent 5,132,093 (1992).
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11. D. D. Macdonald, et.al., "Silicon Air Battery", US Patent, 8,835,060 B2, Sept. 16, 2014.

**RELEVANT PUBLICATIONS** (from a total of ≈ 1000).

1. D. D. Macdonald and G. R. Engelhardt, "Predictive Modeling of Corrosion". In: Richardson J A et al. (eds.), *Shrier's Corrosion*, 2, 1630-1679 (2010). Amsterdam: Elsevier.
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).
3. J. Yao, J. Qiu, F. Carotti, R. Scarlat, D.D. Macdonald, Kinetic study of the hydrogen charging reaction on the graphite in aqueous solution and in room temperature ionic liquid (RTIL), Electrochimica Acta, 330, 135291 (2000).
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5. D Kovalov, B Fekete, G. R Engelhardt, D. D Macdonald, Prediction of Corrosion Fatigue Crack Growth Rate in Alloys. Part II: Effect of Electrochemical Potential, NaCl Concentration, and Temperature on Crack Propagation in AA2024-T351, *Corrosion Science*, 152. 130-139 (2019).
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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

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Spencer, B.; Backman, M.; Chakraborty, P.; and Hoffman, W, 2015. "Reactor Pressure Vessel Fracture Analysis Capabilities in Grizzly", INL/EXT-15-34736, Idaho National Laboratory, Idaho Falls, ID, Mar. 2015.

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# ATTACHMENT B

# An Evaluation of the Status of Diablo Canyon Unit 1 With Respect to Reactor Pressure Vessel Condition Monitoring and Prediction

## Part 1: Addressing Public Concerns

Consultant's report to the Diablo Canyon Independent Safety Committee


Submitted by

Mark Kirk

Phoenix Engineering Associates Inc.

Unity, New Hampshire, USA


Date: 26 January 2024

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***E***ngineering

***A***ssociates

***I***nc.

## Disclaimer

This report summarizes the consultant's evaluation of the state of embrittlement in the Diablo Canyon Unit 1 reactor pressure vessel and various questions related to this topic. The report is provided as information to the Diablo Canyon Independent Safety Committee (DCISC). The consultant has no responsibility for decisions made by the DCISC, or by any other body, based on the information in this report.

# Executive Summary

From 2009 to 2018 the Pacific Gas and Electric Company (PG&E) pursued a 20-year license extension for the nuclear power plant at Diablo Canyon with the Nuclear Regulatory Commission (NRC), an effort terminated in 2018 due to then-projected energy demands and economic factors. In 2022 the State of California directed the California Public Utilities Commission (CPUC) to direct PG&E to again pursue license extension. Subsequently, members of the San Luis Obispo Mothers for Peace (SLOMFP), the Friends of the Earth (FOE), and Mr. Bruce Severance placed before the Diablo Canyon Independent Safety Committee (DCISC) concerns regarding embrittlement[1] of the Unit 1 reactor pressure vessel (RPV) and, consequently, its continued operating safety. SLOMFP and FOE have expressed similar concerns to both the CPUC and the NRC.

This is one of two reports prepared for the DCISC. This report includes introductory material describing the procedures used for nuclear RPV surveillance, for embrittlement and fracture toughness[2] forecasting, and for RPV safety evaluation to address the concerns of SLOMFP and FOE. The companion report evaluates the current state of knowledge of embrittlement in the Diablo Canyon Unit 1 RPV and reviews that Unit's current RPV safety analyses.

A detailed review of all documents provided by SLOMFP, FOE, and Mr. Bruce Severance revealed concerns across a broad range of topics. For clarity of discussion these concerns were parsed into five categories. In this Executive Summary these concerns are necessarily condensed in length, however in the main body of this report the concerns are quoted directly to best retain their meaning and intent. An evaluation of each set of concerns follows:

1. **Questions concerning the analysis of surveillance data:**

   <u>Concern</u>: SLOMFP and FOE expressed concerns regarding the "credibility" of available surveillance data, repeated deferrals to testing of surveillance[3] Capsule B, the use of surveillance data from other plants in the Diablo Canyon Unit 1 evaluation, and concerns about the analysis methodology itself.

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[1] "Embrittlement" is caused by neutron irradiation, which hardens the steel thereby reducing its resistance to fracture.

[2] "Fracture toughness" is a property that quantifies a material's resistance to fracture.

[3] "Surveillance" is a required process by which the toughness of RPV materials are monitored during plant operation.

References: These SLOMFP and FOE concerns are expressed on pdf pages 55-58 and 65-67 of Citation 1 and on pdf page 188 of Citation 2. Citations appear at the end of this Executive Summary.

Evaluation

- These concerns stem from an apparent misunderstanding of the NRC's definition of "credibility" in the context of an embrittlement analysis, and of the process used for embrittlement prediction and RPV surveillance. Diablo Canyon Unit 1 correctly followed NRC guidelines concerning surveillance data set credibility both when the data were first assessed to be not credible (2003-2011) and more recently (since 2011) when, with additional data from the Palisades reactor, the data set was assessed to be credible. Even when data are assessed as not credible, they are neither discarded nor discredited, but rather are used to inform an estimate of future embrittlement trends with added conservatism.
- The multiple deferrals granted by the NRC to PG&E for withdrawal of Capsule B are consistent with NRC guidance and with deferrals granted to other plants under similar circumstances. Currently available surveillance data pertinent to the Diablo Canyon Unit 1 RPV have already quantified embrittlement at irradiation exposure levels that the RPV will not reach until after 60 years of operation, indicating that deferral of the withdrawal and testing of Capsule B has not compromised plant safety.
- Use of surveillance data from similar materials at other plants is required by the NRC to better inform embrittlement predictions. PG&E has met this requirement using data from another plant (Palisades). This material has very similar composition and exposure conditions to the Diablo Canyon Unit 1 weld.
- An alternative analysis methodology proposed by Professor Macdonald was assessed using available data published in the literature from a wide variety of RPV steels. These data demonstrated the alternative methodology to be 2½ times less accurate than the methodology currently mandated by the NRC and used by PG&E.

## 2. Questions concerning inspections of the RPV beltline[4]

Concern: SLOMFP and FOE expressed concerns that the small number of indications[5] found by non-destructive evaluation of the Diablo Canyon Unit 1 RPV are not plausible, that the time interval permitted between inspections is too long, and that important time-dependent cracking phenomena have been ignored.

4 "Beltline" means the region of the cylindrical RPV shell adjacent to the reactor core.

5 "Indication" means a signal detected during a non-destructive examination of the RPV using ultra-sound. An indication needs to be assessed by an inspector and/or engineer to determine its impact, if any, on plant safety.

References: These SLOMFP and FOE concerns are expressed on pdf pages 63, 65, and 68 of Citation 1 and on pdf pages 8 and 218 of Citation 2. Citations appear at the end of this Executive Summary.

Evaluation

- The limited number of indications reported by Diablo Canyon Unit 1 is consistent with inspection results from other plants. The evidence cited supporting the existence of a very large numbers of flaws in the RPV came from a model published by the NRC that conservatively treated every indication, even volumetric indications, as sharp-tipped flaws that could be detrimental to plant safety.
- RPV steel embrittlement does not cause cracking, so there is no relationship between the degree of embrittlement and the frequency of in-service inspections, which can only detect cracks. The in-service inspections of RPV welds that are required by NRC and described by ASME are performed every 10 years. Because there is no time dependent cracking mechanism operative in the RPV beltline, the 10-year inspections become a re-examination of an unchanging condition every 10 years.
- Concerns expressed that environmentally assisted crack growth (hydrogen cracking, stress corrosion cracking) of the RPV steel and of the austenitic stainless-steel liner might occur are not borne out by decades of inspection evidence. Control of the chemistry of the primary coolant to limit the oxygen content is the factor most responsible for this lack of environmentally assisted cracking.

## 3. Suggestions on alternative testing methods to characterize irradiation damage

Concern: SLOMFP and FOE suggested that "nano-indentation hardness can provide additional data on the irradiation damage experienced by Diablo Canyon Unit 1 materials, and with greater certainty than the Charpy impact test."

References: These SLOMFP and FOE concerns are expressed on pdf pages 68, 70, and 7 of Citation 1. Citations appear at the end of this Executive Summary.

Evaluation: In principle a hardness value can be correlated to yield strength, and yield strength can then be correlated to either fracture toughness or Charpy toughness. However, no regulatory precedent exists for use of fracture toughness values estimated in this way. The uncertainties introduced to a measurement – however precisely and repeatedly made – by such a sequence of empirical correlations will degrade the ability of such data to illuminate the embrittlement trends of Diablo Canyon Unit 1. If re-testing of previously irradiated samples is needed, directly measuring fracture toughness using mini compact tension specimens is possible using standardized techniques with which there is regulatory precedent.

4. **Questions regarding the methodology for RPV safety assessment**

   Concern: SLOMFP and FOE expressed concerns about requirements of the NRC's two pressurized thermal shock (PTS[6]) rules, the correct treatment of the RPV's stainless steel liner, the treatment of low-temperature thermal annealing, and the requirements for evaluation of the extended beltline.

   References: These SLOMFP and FOE concerns are expressed on pdf page 191 of Citation 1 and on pdf pages 9, 10, 12, and 13 of Citation 2. Citations appear at the end of this Executive Summary.

   Evaluation

   - For Diablo Canyon Unit 1, PG&E has demonstrated compliance with the NRC's original PTS rule (10 CFR 50.61), making compliance with the alternate PTS rule (10 CFR 50.61a) unnecessary.
   - The treatment of the stainless-steel liner that is weld-deposited on the inside of the RPV is conservative in both PTS rules. Postulates of stress-corrosion cracking of the liner are not supported by inspection data. In any event, stress-corrosion cracking is not expected due to stringent control of the chemistry of the coolant water in the RPV.
   - Low temperature thermal annealing is measured by the RPV surveillance program and thus is accounted for by existing procedures.
   - PG&E has addressed extended beltline materials, such as nozzles, following NRC guidance. This assessment demonstrated that even for a 60-year plant life the weld in the RPV beltline will remain the most embrittled and, thus, will limit the plant's operation to a greater extent than any extended beltline material.

5. **Questions regarding deficient materials**

   Concern: SLOMFP and FOE expressed concerns that the Diablo Canyon Unit 1 RPV was made from a deficient material that was selected in error.

   References: These SLOMFP and FOE concerns are expressed on pdf page 192 of Citation 1. Citations appear at the end of this Executive Summary.

   Evaluation: The Diablo Canyon Unit 1 RPV has mechanical properties and embrittlement sensitivity comparable to that of other early construction plants. Based on the knowledge of the time no error was made in material selection. Many early plants have RPV materials with high copper contents (higher than Diablo Canyon Unit 1), which is now known to elevate the steel's sensitivity to irradiation embrittlement. This embrittlement has been well characterized and managed, leading to modifications

[6] "PTS" is a postulated severe accident that could, under rare circumstances, cause fracture of the RPV.

of the operating conditions allowed, which supports the safe operation of Diablo Canyon Unit 1.

<u>Citations to SLOMFP and FOE Concerns</u>

1. San Luis Obispo Mothers for Peace and Friends of the Earth petition concerning Diablo Canyon Nuclear Power Plant Unit 1 to the NRC, Docket No, 50-275, 14 September 2023. Specifically, the "Macdonald Declaration" that begins on page 38 of this document.
2. Docket No. R.23-01-007, Supplemental Opening Testimony of Dr. Digby Macdonald on Behalf of San Luis Obispo Mothers for Peace on Phase 1 Track 2 Issues, Public Utilities Commission of the State of California, 11 July 2023.

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# Glossary & Acronyms

| Abbreviation | Definition |
|---|---|
| ASME | American Society of Mechanical Engineers |
| ASTM | American Society for Testing and Materials |
| CFR | Code of Federal Regulations |
| CMM or SRM | Correlation monitor material or standard reference material. These are samples of a common steel that is placed in the surveillance capsules of many plants as a quality control measure. |
| CPUC | California Public Utilities Commission |
| DCISC | Diablo Canyon Independent Safety Committee |
| ETC | Embrittlement trend curve |
| FOE | Friends of the Earth |
| NRC | Nuclear Regulatory Commission |
| RPV | Reactor pressure vessel |
| SLOMFP | San Luis Obispo Mothers for Peace |

| Symbol | Definition |
|---|---|
| CF | Chemistry factor: quantified the radiation sensitivity of RPV steel. Based on either the tables in [RG1.99R2] or a fit to surveillance data |
| EoE | Extent of embrittlement |
| LF | Lead factor: defined in [ASTM E185] as "the ratio of the average neutron fluence of the specimens in a surveillance capsule to the peak neutron fluence of the corresponding material at the ferritic steel reactor pressure vessel inside surface calculated over the same time period." |
| $K_{Ic}$ | A linear-elastic stress intensity factor that characterizes fracture toughness on the lower shelf and in lower transition. |
| $K_{Jc}$ | An elastic-plastic stress intensity factor that characterizes fracture toughness in the transition regime. |
| $T_{41J}$ | The temperature at which the average Charpy energy is 41J (30 ft-lbs) |
| $RT_{NDT}$ | Index temperature that is based on Charpy and nil-ductility temperature tests that is used with the ASME $K_{Ic}$ curve. |
| $T_0$ | Index temperature of the $K_{Jc}$ fracture toughness curve determined according to ASTM testing standard E1921. At $T_0$ the median fracture toughness is 100 MPa$\sqrt{m}$. |
| USE | Average upper shelf energy of the Charpy transition curve |

# 1. Background and Objective

In 2009 the Pacific Gas and Electric Company (PG&E) submitted its plans for license renewal of both units at the Diablo Canyon site [Diablo LRA 2009] with the Nuclear Regulatory Commission (NRC) under 10 CFR Part 54 [10CFR54]. If approved by the NRC a license renewal authorizes a 20-year extension to the current operating license[7]. In 2018 PG&E informed the NRC it wished to withdraw that license renewal application due to the then-projected energy demands and other economic factors in California [DCL-18-015]. The California Public Utilities Commission (CPUC) approved this decision to terminate the license renewal application [CPUC 2018]. However, in 2022 Senate Bill No. 846 was passed in the State of California [C-Senate 2022]. This bill invalidated the decision documented in [CPUC 2018] and directed the CPUC "*to set new retirement dates for the Diablo Canyon powerplant … conditioned upon the United States Nuclear Regulatory Commission extending the powerplant's operating licenses.*" Consequently, in October 2022 PG&E informed the NRC that it wished to re-initiate the license renewal application it withdrew four years earlier [DCL-22-085]. Currently the original 40-year operating licenses for Unit 1 and Unit 2 expire in 2024 and 2025, respectively.

Following the 2022 decision to pursue a license renewal, members of the two organizations, the San Luis Obispo Mothers for Peace (SLOMFP) and the Friends of the Earth (FOE), have placed before the Diablo Canyon Independent Safety Committee (DCISC) their concerns regarding the state of embrittlement[8] in the Unit 1 reactor pressure vessel (RPV) and, consequently, its continued operating safety. Mr. Bruce Severance has also provided commentary and analysis to the DCISC on numerous occasions. SLOMFP and FOE have participated in CPUC rulemaking addressing the extended operation of Diablo Canyon Unit 1

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7 In the USA there is no legal limit to the time over which a nuclear power plant may be licensed to operate; six units have already extended their licenses from 60 to 80 years [NRC Renewals 80] and some discussions have been held concerning 100-year licenses [NRC 2021].

8 Chapter 2 will explain the meaning of the word "embrittlement."

[SLOMFP-CPUC 2023] and have initiated legal proceedings with the NRC [SLOMFP-NRC 2023], expressing their concerns with the embrittlement condition of the Unit 1 RPV.

This report is one of two reports prepared for the DCISC. The purpose of these reports is to address the concerns expressed by SLOMFP and FOE in their presentations before the DCISC and in their legal filings with the CPUC and the NRC. The objectives of this report (Part 1 of 2) are twofold. First, this report will explain the current process for predicting material embrittlement and for establishing operating limits consistent with the requirements of the NRC and of the American Society for Mechanical Engineers (ASME)[9]. Second, this report will address concerns raised by SLOMFP and FOE in two documents [SLOMFP-CPUC 2023] and [SLOMFP-NRC 2023].

The remainder of this report is structured as follows:

- Chapter 2 describes how the embrittlement and fracture toughness of reactor pressure vessel (RPV) materials are measured and forecast. This chapter provides both a conceptual description as well as a summary of current practices in the United States.
- Chapter 3 summarizes the concerns raised by SLOMFP and FOE in [SLOMFP-CPUC 2023] and [SLOMFP-NRC 2023] and provides this consultant's evaluation of those concerns.
- Chapter 4 provides a summary of this report and a list of conclusions.
- Chapter 5 provides a list of documents cited in this report.
- Chapter 6 provides the professional resume of Dr. Mark Kirk, DCISC's consultant and author of this report.

The objective of the companion to this report (Part 2 of 2, [Kirk 2024]) is to independently evaluate the state of knowledge concerning the embrittlement of the Diablo Canyon Unit 1 RPV and to review current safety evaluations of Diablo Canyon Unit 1 as performed by PG&E and submitted to the NRC.

[9] Certain provisions of the ASME Code are incorporated by reference into the NRC's Code of Federal Regulations (CFR), giving such provisions the force of law.

# 2. Forecasting RPV Embrittlement & Fracture Toughness

The RPV provides one of three radiological barriers between the nuclear fuel located in the reactor core and the environment. Because of its critical role to plant and public safety the RPV has, since the beginning of electricity production by means of a nuclear-powered steam supply system, been subject to stringent rules concerning its design and operation. These rules ensure the RPV's integrity throughout the plant's operating lifetime. Among other things, these rules require testing and monitoring of the mechanical properties of the steel from which the RPV is constructed. Assessment of these data and, in some cases, modification of the plant's operating limits in view of these data ensures that the RPV continues to perform its intended function and safety role. This section will describe, first conceptually and then in technical detail, the rules and procedures that ensure the operating safety of a nuclear power plant. These explanations are provided before a response to the concerns expressed by SLOMFP and FOE (see Chapter 3) because many of those concerns originate from what appears to be an incomplete understanding of existing rules and procedures.

An evaluation of the operating safety of a nuclear RPV requires knowledge of the fracture toughness of the materials from which a RPV is made. Fracture toughness is a mechanical property that quantifies the material's ability to withstand loading without fracture *in the presence of a pre-existing flaw*. In the context of a nuclear RPV, embrittlement refers to a reduction of the fracture toughness of the steel plates, forgings, and welds that make up the RPV. Embrittlement is caused by neutron irradiation damage, which occurs at a microstructural level. The level of embrittlement is monitored over the lifetime of the plant using a surveillance program.

Understanding the terms "surveillance program," "fracture toughness," and "embrittlement" is critical to being able to understand an explanation of how the structural integrity of a nuclear RPV is ensured. To that end, this Chapter is structured as follows:

- Section 2.1 describes key terms "surveillance program," "fracture toughness," and "embrittlement" conceptually. A more detailed technical understanding is available in textbooks or technical papers on these subjects, see [Kirk 2018b], [Wallin 2011], and Soneda 2014], respectively.
- Section 2.2 provides a conceptual description of the process used to forecast embrittlement and fracture toughness for RPV steels.
- Section 2.3 provides a conceptual description of how embrittlement forecasts are used to assess RPV operating safety.
- Section 2.4 describes, with some technical detail, the current practice in the USA for embrittlement and fracture toughness forecasting.
- Section 2.5 describes, with some technical detail, the current screening criteria on embrittlement used in the USA. Screening criteria are indicators that define the degree of embrittlement that can be justified by routine analysis and measurements. If screening criteria are exceeded, additional analysis and measurements, or possibly plant modifications, may be needed to ensure continued operating safety.

## 2.1. Key Terms

Before the process for embrittlement and fracture toughness forecasting can be understood three key terms need to be defined: a RPV surveillance program, fracture toughness, and embrittlement. The following sub-sections provide a conceptual understanding of each. A more detailed technical understanding is available in textbooks or technical papers on these subjects, see [Kirk 2018b], [Wallin 2011], and [Soneda 2014], respectively.

### 2.1.1. Surveillance Program

NRC regulations stipulate that if the RPV is projected to exceed a fluence on the inner diameter exceeding $1 \times 10^{17}$ n/cm$^2$ by the end of license it must have a surveillance program to monitor the cumulative effects of neutron irradiation damage on the fracture toughness of RPV steels [10CFR50-AppH], a phenomenon referred to as embrittlement (the terms fracture toughness and embrittlement are defined and discussed in Sections 2.1.2 and 2.1.3, respectively). As required by [10CFR50-AppH], the surveillance program follows American Society for Testing and Materials (ASTM) Standard E185 [ASTM E185-82, Kirk 2018b]. Figure 1 illustrates that the surveillance program includes multiple "capsules." Each capsule contains the mechanical property specimens (sometimes called "coupons") used to measure the Charpy impact toughness, the fracture toughness, and the tensile strength of the materials used to fabricate the central region of the RPV[10] that experiences the highest neutron exposure and, thus, will

[10] The central region of the RPV that receives the highest neutron dose is called the beltline shell. For example, in the Diablo Canyon RPV the beltline shell is fabricated from several large plates that are bent into arcs and welded together. Each shell plate and each weld can, in principle, have a different chemical composition and thus a different sensitivity to damage by neutron irradiation.

become the most embrittled over time. Testing of these specimens follows the requirements of ASTM E23 for Charpy impact toughness and ASTM E8 for strength [ASTM E23, ASTM E8]. Fracture toughness specimens, while included in many capsules, are not mandated by either the NRC or ASTM[11]; oftentimes they are not tested but instead are stored for possible future use. The capsules also include dosimeters that provide experimental data quantifying the total neutron exposure that the capsule receives while in the RPV. Dosimeters are read to determine the total neutron exposure (called "fluence") experienced at the dosimeter location. These data are used along with neutron transport calculations to estimate the variation of fluence at locations in the RPV other than where the dosimeters were located. Measurements and calculations are performed following NRC Regulatory Guide 1.190 [RG 1.190].

The surveillance standard, ASTM E185, requires the capsules to contain samples of the steels from the reactor's beltline shell that exhibit the greatest sensitivity to irradiation damage. These are called the "limiting materials" as they are the most likely to limit the operation of the reactor later in its lifetime. ASTM E185 requires monitoring of the most limiting shell base material and the most limiting weld. These materials are selected at the time of plant design, a selection necessarily based on the then-current understanding of irradiation embrittlement. This understanding has improved over time, sometimes revealing that the limiting materials have changed. If this happens, various options can be pursued to provide continued surveillance of the steels that make up the RPV beltline shell.

As illustrated in Figure 1, the capsules *are placed at locations closer* to the nuclear fuel than the RPV shell being monitored. Consequently, the specimens in the capsules accumulate neutron damage faster than the RPV being monitored. A "lead factor" quantifies this acceleration of damage. The capsule lead factor is a critical part of surveillance program design. Depending on the value of the lead factor ("*E185 states "it is recommended that the surveillance capsule lead factors …be in the range of 1-3*") the surveillance program can provide information on the RPV embrittlement that will occur years, or even decades, in advance. For example, if a capsule with a lead factor of 2.5 is removed after 15 years of operation, the data from that capsule will represent the condition of the vessel after 37.5 (=2.5×15) years of operation, 22.5 years in advance of the vessel reaching that condition itself. This practice ensures that decisions on operational limits and regulatory compliance are informed years in advance,

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[11] ASTM E185-82 states that "fracture toughness test specimens shall be employed to supplement the information from the Charpy V-notch specimens if the surveillance materials are predicted to exhibit marginal properties" and that "If supplemental fracture toughness tests are conducted … the test procedures shall be documented." No guidance is given on the type of fracture toughness test nor on the quantity of fracture toughness specimens to be tested, nor is the term "marginal properties" defined. Evidence from the time during which ASTM E185-82 was written (1982) indicates that "marginal properties" was generally interpreted to mean a material having a Charpy Upper Shelf Energy (USE) predicted to fall below 68J (50 ft-lbs) before the end of the plant's licensed lifetime. As will be explained in Section 2.1.2, the understanding of fracture toughness was much less advanced at the time of plant design (1970s) than it is today, which may have led to this very general guidance.

allowing adequate time for action.  For this reason, it is not uncommon to have long time intervals between surveillance capsule withdrawals.

Figure 2 provides a copy of the capsule withdrawal schedule stipulated in the 1982 version of ASTM Standard E185 [ASTM E185-82].  With limited exceptions that appear in [10CFR50-AppH] this is the withdrawal schedule still required by the NRC.  The three columns in the table show that more capsules are required for vessel materials having a greater sensitivity to neutron irradiation embrittlement.  Having been developed in the late 1970s and early 1980s *this withdrawal schedule does not account for extension beyond the original 40-year license.* NRC requirements for surveillance during license extension appear in [NUREG 1801], which *requires that surveillance programs continue to fulfill the requirements of [ASTM E185-82] through the new end-of-license fluence*.  As it is a report rather than a regulation, [NUREG-1801] makes no explicit requirements for additional capsule testing during license extension. NRC stated in 2022 that it may revise this guidance in the future [SECY-22-0019].

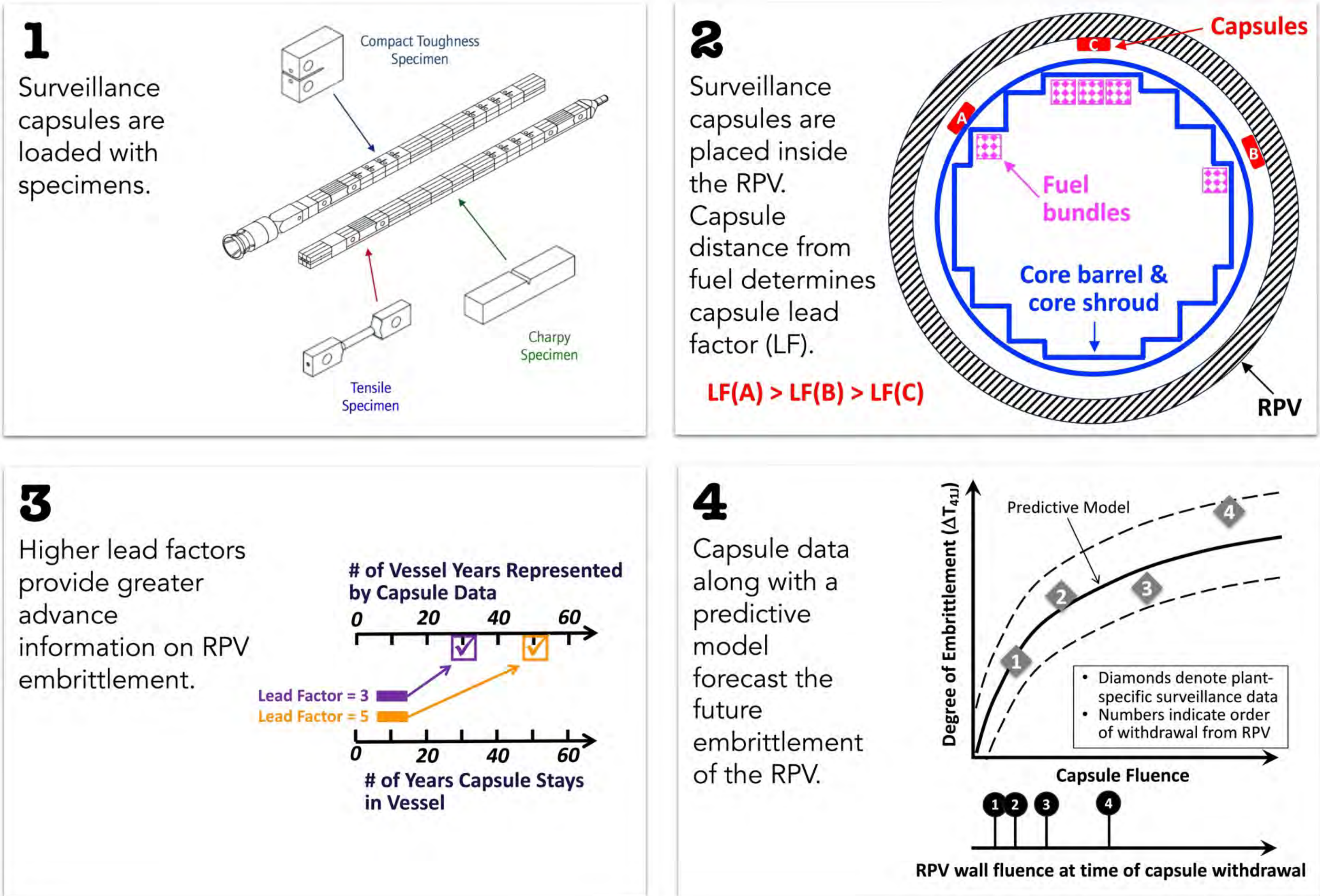


Figure 1. Illustration of the design and conduct of a nuclear RPV surveillance program.

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

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

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

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

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

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

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

Figure 2. ASTM E185-82 capsule withdrawal schedule.

One exception to this plant-specific surveillance approach allowed by [10CFR50-AppH] is an integrated surveillance program, or ISP. *Provided a group of plants meet the NRC's criteria for an ISP [10CFR50-AppH], those plants are allowed to share surveillance data.* An ISP allows reactors of "*similar design and operating features*" to provide data that satisfy each other's [10CFR50-AppH] requirements. Two ISPs have been approved by the NRC and have been operative for over 20 years [BAW-1543, BWRVIP-78, BWRVIP-86]; collectively these programs meet the legal and technical requirements of [10CFR50-AppH] for over 50% of the nuclear reactors now operating in the United States. In one ISP program that includes all boiling water reactors (BWRs), certain materials have been judged sufficiently similar in embrittlement response to other RPV materials needing surveillance as to be designated *representative materials* [BWRVIP-78]. [BWRVIP-78] states that "*the best representative is defined first by the difference between the copper and nickel of the candidate material and the target plant limiting beltline material.*" More recent work inspired by machine learning techniques [Kirk 2022] suggest an approach similar to that of [BWRVIP-78] but using more quantitative similarity criteria than was possible based on the state of knowledge existent when [BWRVIP-78] was published.

## 2.1.2. Fracture Toughness

The "fracture toughness" of a material is a mechanical property whose value quantifies the resistance of a material to fracture (breaking) in the presence of a pre-existing flaw. This information is used along with equations prescribed by the ASME Boiler and Pressure Vessel

Code that are based on an engineering science called "fracture mechanics" to determine the loading conditions under which a nuclear RPV may be safely operated.

A nuclear RPV is made of ferritic steel with a weld-deposited stainless-steel liner. The ferritic steel provides structural resistance to operational and postulated accident loading while the liner provides corrosion protection to the ferritic steel. As illustrated in Figure 3, ferritic steels undergo a ductile to brittle transition. This means that at lower temperatures the material fractures at low values of fracture toughness after a limited amount of plastic deformation[12] while at higher temperatures considerably more load (or energy) needs to be imparted to the material to cause fracture. Terminology used to describe different regimes of a ferritic steel's response to loading also appear on Figure 3. While there is obviously more resistance to fracture at upper shelf temperatures than in fracture mode transition or on the lower shelf, it does not follow that ferritic steel structures can only be operated safely on the upper shelf. With knowledge of the applied loading and the size of flaws that exist in the structure, safe operation of a ferritic steel structure steel can be achieved across the full range of temperatures.

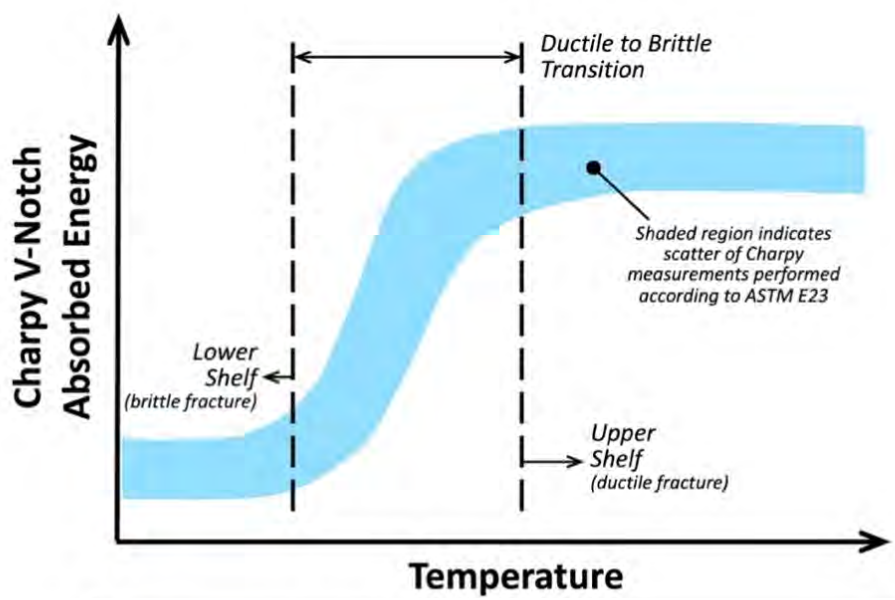


**Figure 3. Schematic illustration of the effect of the ductile to brittle transition of a ferritic steel on toughness properties.**

At the dawn of the nuclear power industry in the United States (1960s-1970s) the understanding of fracture toughness behavior and techniques to measure fracture toughness were not as advanced as they are today. At the time the "$K_{Ic}$" parameter, which applies over the temperature range that includes low transition and lower shelf, could be measured using a

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[12] "Plastic deformation" means that a material is permanently deformed and does not return to the shape it had before loading was applied.

standardized test method [ASTM E399]. However, to meet the validity requirements of that method the test specimens needed to be much larger than could possibly be placed in a surveillance capsule. Since that time, other testing standards[13] have been developed to characterize the fracture toughness of RPV steels over different temperature ranges. At temperatures ranging from the lower shelf through mid-transition [ASTM E1921] can be used. In this standard the measured fracture toughness associated with the brittle fracture of ferritic steels is called $K_{Jc}$ and the standard is used to estimate the fracture toughness transition temperature ($T_0$). On the upper shelf [ASTM E1820] is used to estimate the fracture toughness at the initiation of ductile crack growth ($J_{Ic}$) as well as the variation of resistance to ductile fracture with increasing ductile crack growth (J-R).

To estimate $K_{Ic}$ values indirectly much smaller Charpy V-notch specimens are used in the surveillance program. Even though there are differences between Charpy and fracture toughness specimens both types of data undergo a fracture mode transition with temperature so both tests provide information on the temperature at which this transition occurs. The fracture mode transition temperature for Charpy specimens is defined as the temperature at which the mean Charpy energy is 41J ($T_{41J}$), while the transition temperature for fracture toughness is defined as the temperature at which the mean value of fracture toughness, $K_{Jc}$, is 100 MPa$\sqrt{m}$ ($T_0$). Most importantly, it has been demonstrated for RPV steels that good correlations exist between the Charpy transition temperature ($T_{41J}$) and the fracture toughness transition temperature ($T_0$) [NUREG/CR-6609]. Figure 4 illustrates Charpy toughness data (in light blue) in comparison with fracture toughness data (in light green). These correlations between Charpy and fracture toughness transition temperatures demonstrate that the fracture toughness and embrittlement sensitivity of RPV steels can be reliably estimated using the Charpy data collected via the RPV's surveillance program.

13 The American Society for Testing and Materials (ASTM) develops consensus standards and procedures for the testing and analysis of materials. These include several standards on fracture toughness.

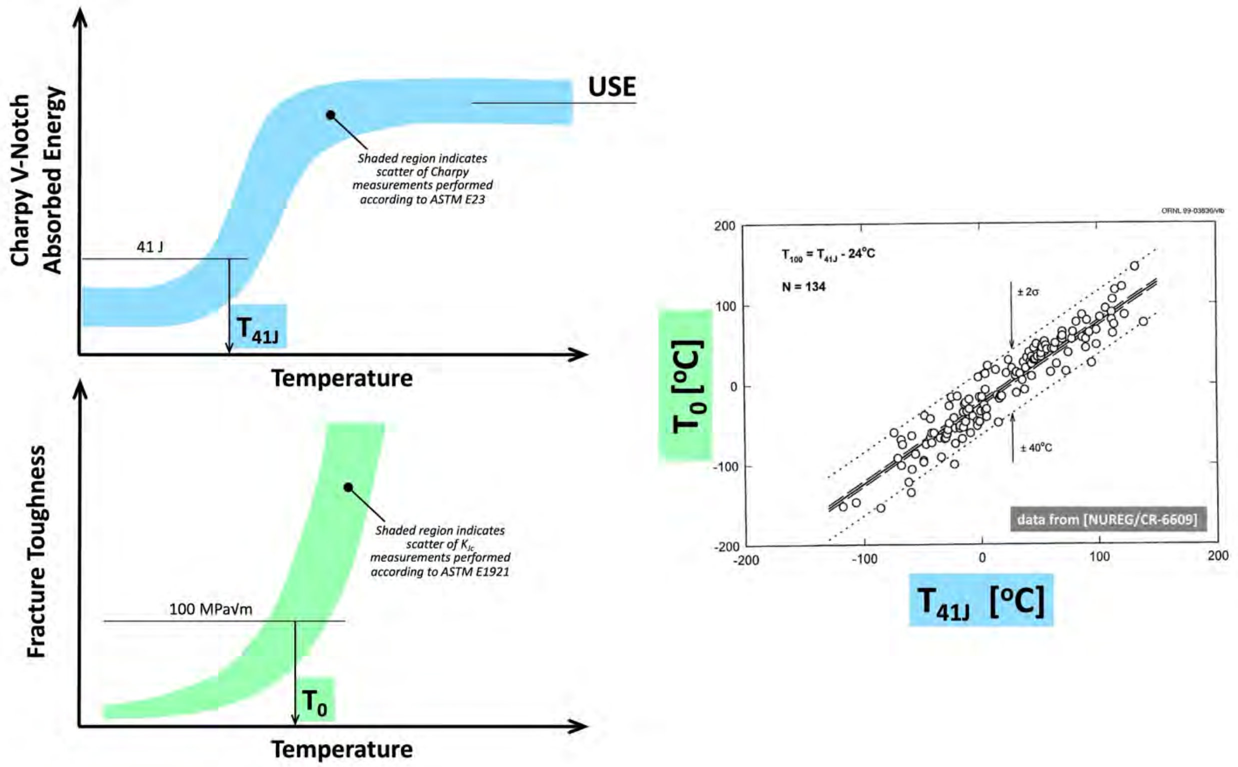


Figure 4. Comparison of Charpy impact toughness and fracture toughness data.

### 2.1.3. Embrittlement

As illustrated in Figure 5, the embrittlement produced by neutron irradiation damage at a microstructural level in a ferritic steel manifests as a reduction in the measured value of fracture toughness or Charpy toughness at any given temperature. For analysis purposes it has been convenient to characterize this reduction as a shift in ductile-to-brittle transition to higher temperatures (i.e., an increase in $T_0$ and $T_{41J}$ for fracture toughness and Charpy impact toughness, respectively) as well as a reduction in the upper-shelf toughness (i.e., a reduction in $J_{Ic}$ and USE for fracture toughness and Charpy impact toughness, respectively). The bottom graph on Figure 5 illustrates schematically how the shift values of transition temperature ($\Delta T_0$ and $\Delta T_{41J}$) change with increasing neutron exposure as plants age.

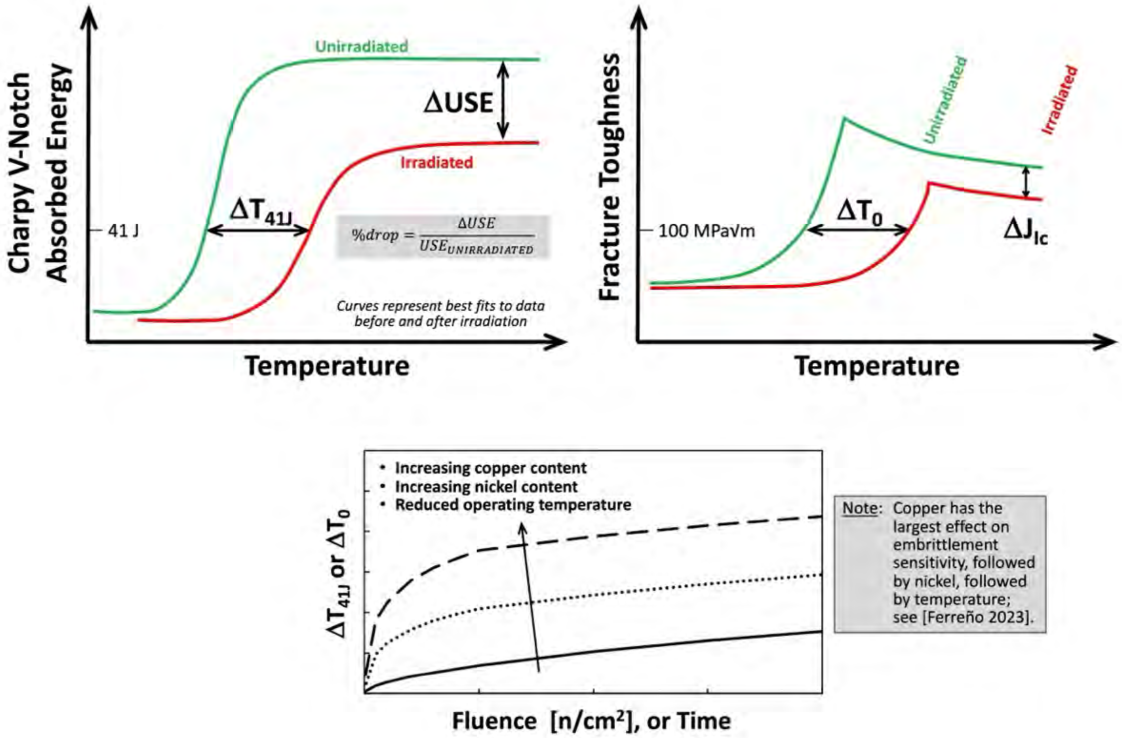


Figure 5. Effect of embrittlement on the fracture toughness and Charpy impact toughness properties of a RPV steel.

## 2.2. Conceptual Description of Embrittlement Forecasting

Embrittlement forecasting relies on two elements:

- The surveillance data for the plant materials that are collected following a [10CFR50-AppH] surveillance program as was described in Section 2.1.1.
- An embrittlement prediction model, typically called an *Embrittlement Trend Curve (ETC).* An ETC is usually based on surveillance program data characterizing the shift in Charpy impact toughness ($\Delta T_{41J}$) caused by irradiation collected following [10CFR50-AppH] surveillance programs from a wide variety of RPV steels.

Surveillance program requirements, as outlined in [10CFR50-AppH] and [ASTM E185-82] require from three to five surveillance capsules for the original 40-year operating license. Due to this limited number of data, it is pragmatic and indeed has become a requirement to compare the plant specific data to the ETC predictions. A quotation from a 1982 NRC memorandum reveals some of the difficulties associated with relying exclusively on plant-specific data [Vagins 1982]:

> *Estimating [$\Delta T_{41J}$] from plant specific surveillance results is very difficult … [because] there is significant scatter in the Charpy data in both the unirradiated and irradiated materials. … Thus, there has developed a preference for using [embrittlement] trend curves developed from a generic database rather than individual, plant specific data.*

Figure 6 illustrates schematically how the plant-specific data collected as part of the surveillance program accumulate over time. The capsule intervals and lead factors in this example conform to the requirements of ASTM E185-82. The figure shows that despite there being long intervals between capsule withdrawals, the capsule data nevertheless represents the level of embrittlement in the reactor vessel many years into the future through appropriate selection of a lead factor. In this illustration the blue curves represent the overall embrittlement trend, which was not known at the time Diablo Canyon Unit 1 entered service. Consequently, this trend was estimated through use of the surveillance data and knowledge of embrittlement trends at the time the surveillance capsules are withdrawn and tested. Specific protocols have been developed in different countries that define when precedence is given to the evidence provided by plant-specific data versus evidence provided by the ETC, and also the conditions under which similar data from other plants may be considered as part of a plant-specific evaluation. Section 2.4 discusses the NRC's approach to these topics.

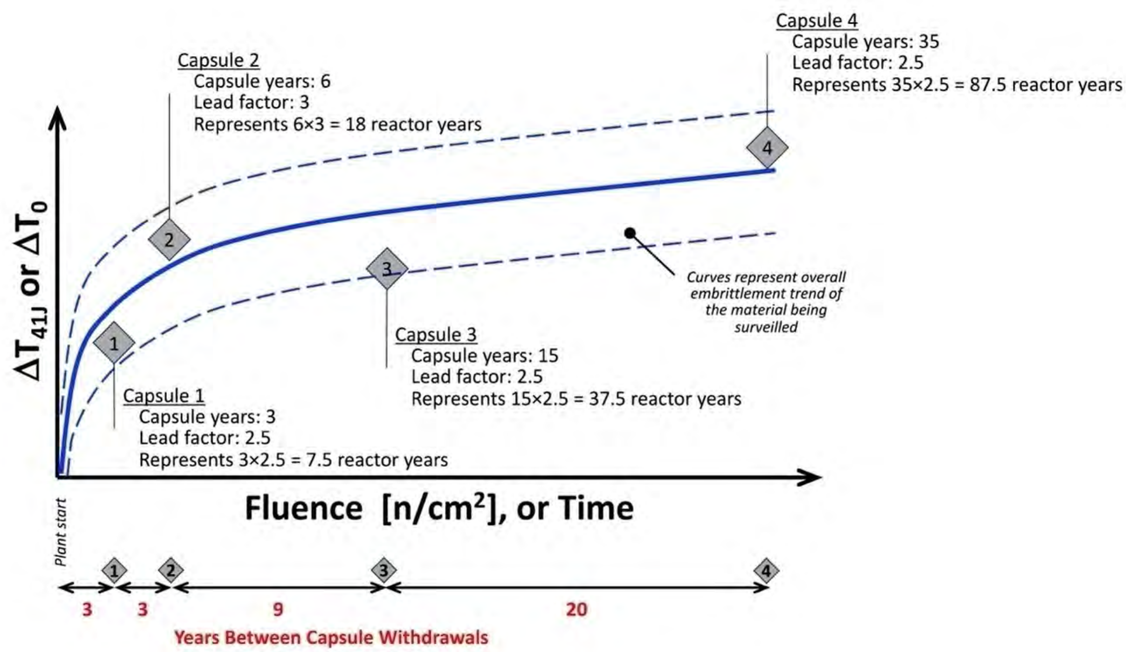


Figure 6. Illustration of how surveillance data accumulates over time.

## 2.3. Conceptual Description of How Embrittlement Forecasts are used to Assess RPV Operating Safety

The schematic diagrams in Figure 7 compare the material fracture toughness available before and after irradiation damage to the amount of force applied to the RPV material by the combined pressure and temperature conditions caused when the RPV is cooled down from its operating condition for an outage. The left-hand figure illustrates the fracture toughness available before irradiation damage occurs and shows that the regulatory screening criteria ensure that the level of force applied by operation of the RPV does not exceed the available fracture toughness. The right-hand figure shows how the fracture toughness curve is shifted by the embrittlement predictions to estimate the fracture toughness after some period of time. The right-hand figure shows that even though embrittlement reduces the allowable region for operation, the fracture toughness of the RPV steel exceeds the level of force imposed by the operating conditions. Figure 8 further illustrates that the level of force imposed by the RPV operating conditions is different during a routine cool down than during a postulated pressurized thermal shock (PTS) accident. Nevertheless, the regulatory screening limits ensure that fracture toughness of the RPV steel exceeds the level of force imposed on the RPV by operating conditions for both scenarios.

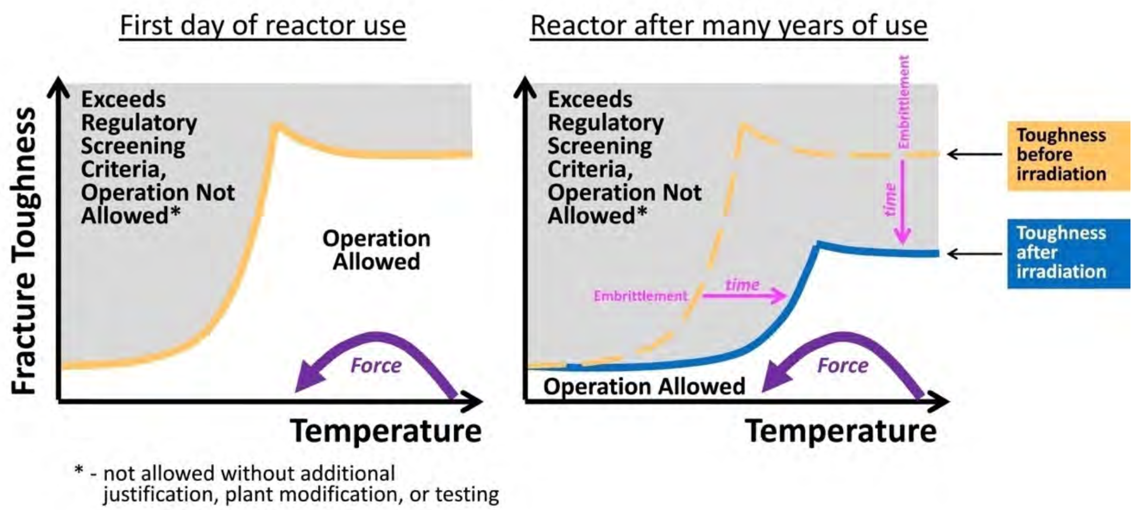


Figure 7. Illustration of how embrittlement forecasts are used to predict fracture toughness, and how fracture toughness compares to the force applied on the RPV structure by the operating temperature and pressure conditions.

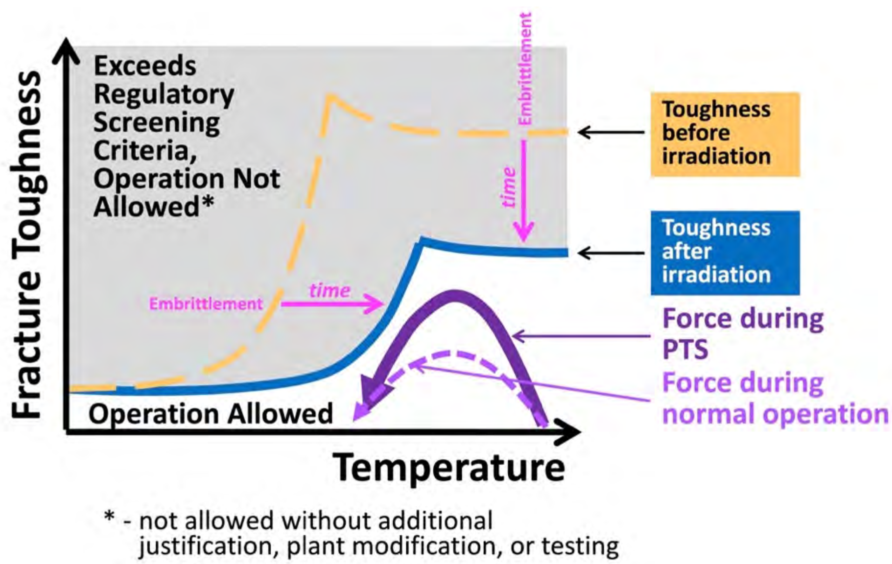


Figure 8. Illustration of how the amount of force applied on the RPV structure differs between normal operations and accident (PTS) conditions, and how these force levels compare to the fracture toughness.

## 2.4. Current Practice for Embrittlement Forecasting in the United States

### 2.4.1. Introduction

Procedures for embrittlement forecasting can be found in three NRC regulatory documents: Revision 2 of Regulatory Guide 1.99 [RG1.99R2], the original PTS rule [10CFR50.61], and the alternate PTS rule [10CFR50.61a]. The requirements of [RG1.99R2] and [10CFR50.61] are identical and are supplemented by example cases presented by the NRC in 1998 [Wichman 1998]. This section describes the combined requirements and guidelines of these documents. Other procedures are available for embrittlement forecasting, including the alternate PTS rule [10CFR50.61a] and recent work by ASTM and ASME. *However, these approaches have not been used by Diablo Canyon Unit 1*, so they are addressed in other parts of this report[14,15].

[RG1.99R2] and [10CFR50.61] establish procedures that use both an ETC and plant-specific surveillance data to forecast the effect of irradiation damage on $\Delta T_{41J}$ and on $\Delta$USE for future conditions. $\Delta T_{41J}$ procedures are identical in both documents while $\Delta$USE procedures appear only in [RG1.99R2]. An estimate of $\Delta T_{41J}$ is needed to assess compliance with the screening criteria of the PTS rule (see Section 2.5.1) and to establish pressure-temperature (P-T) limits for reactor operation (see Section 2.5.3.2). $\Delta$USE is needed to assess compliance with the USE requirements of [10CFR50-AppG] (see Section 2.5.3.1). The $\Delta T_{41J}$ and $\Delta$USE procedures are discussed in the Sections 2.4.2 and 2.4.3, respectively. Section 2.4.4 discusses NRC requirements on what data should be considered as part of these procedures.

### 2.4.2. $\Delta T_{41J}$ Procedure

As described in the following paragraphs, the procedure includes two parts: the ETC and the process to account for plant-specific surveillance data.

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[14] The screening criteria of the original and alternate PTS rules will be compared in Section 2.5 of this report because these differences has been a cause of public concern expressed by the SLOMFP and FOE.

[15] ASTM adopted new procedures for embrittlement forecasting in 2015 [ASTM E900-15, ASTM Adjunct, Kirk 2018a]. The NRC has indicated a favorable attitude towards the ASTM E900-15ETC but has not yet endorsed it [Widrevitz 2019]. The ASTM E900 ETC is now being integrated into an ASME Code Case that was recently balloted in Section XI of the ASME Boiler and Pressure Vessel Code [MRP-462]. A discussion of the procedures of ASTM E900 and the ASME Code Case can be found in Part 2 of this evaluation; Chapter 4 of that report includes a supplementary analysis of Diablo Canyon Unit 1 using these newer methods.

### 2.4.2.1. Embrittlement Trend Curve

The ETC adopted by both [RG1.99R2] and [10CFR50.61] for $T_{41J}$ is described in [Randall 1986]. That ETC estimates $\Delta T_{41J}$ from exposure, composition, and categorical variables as follows:

$$\Delta T_{41J} = \frac{5}{9}(CF) \times f^{(0.28-0.1log(f))} \qquad (2\text{-}1)$$

The $\frac{5}{9}$ factor does not appear in the NRC equations, but is used here to express $\Delta T_{41J}$ in °C. The variables in equation (2-1) are as follows:

*CF* is a "chemistry factor" that characterizes the radiation sensitivity of the steel. *CF* depends on copper content, nickel content, and product form. [RG1.99R2] includes two tables of *CF* values, one for weld material and one for base material. [Randall 1986] describes an algorithm from which the values in the *CF* tables can be calculated.

*f* is the fast neutron fluence in neutrons per square centimeter (E > 1Mev) at the RPV base material inner diameter divided by $10^{19}$ (thus, $f$=1 at a fluence of $1\times10^{19}$ n/cm$^2$).

The uncertainty in the estimate of $\Delta T_{41J}$ calculated using equation (2-1) is called $\sigma_\Delta$, which is the standard deviation of residuals reported for equation (2-1) in [Randall 1986]. $\sigma_\Delta$ has a value of 15.6 °C for welds and 9.4 °C for base materials.

### 2.4.2.2. Process to Account for Surveillance Data

Collectively [RG1.99R2], [10CFR50.61], and [Wichman 1998] describe the following process, which must[16] be followed for PTS evaluations, which are the postulated transients that present the greatest challenge to reactor vessel integrity. The process relies on an assessment of the "credibility" of the data. As will be explained later in this section, the credibility assessment affects the way surveillance data are analyzed and used to forecast the future embrittlement of the RPV. The NRC outlines the following five criteria to assess credibility (the following words are taken from [10CFR50.61], the words in [RG1.99R2] are similar), all of which must be met for a surveillance data set to be considered credible:

Criteria (A) The materials in the surveillance capsules must be those which are the controlling materials with regard to radiation embrittlement.

[16] The process must be followed when performing a PTS evaluation because it is required by the language of [10CFR50.61]. The process is not a requirement for P-T (Pressure-Temperature) limits estimated following [10CFR50-App-G], which incorporates [ASME SC-XI-App G] by reference; however standard practice is to follow the same process to maintain consistency with the PTS assessment.

Criteria (B) Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions must be small enough to permit the determination of the 30-foot-pound (41J) temperature unambiguously.

Criteria (C) Where there are two or more sets of surveillance data from one reactor, the scatter of [$\Delta T_{41J}$] values must be less than 28 °F (15.6 °C) for welds and 17 °F (9.4 °C) for base metal. Even if the range in the capsule fluences is large (two or more orders of magnitude), the scatter may not exceed twice those values.

Criteria (D) The irradiation temperature of the Charpy specimens in the capsule must equal the vessel wall temperature at the cladding/base metal interface within ±25 °F (±13.9 °C).

Criteria (E) The surveillance data for the correlation monitor material in the capsule, if present, must fall within the scatter band of the data base for the material.

**Criteria (B)**, **Criteria (D)**, and **Criteria (E)** are typically met and so are not further discussed here.

Some early construction plants experienced difficulty meeting **Criteria (A)** because they did not have their "controlling" (often called "limiting") materials in their surveillance program due to the limited guidance and/or technical understanding of embrittlement damage mechanisms that existed when the plant was fabricated. These plants will fail Criteria (A) unless one of the following occurs:

1. Similar data for the limiting materials are located as part of another plant's surveillance program (see Section 2.4.4.1).
2. Samples of the limiting materials are loaded into supplemental capsules, which are placed into the vessel, irradiated, and tested.
3. Samples of the limiting materials are loaded into capsules, which are placed into a test reactor, irradiated, and tested.

The first solution is the most common, the second has been pursued by some plants, while the third has not been used.

**Criteria (C)** can be described as follows:

- Criteria (C) assesses whether plant-specific surveillance data follow the embrittlement trends expected by equation (2-1). If the data don't follow these trends, they will fail Criteria (C) and the data are classified as not credible.
- Since Criteria (C) requires "two or more data sets" the first surveillance capsule withdrawn from a reactor is never considered credible. The implication of this requirement combined with the capsule withdrawal schedule (see Figure 2) is that embrittlement sensitive plants will be in-service for at least three years before their surveillance data may be considered credible and influence their embrittlement forecast. Likewise, plants less sensitive to embrittlement may be in service for as much as 15 years before their surveillance data can influence their embrittlement forecast.
- As illustrated in Figure 9, examples provided in [Wichman 1998] make clear the following:

- Before assessing Criteria (C) the CF in equation (2-1) is adjusted so that equation (2-1) provides a best fit to the two or more available $\Delta T_{41J}$ data.
- If data from other plants are used (see Section 2.4.4) the measured $\Delta T_{41J}$ data from those plants may require adjustment to account for the effects of temperature and composition differences between the other plant data and the plant of interest.
- The word "scatter" as used in Criteria (C) denotes the residual between the best-fit form of equation (2-1) and each (possibly adjusted) measurement of $\Delta T_{41J}$.

The statement of Criteria (C) in [10CFR50.61] and [RG1.99R2] is not explicit regarding the number of $\Delta T_{41J}$ data that may lie outside the scatter limits expressed by the criteria. For example, if only one $\Delta T_{41J}$ measurement out of a set of ten were to lie outside of the permissible scatter range of ±9.4 °C for base metals and ±15.6 °C for welds would this be sufficient to make the data set not credible? Examples given in [Wichman 1998] attempted to clarify this position. [Wichman 1998] provides several examples showing that data sets having ≥33% of $\Delta T_{41J}$ data outside the permissible scatter range are considered not credible. [Wichman 1998] also gives an example where 17% of $\Delta T_{41J}$ values lie outside the permissible scatter range and states such a data set "*could be considered credible.*" In the past 10-15 years credibility assessments performed in reports on surveillance capsules tested by the Westinghouse Electric Company have further clarified this criterion. These reports observe that the NRC's permissible scatter range corresponds to ±1σ; as such, 32% of data are expected to fall outside of this range. Thus, surveillance data sets having more than 32% of data outside the permissible scatter range are considered not credible (this is also illustrated in Figure 9). The NRC has not objected to this interpretation of Criteria (C).

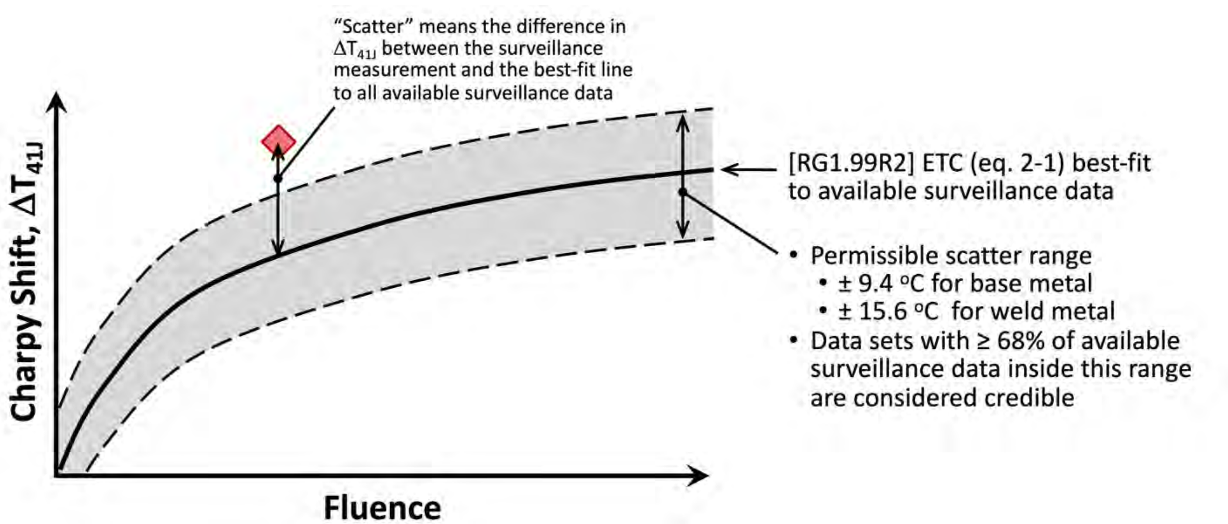


Figure 9. Illustration of credibility Criteria (C).

Having established the credibility of the available data for the limiting materials following the process just described, the NRC stipulates different procedures to account for surveillance data in the $\Delta T_{41J}$ forecast, one for not-credible data and one for credible data (see Figure 10). Key features of the NRC's approach are as follows (the red squares Figure 10 correspond to the letters below):

A. Consideration of data from materials irradiated in the surveillance programs of other plants that are judged to be similar to the materials in the plant of interest is required. Section 2.4.4 describes how similar materials are defined.
B. Small differences in exposure temperature and/or chemical composition between the plant material of interest and similar data from other plants are accounted for by adjusting the measured $\Delta T_{41J}$ values from other plants. The chemistry adjustment is performed using equation (2-1) to represent the effect of both copper and nickel on $\Delta T_{41J}$.
C. When data are judged to be not credible *an intentionally conservative approach* to embrittlement forecasting is adopted. Specifically, *a larger margin term* is used to account for material uncertainty and the highest estimate of embrittlement rate, as quantified by the term CF in equation (2-1), is adopted. It may be noted that the not-credible surveillance data are considered in estimating the value CF (see the box on the right-hand side of the diagram containing the equation CF = MAX ($CF_{TABLE}$, $CF_{DATA}$)). Following the procedure of Figure 10 will *always produce a more conservative* (that is: higher) estimate of the upper-bound value of $\Delta T_{41J}$ for not-credible data than for credible data.
D. When data are judged to be credible the term to account for uncertainty in the $\Delta T_{41J}$ measurements is reduced by a factor of two, reflecting the increased confidence in the understanding of embrittlement trends as supported by the credible data.

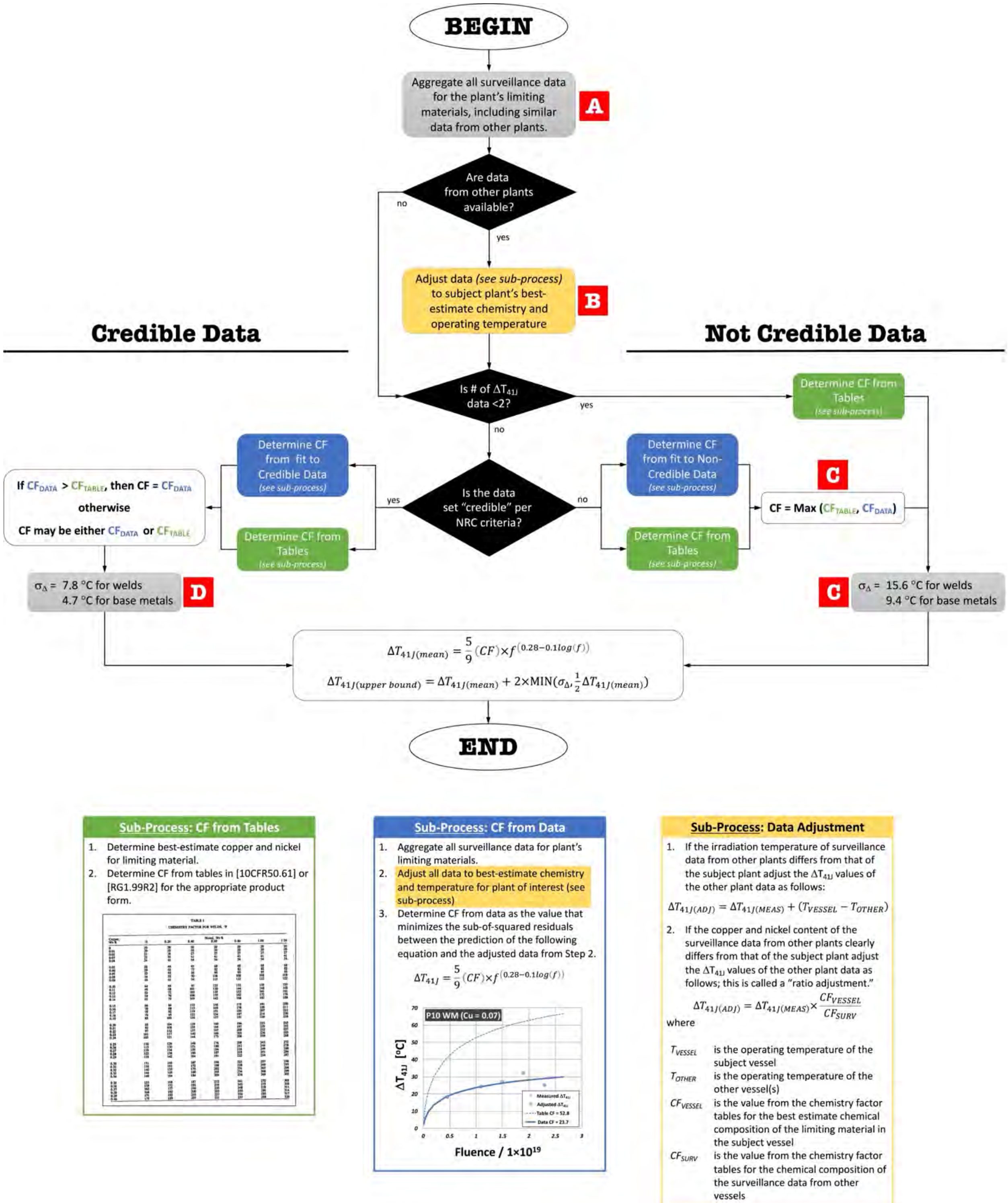


Figure 10. Diagram showing $\Delta T_{41J}$ forecasting procedures for both credible and not-credible data as required by [10CFR50.61] and as described by [RG1.99R2] and [Wichman 1998].

## 2.4.3. ΔUSE Procedures

The procedure used to calculate the change in USE includes two parts: the ETC and the process for adjusting the ETC to account for surveillance data. These are described in the following sections.

### 2.4.3.1. Embrittlement Trend Curve

The value %drop, which is unitless, quantifies the amount by which the Charpy USE decreases from its unirradiated value due to irradiation (see Figure 5). [RG1.99R2] provides the graphical relationship depicted in Figure 11, which relates %drop to fluence, copper, and product form. [RG1.162] provides a formula for the relationship depicted in Figure 11 based on information originally appearing in [Cheverton 1992]:

$$\%drop = \mathrm{MIN}\,(\mathrm{A}, \mathrm{B}) \tag{2-2}$$

$$A = (100Cu + PF + \alpha)f^{0.2368} \tag{2-3}$$

$$B = 42.39f^{0.1502} \tag{2-4}$$

Here Cu is in weight percent, PF is 9 for base metals and 14 for weld metals, $\alpha$=0, and f is fluence divided by $10^{19}$.

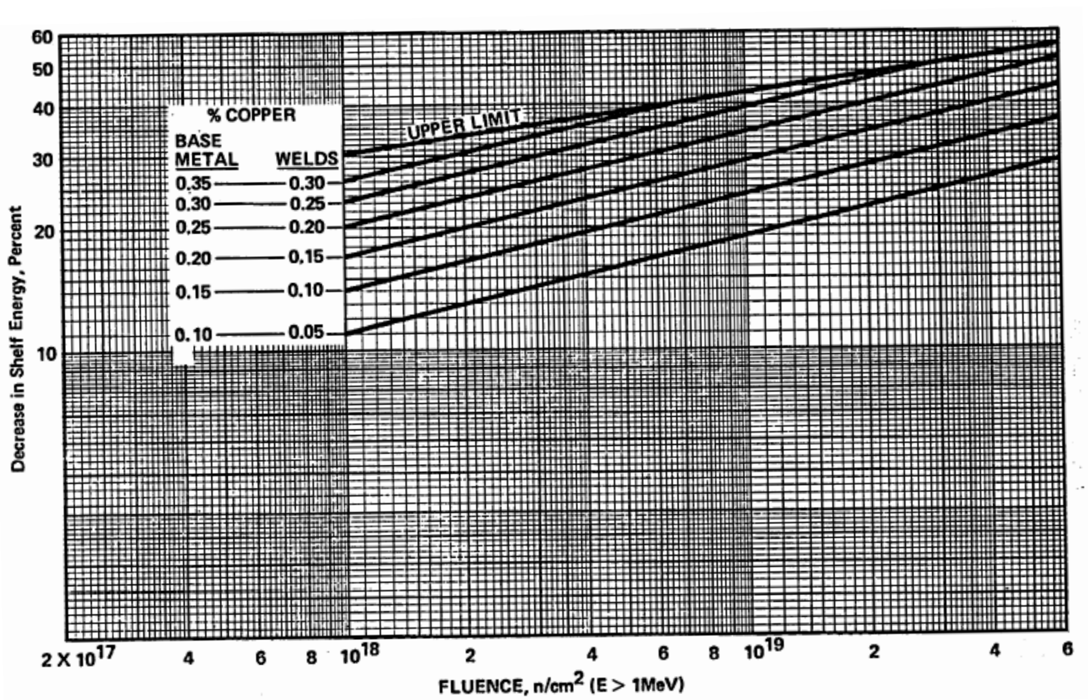


Figure 11. Graphical relation for upper-shelf energy drop from RG1.99R2.

The value ΔUSE, also shown in Figure 5, can be calculated as follows:

$$\Delta USE = USE_U - USE_I = \%drop \times USE_U \qquad (2\text{-}5)$$

The values $USE_U$ and $USE_I$ signify the values of USE measured before and after irradiation, respectively.

Unlike $\Delta T_{41J}$ there is no explicit uncertainty treatment associated with the %drop or ΔUSE estimates. This approach is as described by [RG1.99R2] and is accepted by the NRC. Information published by the NRC, [Widrevitz 2019] indicates that equations (2-2) to (2-4) over-estimate the measured ΔUSE data roughly 80% of the time (see Figure 12). As will be explained in Section 2.5.3.1 the predictions of equations (2-2) to (2-4) may trigger a more detailed evaluation if USE is predicted to fall below 68 J. Such evaluations invariably demonstrate that adequate structural integrity is maintained to USE levels significantly below 68 J.

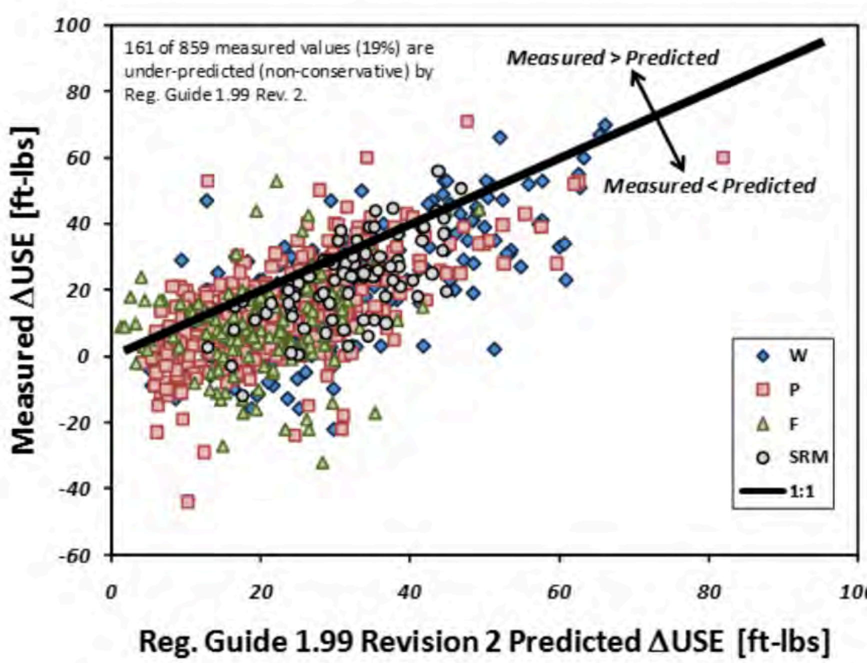


**Figure 12. Comparison of the [RG1.99R2] prediction of ΔUSE to measured data, graph from [Widrevitz 2019]. In the legend W=weld, P=plate, F=forging, and SRM=-standard reference material.**

### 2.4.3.2. Process to Account for Surveillance Data

[RG1.99R2] describes the following process to use when estimating %drop, which is needed to assess compliance with the USE screening criteria of [10CFR50-AppG] described in Section 2.5.3.1. As was the case for $\Delta T_{41J}$ estimation (see Section 2.4.2.2), the process relies on an assessment of the "credibility" of the data. The NRC outlines five criteria to assess credibility.

These are the same as described in Section 2.4.2.2 for $\Delta T_{41J}$, with the following additional statement added to Criteria (C):

> *Even if the data fail [Criteria (C)] for use in shift calculations, they may be credible for determining decreases in upper-shelf energy if the upper shelf can be clearly determined, following the guidelines given in ASTM E185-82.*

In practice, few if any plants have ever failed this criterion. Thus, for all practical purposes if there are two or more measurements of %drop the NRC considers those data to be credible.

For not-credible data, [RG1.99R2] directs that equations (2-2) through (2-4) be used to estimate %drop. For credible data [RG1.99R2] directs that the value $\alpha$ in equation (2-2) be adjusted until the prediction of %drop over-estimates all available data ($\alpha$ may be positive or negative). Figure 13 depicts this process. Thus, both credible and not credible data can be used to assess the % drop in Upper Shelf Energy, with different conservativisms applied as illustrated in Figure 13.

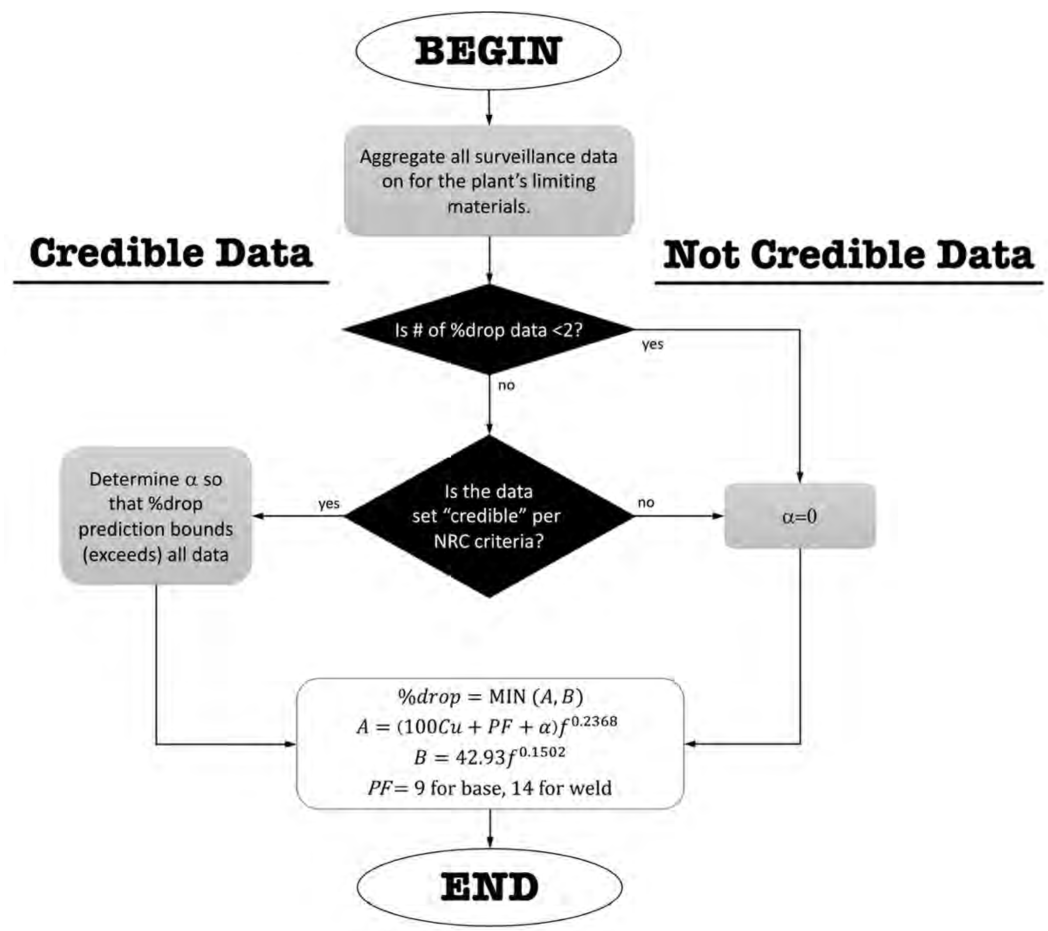


Figure 13. Diagram showing %drop forecasting procedures for both credible and not-credible data as described by [RG1.99R2].

### 2.4.4. Data to Use

#### 2.4.4.1. To estimate $\Delta T_{41J}$

For $\Delta T_{41J}$, the NRC has long recognized that surveillance data that are by some measure similar to the materials used to fabricate the RPV in the plant of interest but are obtained from the surveillance programs of other plants may provide valuable information. [10CFR50.61] contains the following statement (emphasis added underlines added for clarity):

> *"To verify that [the embrittlement forecast] for each vessel beltline material is a bounding value for the specific reactor vessel, licensees shall consider plant-specific information that could affect the level of embrittlement. This information includes but is not limited to the reactor vessel operating temperature and any related surveillance program results. Surveillance program results means any data that demonstrated the embrittlement trends for the limiting beltline material, including but not limited to data from test reactors or from surveillance programs at other plants with or without surveillance program integrated per 10 CFR Part 50 Appendix H."*

These words make clear that the NRC requires the analysis of surveillance data performed in support of any particular plant to consider data obtained from the surveillance programs of other plants. However, these words do not describe a process for determining which data from other plants should be considered and which should not. Information on this process can be inferred from the example cases presented by the NRC in 1998 [Wichman 1998]. [Wichman 1998] provides an example (Case 4) that considers "*surveillance data from plant* ***and*** *other sources*" where plants containing the same weld wire heat are the "other sources" of data. A 1993 EPRI report introduced the term "sister plants" to describe different plants having welds made from the same heat of material [EPRI 1993]. A few sister plants share a common heat of base material, but this is not common owing to the large size of plates and forgings needed for RPV construction.

Another example in [Wichman 1998] (Case 5) describes methods that consider "*surveillance data from other sources* ***only***," suggesting that a forecast of embrittlement can be constructed entirely from data other than that obtained via the surveillance program of the plant of interest. In combination, Cases 4 and 5 in [Wichman 1998] indicate that the NRC requires licensees to consider similar data from other plants as part of surveillance data analysis, and that when such analyses are performed the NRC accepts such similar "sister plant" data as being equivalent to data obtained from the subject plant's [10CFR50-AppH] mandated surveillance program. The wording from the [10CFR50.61] quotation earlier in this section (i.e., "licensees *shall* consider") make this consideration a requirement. While the NRC has never formalized [Wichman 1998], the industry has adopted it as *de facto* guidance for use in $\Delta T_{41J}$ embrittlement forecasting; the NRC has never objected to this usage.

The other aspect of the process for determining which $\Delta T_{41J}$data from other plants should be considered can only be inferred from the experience of the industry implementing, and the NRC reviewing, these practices over the last quarter century. During this time data from other plants has always been taken from the same reactor type, that is: data from pressurized water reactors (PWRs) has only been considered with other PWR data while data from boiling water reactors (BWRs) has only been considered with other BWR data.

In summary, the NRC's requirement from [10CFR50.61] that data from other plants be considered when analyzing $\Delta T_{41J}$ surveillance data from a plant of interest can be stated as follows:

> **Plants must consider all $\Delta T_{41J}$ surveillance data that is available for the same weld wire heat as was used for the welds in their RPV beltline. These data should come from a reactor of similar design (i.e., PWRs should consider PWR data only, BWRs should consider BWR data only).**

#### 2.4.4.2. To Estimate ΔUSE

For ΔUSE, the data considered is typically that from the plant specific surveillance program. This differs from the situation for $\Delta T_{41J}$ data, where [10CFR50.61] explicitly requires the consideration of similar data from other ("sister") plants. In [10CFR50-AppG] where the 68J screening limit on USE is established the NRC makes no statements of this kind. Thus, the NRC's guidance concerning ΔUSE can be summarized as follows:

> **Plants typically consider only ΔUSE surveillance data obtained from the plant in question.**

## 2.5. Current Embrittlement Screening Criteria Used in the United States

Screening criteria and other limitations on embrittlement to ensure the safe operability of a nuclear RPV appear in the original PTS rule [10CFR50.61], the alternate PTS rule [10CFR50.61a], and in the NRC's Rule entitled "*Fracture Toughness Requirements*" [10CFR50-App G]. This section summarizes the requirements of all three documents.

### 2.5.1. Original PTS Rule [10CFR50.61]

The original PTS rule, which was first published in the mid 1980s imposes a screening-criteria on the amount of embrittlement permitted before additional action is required. These criteria are quantified as a maximum value of a fracture transition temperature which is called $RT_{PTS}$. $RT_{PTS}$ locates the fracture toughness data for the reactor materials on the temperature axis, as such it provides a similar function to the $T_0$ metric which was illustrated in Figure 4 and Figure 5. The $RT_{PTS}$ screening criteria quantifies the minimum fracture toughness allowed without additional analysis. This value was determined in [SECY-82-465] to limit the yearly probability of RPV failure to a value lower than $5\times10^{-6}$ events per reactor operating year. The [10CFR50.61] $RT_{PTS}$ screening criteria are 270 °F (132 °C) for axial welds, plates, and forgings and 300 °F (149 °C) for circumferential welds. The higher value for circumferential welds is possible due to the smaller effect of internal pressure on the cracks that would exist in circumferentially oriented welds.

[10CFR50.61] explains that if the screening criteria cannot be satisfied or are projected to not be satisfied in the future, other alternatives may be pursued to demonstrate the continued operating safety of the plant in question. These alternatives include the following:

- Flux reduction to limit the number of neutrons escaping the core and causing embrittlement of the RPV. If implemented early in life flux reduction can keep plant values below the $RT_{PTS}$ screening criteria.
- Plant operational modifications (e.g., heating the cooling water) that lessen the severity or likelihood of PTS. As described by the NRC "*the licensee shall submit a safety analysis to determine what, if any, modifications to equipment, systems, and operation are necessary to prevent potential failure of the reactor vessel as a result of postulated PTS events if continued operation beyond the screening criterion is allowed. In the analysis, the licensee may determine the properties of the reactor vessel materials based on available information, research results, and plant surveillance data, and may use probabilistic fracture mechanics techniques.*"
- Thermal annealing of the vessel following the requirements of [10CFR50.66] to recover the fracture toughness of the vessel beltline materials.

Finally, paragraph (b) of [10CFR50.61] makes clear that demonstrating compliance with the alternate PTS rule [10CFR50.61a] is an acceptable means of demonstrating compliance with [10CFR50.61]. The alternate PTS rule is described in the next section.

## 2.5.2. Alternate PTS Rule [10CFR50.61a]

At the time the original PTS rule was adopted there was little thought of license extension beyond 40-years. Consequently, the many conservativisms inherent to the original rule associated with the limited state knowledge, data, and calculational capabilities of the time seemed acceptable. By the mid 1990s that situation had changed as many plants sought to extend their operating licenses to 60 years under [10CFR54]. These extended operations would cause several PWRs to exceed the screening criteria of [10CFR50.61]. As described in Section 2.5.1 each plant that exceeded the screening criteria could have taken its own compensatory measures to remain compliant with [10CFR50.61]. However, the NRC recognized that the conservatism of the original rule created unnecessary burden. To improve the NRC's efficiency and uniformity in processing license extension requests, the NRC therefore undertook the PTS re-evaluation project [NUREG-1806, NUREG-1874, Kirk 2013].

This PTS re-evaluation project was conducted between 1998 and 2009 by the NRC, with assistance and data provided by the commercial nuclear power industry operating under the auspices of the Electric Power Research Institute (EPRI). The project included a comprehensive evaluation of all the sub-models and inputs needed to estimate the risk of vessel fracture caused by PTS, a consideration of the materials data and operating experience obtained since adoption of the original rule, and a use of modern computational tools. The project focused on use of realistic input values and models coupled with explicit treatment of uncertainties where practicable. Even though the tolerable risk of vessel failure had been reduced by a factor of five since the original project ($5\times10^{-6}$ for [10CFR50.61], to $1\times10^{-6}$ for [10CFR50.61a]), the PTS re-evaluation project demonstrated that the reference temperature screening criteria could nevertheless be increased relative to the [10CFR50.61] limits (see Figure 14). Thus, while compliance with the original rule has been judged by the NRC as adequate to maintain safety, the alternate rule is more conservative owing to the lower tolerable risk of vessel failure used to establish the reference temperature screening limits.

This increase in the reference temperature screening limits was made possible by the more accurate modeling adopted in the PTS re-evaluation project, especially with regards to the flaw distribution, material fracture toughness models, initiating event frequency, and transient characterization. To meet the reference temperature criteria, plants wishing to employ [10CFR50.61a] must also analyze their surveillance data using a process of similar intent but different detail to that of [10CFR50.61]. Plants must also perform a non-destructive examination of their beltline welds and use data on flaw size from that examination to demonstrate that the flaw model assumed in the probabilistic calculations on which the [10CFR50.61a] screening criteria were based is appropriate for the plant in question. The Palisades plant in Michigan was granted a license amendment to implement [10CFR50.61a] in 2015 [NRC 2015b].

As is the case with [10CFR50.61], an inability to comply with the provisions of [10CFR50.61a] does not mean the plant must shut down provided appropriate compensatory actions are taken. These include neutron flux reduction, plant modifications designed to reduce PTS event probability or severity, reactor vessel annealing, or performing a probabilistic analysis including refined models and plant-specific information. The Diablo Canyon Unit 1 reactor pressure vessel is 8.625 inches thick in the beltline region, so if PG&E were to use [10CFR50.61a] in the future the screening criteria in the column headed $T_{WALL} \leq$ 9.5-in. would apply.

TABLE 1—PTS SCREENING CRITERIA

| Product form and $RT_{MAX-X}$ Values | $RT_{MAX-X}$ limits [°F] for different vessel wall thicknesses [6] ($T_{WALL}$) | | |
|---|---|---|---|
| | $T_{WALL} \leq$ 9.5 in. | 9.5 in. < $T_{WALL} \leq$ 10.5 in. | 10.5 in. < $T_{WALL} \leq$ 11.5 in. |
| Axial Weld $RT_{MAX-AW}$ | 269 | 230 | 222 |
| Plate $RT_{MAX-PL}$ | 356 | 305 | 293 |
| Forging without underclad cracks $RT_{MAX-FO}$ [7] | 356 | 305 | 293 |
| Axial Weld and Plate $RT_{MAX-AW}$ + $RT_{MAX-PL}$ | 538 | 476 | 445 |
| Circumferential Weld $RT_{MAX-CW}$ [8] | 312 | 277 | 269 |
| Forging with underclad cracks $RT_{MAX-FO}$ [9] | 246 | 241 | 239 |

[6] Wall thickness is the beltline wall thickness including the clad thickness.

[7] Forgings without underclad cracks apply to forgings for which no underclad cracks have been detected and that were fabricated in accordance with Regulatory Guide 1.43.

[8] $RT_{PTS}$ limits contribute $1\times10^{-8}$ per reactor year to the reactor vessel TWCF.

[9] Forgings with underclad cracks apply to forgings that have detected underclad cracking or were not fabricated in accordance with Regulatory Guide 1.43.

Figure 14. Reference temperature screening criteria from the alternate PTS rule [10CFR50.61a].

## 2.5.3. NRC Fracture Toughness Requirements [10CFR50-AppG]

[10CFR50-App G] states its purpose as follows:

> *This appendix specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime.*

To achieve this goal [10CFR50-App G] imposes requirements on the Charpy USE and also requirements on how pressure-temperature (P-T) operating limits are established. The following sub-sections explain these requirements.

### 2.5.3.1. Upper Shelf Energy Requirements

Earlier in this report, Figure 5 illustrated the effect of embrittlement on the Charpy impact curve and showed that increasing embrittlement produces a reduction in the Charpy USE. [10CFR50-App G] establishes the following screening criteria on USE:

> *Reactor vessel beltline materials must have Charpy upper-shelf energy in the transverse direction for base material and along the weld for weld material according to the ASME Code, of no less than 75 ft-lb (102 J) initially and must maintain Charpy upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb (68 J), unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of Charpy upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code.*

While not explicitly stated by [10CFR50-App G], the NRC's Regulatory Guide 1.161 provides an acceptable means to achieve the goal indicated by the underlined text above using advanced analytical techniques and data based on the engineering science of elastic-plastic fracture mechanics [RG1.161]. Appendix K of ASME SC-XI provides a similar methodology, which is currently being updated to reflect state-of-practice analytical techniques and material models [ASME SC-XI App-K]. USE is not projected to be an issue for Diablo Canyon Unit 1 through 60 years of operation (see [WCAP-17315], Table 7.1-2). Several operating plants have projected USE values below 68J and have had their continued operating integrity demonstrated using [RG1.161] or similar methods.

In summary, as was the case for both of the PTS rules, falling below the [10CFR50-App G] 68J screening criteria on USE does not mean that the plant can no longer operate but rather that additional justifications and/or measurements are needed to demonstrate that continued operations will be safe.

### 2.5.3.2. Pressure-Temperature (P-T) Limit Requirements

During routine operations of a nuclear RPV, limits are imposed on pressure, on temperature, and on temperature change rate by the provisions of Appendix G to Section XI of the ASME Code [ASME SC-XI App-G], which is incorporated into [10CFR50-AppG] by reference. These limits ensure that the vessel cannot fracture during normal operations. These are called "P-T Limits" and, as illustrated by the red and green curves in Figure 15, have a shape similar to the fracture toughness curves illustrated previously in Figure 4. The equations of [ASME SC-XI App-G] transform the fracture toughness values of Figure 4 into allowable pressure values using the engineering science of fracture mechanics. The limits are set with extreme conservatism, this being achieved by assuming (1) loading more severe than occurs, (2) fracture toughness less than actually measured, and (3) a flaw size far bigger than any ever encountered in service [MRP-450]. Figure 15 also shows that the shift of the P-T curve caused by embrittlement combined with plant low-temperature overprotection (LTOP)[17] safety limits can severely constrict the permissible operating window (the grey-shaded region). While P-T limits do not explicitly establish a fracture toughness screening criteria, the size of the permissible operating window, and thus the ease of plant operability, is clearly influenced by the value of fracture toughness after embrittlement. If the operating window closes to the point that plant operations becomes difficult, options are available to demonstrate the safe operability of the plant. These include installing a variable setpoint LTOP control valve or making refinements and improvements to the various input variables that influence the calculated P-T limit curve.

P-T operating limits are assessed by plants when new information on embrittlement forecasting becomes available (e.g., when a surveillance capsule is tested) or when operational factors are modified (e.g., the plant undergoes a power uprate). Depending on the nature and magnitude of these changes an update to P-T limits may be needed from time to time during the plant's lifetime.

## 2.6. Summary

This Chapter provided definitions of key terms needed to understand existing procedures for RPV structural integrity management along with both conceptual and technically detailed descriptions of these procedures. This information applies to all reactors operating under license of the NRC; it is not specific to Diablo Canyon Unit 1. This general background was provided before a response to the concerns expressed by SLOMFP and FOE concerning Diablo Canyon Unit 1 because many of those concerns originate from an incomplete understanding of existing rules and procedures. Direct responses to their public concerns appear next in Chapter 3.

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[17] LTOP limits are based on the value of the calculated P-T limit curve at low temperatures. The LTOP system places a physical limitation on the pressure the plant operator can impose at any given temperature and are not disabled in single-setpoint plants until the pressure of the P-T limit curve exceeds the plant's operating pressure.

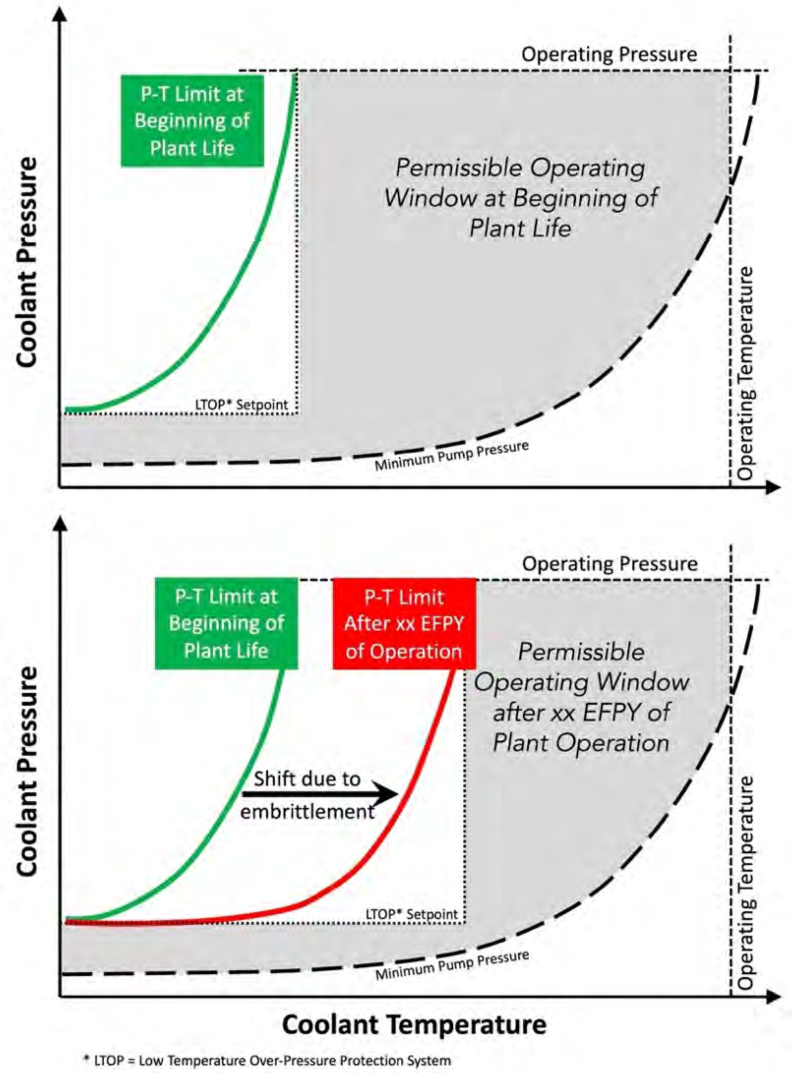


Figure 15. Schematic illustration of the P-T limit curve before (top) and after (bottom) irradiation embrittlement. The curve labeled "minimum pump pressure" depends on plant design and is unchanged by embrittlement; this curve represents the pressure needed to prevent cavitation in the main coolant pumps and/or the minimum pressure needed to activate the pump seals.

# 3. Public Concerns

## 3.1. Introduction

The SLOMFP and the FOE have raised concerns regarding the condition of the Diablo Canyon Unit 1 RPV and its continued operating safety. SLOMFP and FOE have produced many documents detailing their concerns. A review of these documents revealed that all concerns are summarized in two key documents, one providing testimony to the California Public Utilities Commission [SLOMFP-CPUC 2023] and another a request for action to the NRC [SLOMFP-NRC 2023]. Mr. Bruce Severance has also commented independently to the DCISC; his concerns had been incorporated into the two referenced documents.

For clarity of discussion the concerns of SLOMFP, FOE, and Mr. Severance were parsed into the following five categories:

- Questions concerning the analysis of surveillance data.
- Questions concerning the inspections of the RPV beltline.
- Suggestions on alternative testing methods to characterize irradiation damage.
- Questions regarding the methodology for RPV safety assessment.
- Questions regarding deficient materials

Each of the next five sections summarizes the concerns of the SLOMFP and the FOE in each category and then addresses them. To the extent practicable the concerns expressed in [SLOMFP-CPUC 2023] and [SLOMFP-NRC 2023] are quoted directly in the various sub-sections headed "SLOMFP and FOE Concerns" that follow to best retain the original meaning.

## 3.2. Questions on Analysis of Surveillance Data

### 3.2.1. Introduction

On this topic, the SLOMFP and the FOE comments concerned the following four topics:

1. Data "credibility"
2. Capsule testing schedule (number of capsules, periodicity of testing)
3. Use of information from other plants
4. Data analysis methodology

The following four sub-sections each contain a summary of the SLOMFP and FOE comments followed by this consultant's evaluation of the comments.

### 3.2.2. Data Credibility

#### 3.2.2.1. SLOMFP and FOE Concerns

The concerns of SLOMFP and FOE concerning data credibility are summarized as follows:

1. 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." ([SLOMFP-NRC 2023], pdf page 55)
2. 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. ([SLOMFP-NRC 2023], pdf page 55)
3. PG&E's methodology for assessing the credibility of the data is inconsistent with NRC's own guidance for performing credibility assessments [Wichman 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 and a physical basis exists for believing the datapoint to be atypical, then
    - All 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".

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." ([SLOMFP-NRC 2023], pdf page 56).

4. PG&E misinterpreted the [RG1.99R2] credibility criteria to restrict the "scatter" or deviation to one standard deviation of 28°F for welds and 17°F for base metal. Based on this interpretation, they use Criterion 3 to deem the 2003 stress test data are "not credible." However, PG&E seems to note correctly the 2-sigma deviation allowed in their 1993 Capsule Y report, even though they argue for over a decade in their correspondence to the NRC (2003 to 2016) that the Capsule V data did not meet the RG1.99 Criterion 3. … PG&E makes the argument from 2003 into 2016 that all their surveillance data are "not credible" based on a 1-sigma scatter rather than 2-sigma. ([SLOMFP-CPUC 2023], pdf page 188).
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. ([SLOMFP-NRC 2023], pdf page 56)
6. 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. ([SLOMFP-NRC 2023], pdf page 57).

### 3.2.2.2. Consultant's Evaluation

The concerns expressed in the preceding section indicate a misunderstanding of how the terms "credible" and "not credible" are defined and used in an analysis performed according to [RG1.99R2] or [10CFR50.61]. These procedures were described in Section 2.4, with Section 2.4.2.2 explaining in detail the procedures associated with credibility assessment. In brief, the NRC defines the term "credible" in [RG1.99R2] and [10CFR50.61]; it simply means that plant-specific surveillance data follow expected trends (credible), or they deviate from these trends to some extent (not credible). The NRC's required procedures always use the data, whether credible or not credible, to inform the estimate of $\Delta T_{41J}$ for the plant moving forward. The NRC's procedures require a more conservative treatment of not credible data than they do of credible data.

The following consultant's evaluation on each of the concerns, which maintains the numbering scheme used in the preceding section, draws heavily from the previous explanations in Section 2.4.

1. The safety analyses performed by PG&E for the Diablo Canyon Unit 1 RPV did not "discredit" the surveillance data from Capsules S, Y, and V when the Capsule V data became available in 2003. Rather the analyses performed by PG&E followed the

requirements of NRC for treatment of $\Delta T_{41J}$ surveillance data, which was described in Section 2.4.2.2 of this report. As pointed out in that section,

   a. The plant data (from capsules S, Y, and V in this case) are used directly in the credibility assessment; they are not ignored.
   b. Determination that plant surveillance data are "not credible" following the NRC's procedures invariably results in a more conservative (that is: higher) estimate of $\Delta T_{41J}$ due to the more conservative estimation of the *CF* and $\sigma_\Delta$ values than would result if the data were determined to be credible.

2. As discussed in item 1, PG&E correctly followed the NRC's requirements and guidelines concerning credibility assessment. Also, as was described in Section 2.4.2.2 of this report, the credibility assessment procedures are applied to the data set as a whole, not to individual datum within a surveillance data set. PG&E's determination that the data set was not credible based on one datum out of three having a difference of more than 15.6 °C from the best-fit of equation (2-1) to the data is consistent with both the NRC's guidelines [Wichman 1998] and Westinghouse practice [WCAP-18660]; which has been accepted by the NRC for many plants. Finally, it should be noted that the analysis in question, which included only Capsules S, Y, and V was the analysis of record between 2003-2011 [WCAP-15958]. The current analysis of record, which was established in 2011 by [WCAP-17315], is based on a larger data set which has been correctly assessed as being credible following the NRC's procedures (see [Kirk 2024] for details).
3. The guidance cited in this comment appears in a NRC document [Wichman 1998] that concerns two different topics. As identified in page 5 of [Wichman 1998] these topics were (1) the determination of RPV weld (weld wire heat) and surveillance weld best-estimate chemistries, and (2) the evaluation and use of surveillance data. The guidance quoted in Section 3.2.2.1 pertain to the best-estimate chemistry topic and thus have no bearing on evaluation and use of surveillance data.
4. As described in Section 2.4.2.2 of this report and illustrated in Figure 9, the definition of "scatter" in credibility criteria (C) is $\pm 1\sigma_\Delta$. The PG&E use of $\pm 1\sigma_\Delta$ to evaluate credibility criteria (C) is consistent with NRC guidance as documented in [RG1.99R2], as affirmed by the examples in [Wichman 1998] (see Cases 4 and 5), and as affirmed by the methodology applied consistently by Westinghouse [WCAP-18660].
5. The statement that "*in analyzing scattered data, it is common to find points that lie outside of a preconceived scatter band*" is correct. Nevertheless, the treatment of the Capsule S, Y, and V data as not credible from 2003-2011 is in accordance with NRC guidance. Additionally, and also in keeping with NRC guidance, PG&E did not reject "*all the data because one datum did not fall within the bounds*" but rather, in accordance with the NRC guidance on treatment of not-credible data, used the data from Capsules S, Y and V along with NRC's $\Delta T_{41J}$ forecasting procedure to obtain a conservative upper-bound estimate of $\Delta T_{41J}$ based on the state of knowledge at the time.
6. The NRC has no requirements to gather additional data beyond the requirements of the [10CFR50-App H] surveillance program. In the event data are determined to be not

credible the $\Delta T_{41J}$ predictions are made intentionally more conservative by the NRC's procedures.

## 3.2.3. Capsule Testing Schedule

### 3.2.3.1. SLOMFP and FOE Concerns

The concerns of SLOMFP and FOE concerning capsule testing schedule are summarized as follows:

1. The paucity of plant-specific data from 14.27 EFPY (when the Capsule V was withdrawn and tested, 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. ([SLOMFP-NRC 2023], pdf page 58).
2. 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. ([SLOMFP-NRC 2023], pdf page 65).
3. 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. ([SLOMFP-NRC 2023], pdf page 67).

### 3.2.3.2. Consultant's Evaluation

All of the concerns expressed in the preceding section can be addressed by an understanding of capsule lead factor. As was explained in Section 2.1.1 and illustrated in Figure 1, the specimens contained in surveillance capsules accumulate irradiation damage at a rate faster than the RPV that is being surveilled. Capsule V was removed in 2003; after 14.23 EFPY of operation the specimens in the capsule had reached a fluence of $1.34 \times 10^{19}$ n/cm$^2$. Using information from [WCAP-18655] permits estimation that the Diablo Canyon Unit 1 RPV reach this fluence at 35.2 EFPY, which should occur sometime in 2025. In 2011 the $\Delta T_{41J}$ forecast for the Diablo Canyon Unit 1 RPV was augmented by considering two additional $\Delta T_{41J}$ surveillance data collected by the Palisades plant located in Michigan. The data from Palisades extends to a higher fluence ($2.36\times10^{19}$ n/cm$^2$) than the Capsule V data, a fluence that extends to beyond 54 EFPY, a fluence corresponding to the end of 60 calendar years of operation and the highest fluence reported in [WCAP-18655]. Linearly extrapolating the information from this report demonstrates that a fluence of $2.36\times10^{19}$ n/cm$^2$ will be reached after approximately 64 EFPY, an operational duration that cannot be reached even if Diablo Canyon Unit 1 is granted a 20-year license extension by the NRC and operates for the entire extension period. Thus, the data already available to support $\Delta T_{41J}$ forecasting for the limiting Diablo Canyon Unit 1 weld represents conditions beyond 60 years of operation even without benefit of data from Capsule B. When Capsule B, which does contain samples of the limiting weld material [DCL-92-072], is tested the projected capsule fluence will be $\approx3.6\times10^{19}$ n/cm$^2$. These data will enable forecasting of the embrittlement of the Diablo Canyon Unit 1 RPV for operating times beyond that associated with a second license renewal (60-80 operating years).

The extensions requested and granted to the Capsule B removal schedule are consistent with the NRC's guidance [NUREG-1801] and are also consistent with extensions routinely granted to other plants. The NRC staff have indicated their intention to review and possibly update this guidance [SECY-22-0019], a recommendation now under review by the NRC's Commissioners.

## 3.2.4. Use of Information from Other Plants

### 3.2.4.1. SLOMFP and FOE Concerns

The concerns of SLOMFP and FOE about use of information from other plants are summarized as follows:

1. Concerns regarding current "sister plant" data sharing practices
    a. 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, 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. ([SLOMFP-NRC 2023], pdf page 66).
    b. 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. ([SLOMFP-NRC 2023], pdf page 66).

2. Concerns regarding the mechanics of irradiation damage accumulation.
   a. 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. ([SLOMFP-NRC 2023], pdf page 65).
   b. 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. ([SLOMFP-NRC 2023], pdf page 65).

c. 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. 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. ([SLOMFP-NRC 2023], pdf page 67).

### 3.2.4.2. Consultant's Evaluation

Regarding the SLOMFP's concerns about current sister plant data sharing practices (Item 1 in Section 3.2.4.1), use of similar data from another plant's surveillance program is not a "*stop-gap measure*," but rather fulfils a NRC regulatory requirement in [10CFR50.61] (see Section 2.4.2.2). PG&E's assessment that its available surveillance data were not credible between 2003-2011 meets this NRC requirement. The use of a $\pm 1\sigma_{\Delta}$ band in the NRC's credibility criteria (C) rather than a $\pm 2\sigma_{\Delta}$ band as suggested by the SLOMFP is, in my opinion, a statement of NRC preference to use a conservative implementation of the equation (2-1) ETC when data do not comply with criteria (C). Adopting a more conservative treatment than justified on purely scientific grounds is not an unusual stance for a regulatory agency to adopt. Thus, from 2003-2011 the data from Capsules S, Y, and V were deemed non-credible and were used (not discarded and not discredited) along with the procedure illustrated in Figure 10 to inform a conservative estimate (i.e., over-estimate) of the upper-bound$\Delta T_{41J}$ value of the limiting weld. Since 2011 these data were augmented, as required by NRC, with sister plant data from Palisades. Consequently from 2011 until the current day Capsules S, Y, and V data along with the two sister plant data from Palisades have been considered credible and used to inform a $\Delta T_{41J}$ forecast for the Diablo Canyon Unit 1 RPV.

Regarding the SLOMFP's concerns about the mechanics of irradiation damage and how this might inform data sharing practices between plants, it is correct to note that irradiation damage, and thus embrittlement, is thermally activated and so depends also on the operating temperature of the plant. Plant operating temperature is not considered by the NRC's [RG1.99R2] ETC, see equation (2-1). However, the NRC has long recognized that temperature plays a role and in [Wichman 1998] they provide an approximate means of accounting for temperature effects on irradiation damage. Figure 16 shows the guidance from [Wichman 1998], which is referred to as the "degree-per-degree adjustment." Figure 17 compares the degree-per-degree adjustment (**red curve**) with the temperature function adopted in ASTM E900-15 (**blue curve**), which was calibrated to a large set of domestic and international surveillance data and well represents the embrittlement trends in these data [ASTM Adjunct]. This comparison, which is made for the chemical composition of the Diablo Canyon Unit 1 limiting weld, shows that when the degree-per-degree adjustment that the NRC proposed in 1998 does not agree with the temperature dependence inferred from the much larger set of data available today. Were the degree-per-degree adjustment applied over a large temperature range it may under-estimate the effect of temperature on $\Delta T_{41J}$ for the Diablo Canyon Unit 1 weld. Fortunately, the degree-per-degree adjustment is only used in the analysis of the Diablo Canyon Unit 1 data over the small differential between the Diablo Canyon Unit 1 cold leg temperature (roughly 283 °C) and the cold leg temperature of the Palisades sister plant (roughly 280 °C). For this limited range the difference between the NRC's degree-per-degree adjustment and the temperature adjustment inferred from the much larger amount of surveillance data now available is small, at most ≈ 2 °C[18]. This potential non-conservativism relative to currently available data will be considered in evaluation of the RPV embrittlement of Diablo Canyon Unit 1, which is contained in the Part 2 report [Kirk 2024].

[18] It is not possible to make a general statement regarding the conservatism, or lack thereof, of the NRC's degree-per-degree adjustment and currently available surveillance data. The information given here is specific to the chemistry of the Diablo Canyon Unit 1 limiting weld at the stated fluences.

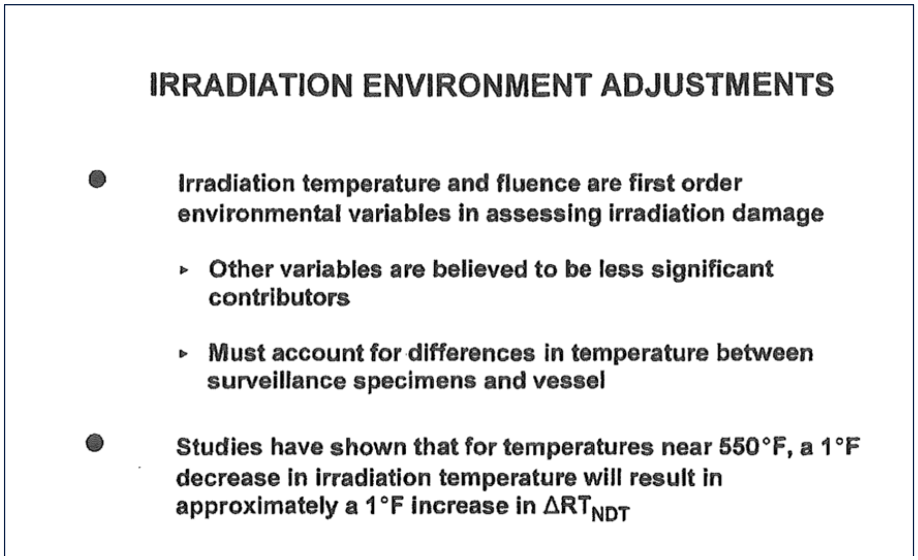


Figure 16. NRC's "degree-per-degree" $\Delta T_{41J}$ adjustment, after [Wichman 1998].

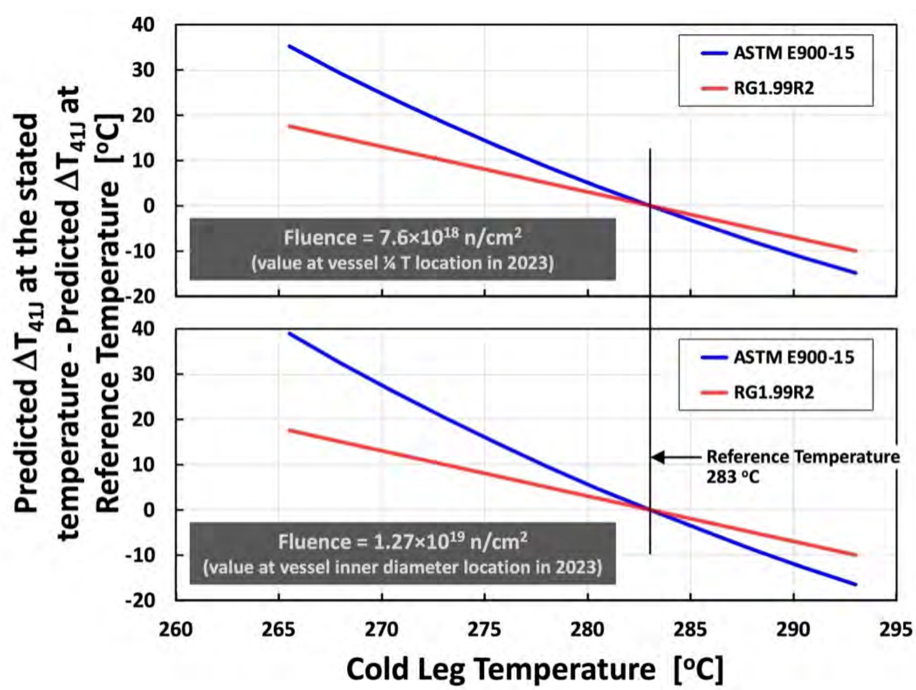


Figure 17. Comparison of the predictions of the NRC's "degree-per-degree" rule that is used with equation (2-1) with the temperature function of ASTM E900-15 ETC for the Diablo Canyon Unit 1 limiting weld material (Cu=0.196, Ni=1, Mn=1.347, P=0.013). An Appendix to this report provides the formula for the ASTM E900-15 ETC.

SLOMFP's concerns expressed in Item 2 in of Section 3.2.4.1) that "the complex nature of radiation embrittlement … is idiosyncratic to individual reactors and may change unexpectedly over time" and that "the accumulation of damage depends upon the temperature history of the component, i.e., on the power level history" have not been borne out by available data.

As will be explained in the following paragraphs, such history effects, and postulated complexities, if existent, do not manifest themselves at a magnitude that inhibits development of generic embrittlement trend curves applicable to all currently operating light water reactors of non-Soviet design. Within this population of reactors exists a great diversity of operating histories and reactor designs. If the complex and idiosyncratic nature of radiation embrittlement were as significant a problem as postulated, efforts to develop embrittlement prediction models over large data populations would fail, and yet they do not.

Two points help to illustrate the robustness of the current approach to embrittlement prediction and, by association, the appropriateness of the practice of using similar data from other plants (so-called "sister plants") to help inform the embrittlement predictions of specific plants of interest:

1. Reactor coolant temperature and its relationship to power level: Information from Table 4.1-1 of the Final Safety Analysis Report Update for Diablo Canyon Unit 1 [Diablo FSARU] allows estimation of the variation of cold leg temperature[19] with power level. The cold leg temperature at 0% power is approximately 303 °C, this falls to 285 °C as the plant reaches 100% power. A non-load following plant like Diablo Canyon Unit 1 will spend most of its operating time at 100% power, so the 100% power temperature is typically used for embrittlement estimation. This approach, which cannot directly account for limited operating time spent at lower power levels, is nevertheless conservative because higher metal temperatures when operating at lower power levels result in lower levels of irradiation damage. Similar information on the effect of power level on cold-leg temperature can be found in [OECD 2011].

2. Factors other than plant design and operating history seem to make the greatest contribution of uncertainty in $\Delta T_{41J}$ estimation. Several different ETCs have been developed worldwide over the past two decades. Many ETCs represent trends for surveillance data collected in a single nation [Eason 2006, Soneda 2013, Todeschini 2011] while the ASTM E900-15 ETC [ASTM Adjunct] was developed using data from reactors operated in 12 different nations. Here the French ETC for their 900 MWe PWR units is compared with the ASTM E900-15 ETC. The French 900 MWe PWR ETC is selected for comparison because the plant design and operating history covered by this ETC is very consistent, applying to 6 older units and 28 newer units of common design. This contrasts with the diversity of plant designs included in the ASTM E900 international dataset which, as mentioned before, includes data from 12 nations (Brazil, Belgium, France, Germany, Japan, Mexico, The Netherlands, South Korea, Sweden, Switzerland, Taiwan, and the United States) and considers both PWRs and BWRs. The quantity of the chemical elements copper and nickel, which are most responsible for irradiation damage sensitivity, is also more consistent in the French reactor fleet than in

[19] The cold-leg temperature best reflects the temperature at the beltline vessel wall and at the surveillance capsule locations during plant operations.

the international dataset (see Figure 18). Table 1 shows that the uncertainty in the$\Delta T_{41J}$ prediction errors are only marginally affected by this greater similarity of plant design and operating practice associated with the French ETC. This small effect could not occur were irradiation damage "*idiosyncratic to individual reactors*" or if "*the accumulation of damage depends upon the temperature history of the component.*"

In conclusion, it should be recognized that existing ETCs, including [RG1.99R2], [ASTM E900-15], [Eason 2006], [Todeschini 2011], and [JEAC4201], are engineering models used to make estimate embrittlement for use in safety and regulatory decision making. In these applications the mean estimate of $\Delta T_{41J}$ is always increased by a factor equal to two times the standard deviation associated with that ETC (see the requirements of [RG1.99R2] as an example). The standard deviation values reflect both the measurement uncertainty associated with the data as well as the complexities of the irradiation damage process not fully captured by the engineering models. This upper-bound estimate of $\Delta T_{41Jc}$ combined with other conservatisms inherent to the assessment process provide protection against making erroneous decisions.

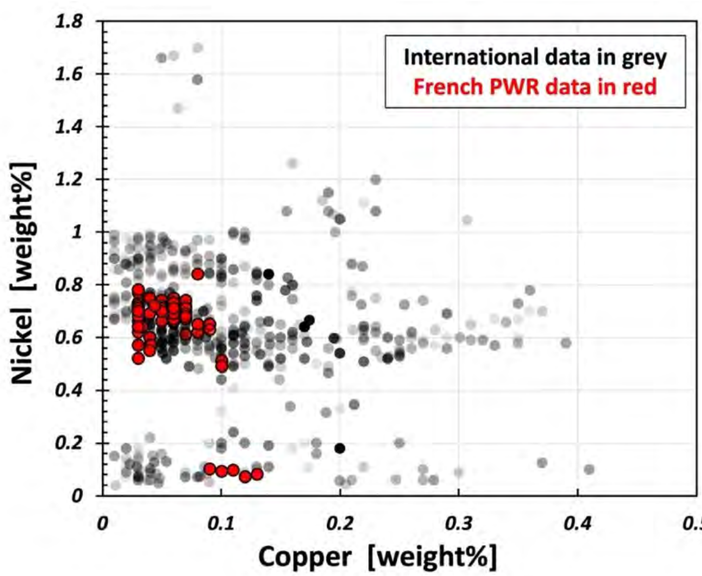


Figure 18. Composition of French RPV steels compared with international data.

Table 1. Comparison of $\Delta T_{41J}$ prediction uncertainty between two ETCs.

| Product Form | Standard deviation of $\Delta T_{41J}$ prediction uncertainty for ASTM E900-15 relative to all international data | Standard deviation of $\Delta T_{41J}$ prediction uncertainty for the French 900 MWe PWR ETC* relative only to French 900 MWe PWRs |
|---|---|---|
| Base Metals | σ=12.4°C n=1,295 | σ=12.7°C n=139 |
| Weld Metals | σ=14.8°C n=757 | σ=13.3°C n=130 |
| * Removal of data from the 6 older 900 MWe PWR designs to make the dataset even more homogeneous based on reactor type changes these σ vales by less than 0.5 °C. | | |

### 3.2.5. Data Analysis Methodology

#### 3.2.5.1. SLOMFP and FOE Concerns

The concerns of SLOMFP and FOE on this topic were expressed as follows, see pdf page 58 of [SLOMFP-NRC 2023]:

> 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.By mathematically deriving an expression for the EoE from coefficients (A, B, C, and D[20]) obtained for the symmetric hyperbolic tangent function
>
> $$FE = A + B \times tanh\left(\frac{T - D}{C}\right)$$
>
> I have calculated
>
> $$EoE = \frac{1}{2}\left[1 + \frac{e^{x} - e^{-x}}{e^{x} + e^{-x}}\right]$$
>
> where[21]
>
> $$x = \frac{RT_{NDT,30} - D}{C}$$
>
> 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.

#### 3.2.5.2. Consultant's Evaluation

It is correct to note that the decision to estimate a transition fracture reference temperature at 41J (30 foot-pounds) is arbitrary. It is further correct to note that such a definition is less than optimal for materials having very low upper shelf energies, because if the USE falls below the energy at which the reference temperature is defined that reference temperature cannot be

20 In the [SLOMFP-NRC 2023] comments the quantity here called "D" was called "$T_0$." I have changed the nomenclature to prevent confusion with the fracture toughness transition temperature defined by [ASTM E1921], which is called $T_0$.

21 The quantity in this equation called $RT_{NDT,30}$ is the same as the value $T_{41J}$ used elsewhere in this report.

calculated. This is not a problem for the Diablo Canyon surveillance weld; at the highest fluence measured so far, the measured USE is 90J (66 ft-lb). It is also not a problem for the operating fleet because any plant having a material forecast to have an upper shelf energy less than 68J is required by [10CFR50-AppG] to perform an analysis using elastic-plastic fracture mechanics data (not Charpy data) to demonstrate continued operating safety (see Section 2.5.3.1).

The Extent of Embrittlement (EoE) metric defined in [SLOMFP-NRC 2023] is a new transition temperature metric; it has not been previously discussed or reviewed in the professional literature. Since the Charpy test does not provide a direct quantification of fracture toughness, any metric determined from Charpy test data ($T_{41J}$, EoE, …) can only be correlated to the fracture toughness transition temperature. In the 1970s when $T_{41J}$ was adopted by the nuclear industry no testing standards existed to estimate the fracture toughness transition temperature, so the selection of $T_{41J}$ over other metrics had to be based on judgement and limited data. The first edition of Regulatory Guide 1.99 [RG1.99R0] published in 1975 states explicitly "*it has been assumed herein that the adjustment of the reference temperature* [for fracture toughness] *is equal to the 30 foot-pound shift* [in Charpy energy]." However, since 1997 an ASTM testing standard has been available to quantify the transition temperature of fracture toughness data, which as discussed in Chapter 2 is called $T_0$ [ASTM E1921]. Figure 19 compares the EoE and $T_{41J}$ Charpy metrics to values of the fracture toughness transition temperature $T_0$ based on data

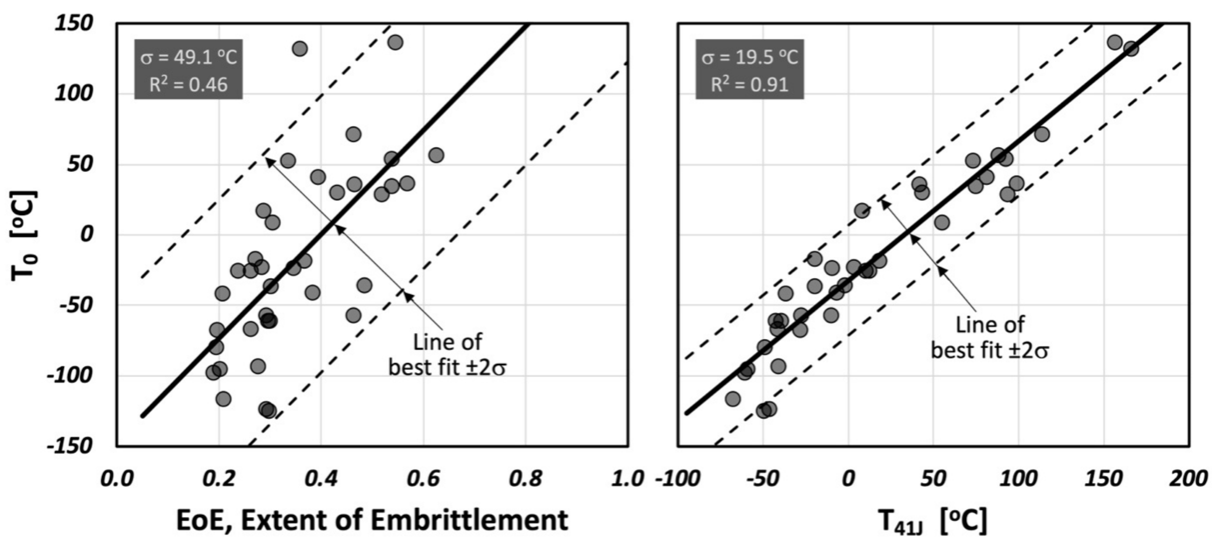


**Figure 19. Comparison of the EoE and $T_{41J}$ transition temperature metrics estimated from Charpy data (left and right graph, respectively) to the fracture toughness transition temperature $T_0$, estimated by ASTM Standard E1921 (vertical axis on both graphs). Data from [EricksonKirk 2009].**

for a variety of RPV steels tested over a wide range of embrittlement reported in [EricksonKirk 2009]. This comparison shows that EoE is not as well correlated with the fracture toughness transition temperature, $T_0$, as is $T_{41J}$. Also, if EoE were used to predict $T_0$ the uncertainty in that prediction would be 2½-times greater than the uncertainty associated with a prediction of $T_0$

from $T_{41J}$ (compare a standard deviation ($\sigma$) value of 49.1 °C for EoE with a $\sigma$ value of 19.5 °C for $\Delta T_{41J}$, as noted in the legends of Figure 19). Thus, it appears appropriate to continue with the use of $T_{41J}$ for embrittlement trending and structural integrity analyses of nuclear RPVs.

## 3.3. Questions Concerning Inspections of the RPV Beltline

On this topic, the SLOMFP and the FOE comments concerned the following three topics:

1. The small number of indications revealed by UT inspections of Diablo Canyon Unit 1 is not plausible.
2. The time interval permitted between UT inspections is too long.
3. The time-dependent damage and cracking phenomena operative in current design light water reactor RPVs

The following three sub-sections each contain a summary of the SLOMFP and FOE comments; this is followed by the consultant's evaluation of the comments.

### 3.3.1. The Small Number of Indications Revealed by UT are not Plausible

#### 3.3.1.1. SLOMFP and FOE Concerns

The concerns of SLOMFP and FOE on this topic were expressed as follows:

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. ([SLOMFP-NRC 2023], pdf page 63)
2. 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. 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. ([SLOMFP-NRC 2023], pdf page 65).

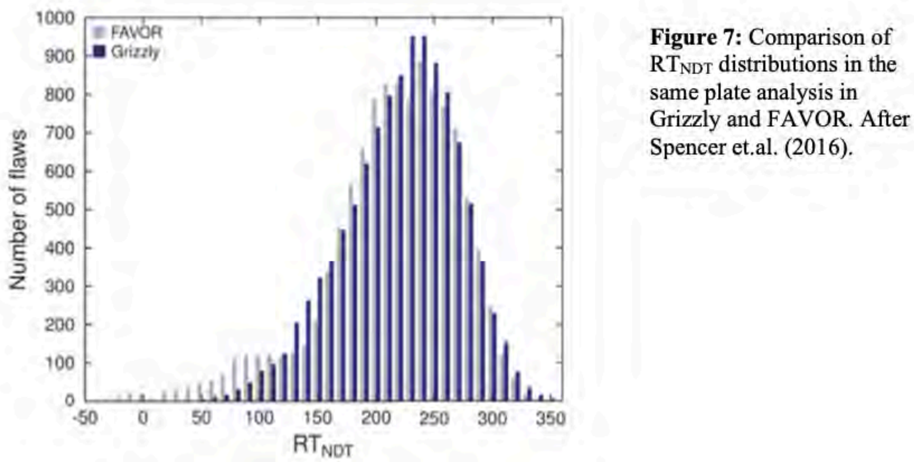


Figure 20. Figure 7 from [SLOMFP-NRC 2023].

### 3.3.1.2. Consultant's Evaluation

Concerning Item 1, SLOMFP have provided no evidence that the large number of UT indications detected in the Doel 3 and Tihange 2 reactors in Belgium in 2013 can plausibly exist in Diablo Canyon Unit 1. The root-cause of these flakes was tied to unusual aspects of the initial manufacturing process that caused hydrogen flakes to exist in the Doel and Tihange forgings [Electrabel 2012]. Review of the information from Doel 3 and Tihange 2 by EPRI led to the conclusion that "*it is unlikely that conditions similar to those observed at Doel 3 exist in U.S. PWRs; and even if substantial* [flake] *indications are postulated to exist in beltline ring forgings in U.S. PWRs, the potential for vessel failure is acceptably low*." The NRC later concurred with this assessment [NRC 2015a]. Also, in 2015 the nuclear regulatory agency in Belgium (FANC) convened panels of national and international experts to review the concerns of Professors Macdonald and Bogaerts [FANC 2015]. Based on this investigation the FANC concluded the following (***emphasis added***):

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

> *the FANC requires the Licensee to perform follow-up UT-inspections, using the qualified procedure on the entire reactor pressure vessels wall thickness at the end of the next cycle of Doel 3 and Tihange 2, and there after at least every three years.*

While subsequent inspections performed between 2016 and 2020 revealed some indications that had not been found in the original inspections, other indications that were found in 2012 and 2014 could not be detected subsequently [Framatome 2019, Framatome 2021, AREVA 2016, AREVA 2017]. This was diagnosed as a "threshold effect" with different indications being detected (or not) based on if they exceeded the detection threshold (or not). The numbers of these appearing or disappearing indications was a small percentage, ≈1-6%, of the total number of indications found originally. The average size of the indications reduced slightly (by less than 1mm) over time. This and other evidence led the inspection company and the utility to conclude that the flaws are not evolving with time [Framatome 2019, Framatome 2021, AREVA 2016, AREVA 2017]. An independent evaluation performed by a student of Professor Bogaerts[22] reached the same conclusion [Dumont 2022].

Concerning Item 2, neither the FAVOR [Williams 2016] nor the GRIZZLY [Spencer 2016] probabilistic fracture mechanics codes predict the number and distribution of flaws in an RPV. Rather, both codes sample from a flaw distribution established in [NUREG/CR-6817]. The [NUREG/CR-6817] work is based on experimental information on flaw sizes and aspect ratios collected by non-destructive and destructive examinations of RPV materials from four plants, on physical models, and on expert judgement. [NUREG/CR-6817] developed a mathematical model and program called VFLAW to simulate flaw populations. These flaw populations are read as input flies by the FAVOR and GRIZZLY codes; neither FAVOR nor GRIZZLY predicts flaw populations as asserted in [SLOMFP-NRC 2023]. Because the flaw model used by both computer codes is an input the near exact agreement of the FAVOR and GRIZZLY outputs cited by the SLOMFP is not surprising.

The very large number of flaws simulated by the VFLAW code is an over-representation of the flaw density expected in an operating RPV due to the several conservatisms adopted during development of the VFLAW model. In particular, as stated in Section 4.1 of [NUREG-1808] "*All NDE indications used in constructing the flaw models were treated as cracks and, therefore, potentially deleterious to RPV integrity. However, many of these indications were in fact volumetric, which lessens significantly the probability of brittle fracture initiation.*" NDE of the weld beltline region typically reveals a much smaller quantify of indications. For example [MRP-207] documents 19 weld indications within 25.4 mm of the vessel ID surface for Beaver Valley Unit 2, while [WCAP-17628] documents 42 indications for the Palisades beltline and extended beltline regions.

[22] In 2015 Professor Bogaerts collaborated with Professor Macdonald in raising the possibility of defect evolution driven by hydrogen to the Belgian regulatory authority.

## 3.3.2. The Time Interval Permitted Between UT Inspections is Too Long

### 3.3.2.1. SLOMFP and FOE Concerns

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 are suspect and the reasoning is circular. ([SLOMFP-NRC 2023], pdf page 68).

### 3.3.2.2. Consultant's Evaluation

There is no relationship between the timing of the surveillance capsule withdrawals, which monitor embrittlement and the timing of reactor vessel inspections, which monitor cracking. Also, neither the results of the surveillance capsule evaluation nor the data used in that evaluation has any impact on the reactor vessel inspection schedule. As described previously (see Figure 2) the surveillance capsule withdrawals are spread out over the operating lifetime of the plant following the schedule of ASTM E185 as sometimes amended by the NRC. The postulated relationship between these two monitoring and inspection schedules stated in Section 3.3.2.1 results from a misunderstanding of the current regulatory process.

NRC regulations and the ASME Code require a once every 10-year in-service inspection (ISI) of the reactor vessel beltline welds and surrounding base materials, see [10 CFR 50.55a] paragraph (g). These periodic inspections monitor flaws that may exist within the RPV beltline to determine if they increase in size over time. The 10-year inspection interval was established at the beginning of electricity production by nuclear power, a time when there was very little experience concerning crack growth rates. Now decades of operational experience demonstrates that there are not any sub-critical cracking mechanisms that are increasing the size of these flaws. Thus, the ISI exams of the vessel beltline have become, for all practical purposes, a re-examination of a static condition every 10 years. Occasionally ISI will find a "new" flaw in a location where none was previously recorded; however, this is typically caused by the increase of inspection quality and accuracy over time.

Between 2005-2011 the PWR Owner's group performed research to justify extension of the ISI interval from 10 to 20 years [WCAP-16168]. This risk-informed approach used the NRC's FAVOR model to demonstrate that the increase of RPV fracture risk resulting from increased crack depth due to fatigue was well within the NRC's guidelines as expressed in Regulatory Guide 1.174 [RG 1.174]. Plants wishing to extend their ISI interval following this approach need to submit an exemption request, in the form of a Relief Request letter, to the NRC because the 10-year inspection interval remains a requirement of SC-XI IWB-2412 as incorporated by reference in USA regulation by 10 CFR 50.55a (see, for example, [Southern Co. 2015]). A plant specific request for ISI interval extension can be justified by demonstrating that the bounding plants analyzed in [WCAP-16168] cover the plant of interest. PG&E has made such a request of the NRC and gained approval [DCL-14-074]

### 3.3.3. The Time-Dependent Damage and Cracking Phenomena Operative in Current-Design Light Water Reactor RPVs

#### 3.3.3.1. SLOMFP and FOE Concerns

1. Why has hydrogen embrittlement from the coolant side not been considered as it is a well-known failure mode of embrittled steels in the oil and gas industry, for example? ([SLOMFP-CPU C2023], pdf page 8).
2. In our opinion, RG1.99 falls short in not accounting for the approximately half-inch thick, ductile austenitic stainless steel liner and the possibility of hydrogen injection into the RPV from the radiolysis of the coolant that contains considerable hydrogen [typically 25 cc (STP)H2/kg H2O]. In the case of the Davis Besse PWR a few years ago, the RPV had been breached at the control rod guide tube penetrations via corrosion by concentrated boric acid, yet the reactor continued to operate at full power with the coolant only being contained by the stainless steel (SS) liner. In the case of atomic hydrogen injection, hydrogen embrittlement is a well-known phenomenon in many other technological areas including the oil and gas industry (embrittled heat-affected zones welds in production tubing), naval aviation (embrittled landing gear), and bridges (e.g., failed high-strength steel tendons in the new Bay bridge a few years ago), to name but a few. In our opinion, the former (SS liner) will likely mitigate the RPV radiation embrittlement phenomenon but the latter will certainly exacerbate the problem. It is for this reason that we describe the latter as a "force multiplier." [SLOMFP-CPUC 2023], pdf page 218).

#### 3.3.3.2. Consultant's Evaluation

Environmentally assisted crack growth (hydrogen cracking, stress corrosion cracking) of the low-alloy ferritic structural steel from which the RPV is made as well as the austenitic stainless steel that is weld-deposited on the RPV inner diameter has not been observed via operating experience in a PWR like those at Diablo Canyon. Reinforcing this extensive operating experience, the following paragraphs, taken from [NUREG-1806], explain why stress corrosion cracking of both the low alloy steel and of the stainless-steel cladding is highly unlikely for a nuclear RPV.

Concerning the assumption that there is no subcritical crack growth due to environmental effects on the low-alloy pressure vessel steel, [NUREG-1806] states the following:

> *Stress corrosion cracking (SCC) requires the presence of three factors: an aggressive environment, a susceptible material, and a significant tensile stress. If these three factors exist and SCC can occur, growth of intrinsic surface flaws in a material is possible. Since an accurate PTS calculation for the low-alloy steel (LAS) pressure vessel should address realistic flaw sizes, the potential for crack growth in the reactor vessel LAS as a result of SCC needs to be analyzed, in principle. However, for the reasons detailed in the following paragraphs, SCC for LAS in PWR environments is highly unlikely and, therefore, is appropriately assumed not to occur… .*
>
> *The first line of defense against SCC of LAS is the cladding that covers much of the LAS surface area of the reactor vessel and main coolant lines. This prevents the environment from contacting the LAS and, therefore, obviates any possibility of SCC of the pressure boundary.*
>
> *Additionally, several test programs have been conducted over the past three decades, all of which show that SCC in LAS cannot occur in normal PWR or boiling-water reactor (BWR) operating environments. SCC of LAS in the reactor coolant environment is controlled by the electrochemical potential (often called the free corrosion potential). The main variable that controls the LAS electrochemical potential is the oxygen concentration in the coolant. During normal operation of a PWR, the oxygen concentration is below 5ppb. The electrochemical potential of LAS in this environment cannot reach the value necessary to cause SCC [IAEA 1990, Hurst 1985, Rippstein1989, Congleton1985]. During refueling conditions, the oxygen concentration in the reactor coolant does increase. However, the temperature during an outage is low, rendering SCC kinetically unfavorable. During refueling outage conditions with higher oxygen concentrations but lower temperatures, the electrochemical potential of the LAS would still not reach the values necessary for SCC to occur [Congleton1985].*

Concerning the assumption that there is no subcritical crack growth due to environmental effects on the austenitic stainless-steel cladding, [NUREG-1806] states the following:

> *Under conditions of normal operation, the chemistry of the water in the primary pressure circuit is controlled with the express purpose of ensuring that SCC of the stainless-steel cladding cannot occur. Even under chemical upset conditions (during which control of water chemistry is temporarily lost), the rate of crack growth in the cladding is exceedingly small. For example, Ruther et al. reported an upper bound crack growth rate of ≈$10^{-5}$ mm/s (≈$4x10^{-7}$ in/s) in poor-quality water (i.e., high oxygen) environments [Ruther 1984]. The amount of crack extension that could occur during*

*a chemical upset is therefore quite limited, and certainly not sufficient to compromise the integrity of the clad layer.*

## 3.4. Suggestions on Alternative Testing Methods to Characterize Irradiation Damage

## 3.4.1. Introduction

The SLOMFP and the FOE stated that the "Use of nano-indentation hardness can provide additional data on the irradiation damage experienced by Diablo Canyon Unit 1 materials, and with greater certainty than the Charpy impact test." The following sub-section contains a summary of the SLOMFP and FOE comment; this is followed by the consultant's evaluation of the comment.

## 3.4.2. Use of Nano-Indentation Hardness Measurements would Provide Additional Insights

### 3.4.2.1. SLOMFP and FOE Concerns

The concerns of SLOMFP and FOE on this topic were expressed as follows:

1. 10CFR50.61(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 Charpy impact testing, 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. ([SLOMFP-NRC 2023], pdf page 68).
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. ([SLOMFP-NRC 2023], pdf page 68).

3. 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. ([SLOMFP-NRC 2023], pdf page 70).
4. 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 (Figure 10 from [SLOMFP-NRC 2023] appears here as Figure 21) 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. ([SLOMFP-NRC 2023], pdf page 71).

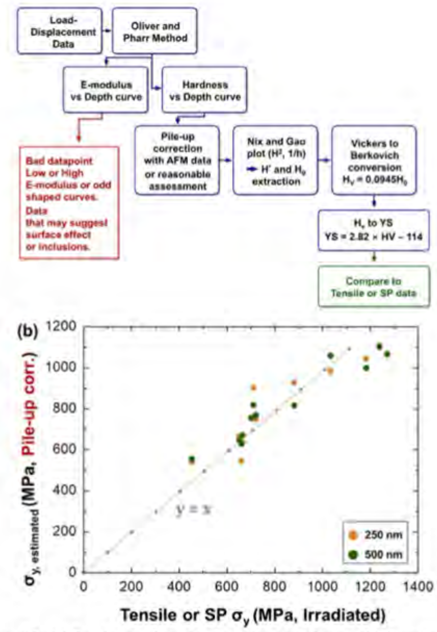


**Figure 10:** (a) Roadmap of nano indentation techniques. (b) Correlation between tensile test and shear punch test generated data to data obtained from nano hardness.

Figure 21. Figure 10 from [SLOMFP-NRC 2023].

### 3.4.2.2. Consultant's Evaluation

Professor Hosemann's work on nano-indentation hardness techniques is well documented in the literature [Hosemann 2009, Hosemann 2018, Krumwiede 2018, Hosemann NEUP]. In [Hosemann 2009] he explains that the nano-indentation test permits investigation of samples exposed to low-energy ion-irradiation. In such experiments the irradiation damage to the material sample, and thus the region within which the material properties are affected, is limited to the surface. The low indentation loads used by the nano-technique allow the resultant hardness values to reflect the properties of the irradiated surface layer alone, uninfluenced by the unirradiated substrate. The Charpy samples from Diablo Canyon Unit 1 were exposed to high-energy neutron irradiation during their many years in the RPV. Consequently, neutron irradiation has affected the material properties of the entire sample, not just the surface layer. While nano-indentation could be used to perform a hardness investigation on the Diablo Canyon Unit 1 Charpy samples it would also be possible to use more conventional hardness testing techniques such as Rockwell [ASTM E18] or Vickers [ASTM E384]. Both tests make larger indents and thus sample more of the material than will nano-indentation, while still being able to perform multiple indentation tests on a single previously tested Charpy sample. The greater sampling/averaging of material properties by the conventional hardness tests should offer some advantage relative to the proposed nano technique in terms of reduced data scatter.

In principle a measured hardness value can be correlated to yield strength, and yield strength can then be correlated to either fracture toughness or Charpy toughness. Such correlations between mechanical properties are well established and often used in both research studies and forensic investigations, see [Wagenhofer 2002] for example. However, there is no regulatory precedent of which I am aware for using fracture toughness values that have been estimated through hardness tests and a sequence of correlative relationships. The uncertainties introduced to the measurement – however precisely and repeatedly made – by the sequence of empirical correlations will degrade the ability of such data to illuminate the embrittlement trends of the Diablo Canyon Unit 1 surveillance materials. If such testing were performed it would be valuable to also collect hardness data on the unirradiated samples of the surveillance materials so that the increase in hardness produced by irradiation damage could be calculated.

In summary, hardness testing as proposed by SLOMFP may provide additional information concerning the embrittlement trends of the Diablo Canyon Unit 1 surveillance materials. However, the uncertainties introduced by the multiple correlations needed to get from measured hardness values to an estimate of fracture toughness will complicate interpretation of the measurements, as will the lack of regulatory precedent for using fracture toughness values estimated from hardness data as part of a safety assessment.

Should there be a need to perform additional testing on the Diablo Canyon Unit 1 beltline materials, techniques exist to directly estimate fracture toughness from small compact tension specimens that can be machined from the broken halves of the tested Charpy specimens. These small specimens are called "mini compact tension" specimens, or mini-CTs. [Sánchez 2023] reviews the literature on mini-CT testing and demonstrates that these small samples provide comparable estimates of the Master Curve transition temperature $T_0$ to values estimated from tests performed on larger specimens. As described in [MRP-418] and [MRP-462] there is both regulatory precedent and standard ASME procedures for using values of $T_0$ in RPV integrity assessment. An evaluation of Diablo Canyon Unit 1 informed in part by Master Curve $T_0$ data appears in the Part 2 report [Kirk 2024], including the possible need for additional testing.

## 3.5. Questions regarding the methodology for RPV safety assessment

### 3.5.1. Introduction

On this topic, the SLOMFP and the FOE comments concerned the following five topics:

1. Apparent fluence reduction from $3 \times 10^{19}$ to $1 \times 10^{17}$ n/cm$^2$.
2. Requirements of the alternate PTS rule (10CFR50.61a) as compared with the original PTS rule (10CFR50.61)
3. Treatment of the stainless-steel liner
4. Treatment of low-temperature thermal annealing of irradiation damage
5. Evaluation of the extended beltline

The following five sub-sections each contain a summary of the SLOMFP and FOE comments; this is followed by the consultant's evaluation of the comments.

### 3.5.2. Apparent Fluence Reduction from $3\times10^{19}$ to $1\times10^{17}$ n/cm$^2$

#### 3.5.2.1. SLOMFP and FOE Concerns

What is the justification for reducing the fluence from $3\text{x}10^{19}$ n/cm$^2$ to $10^{17}$ n/cm$^2$ after 20 EFPYs of extended operation? ([SLOMFP-CPUC 2023] pdf page 9)

#### 3.5.2.2. Consultant's Evaluation

While preparing this document Mr. Severance was contacted to provide information on where in PG&E's records such a reduction is documented. Mr. Severance could not locate such documentation. It is believed that their statement may be based on based on a misreading of the minimum fluence level at which evaluation of the extended beltline is required by the NRC [NRC 2014].

If documentation of the cause for this concern is located in the future this question can then be evaluated.

### 3.5.3. Requirements of the alternate PTS rule (10CFR50.61a) as compared with the original PTS rule (10CFR50.61)

#### 3.5.3.1. SLOMFP and FOE Concerns

Why was PG&E, when unable to meet the required criteria of 10 CFR 50.61a, allowed to invalidate their own test data to fall back to the more generous RG1.99 Position 1.2 under 10 CFR 50.61 which enabled them to ignore the credible 2003 test data? ([SLOMFP-CPUC 2023], pdf page 10).

#### 3.5.3.2. Consultant's Evaluation

This comment reveals several misunderstandings.

- As explained in Sections 2.5.1 and 2.5.2, the alternate PTS rule [10CFR50.61a] has never established "required criteria." As for all PWRs, Diablo Canyon Unit 1 must comply with the requirements of the original PTS rule [10CFR50.61]. One means to comply with this rule is to elect to use the alternate rule [10CFR50.61a]. PG&E stated in [DiabloLRA 2009] its intent to use the alternate rule during its period of license renewal based on data available at that time. However, data available since 2011 makes this action unnecessary [WCAP-17315.
- [RG1.99R2] has been incorporated directly into the original PTS rule [10CFR50.61], so there is no need to talk about the provisions of [RG1.99R2] if [10CFR50.61] is being followed.
- As discussed in Section 2.4.2.2, the process followed by PG&E between 2003-2011 that determined their then-existing three value set of $\Delta T_{41J}$ data to be not credible is the

process required by the NRC in [10CFR50.61]. As illustrated in Figure 10, the Unit 1 data was not invalidated but, rather, was used following the NRC's process to establish a conservative upper-bound estimate for $\Delta T_{41J}$, an estimate that was then used in the [10CFR50.61] PTS evaluation. When the Palisades data was used in 2011, the data set was judged to be credible following the NRC's criteria.

## 3.5.4. Treatment of the Stainless-Steel Liner

### 3.5.4.1. SLOMFP and FOE Concerns

Why has no account been taken for the stainless-steel liner in determining the susceptibility of the RPV to brittle fracture and hence a LOCA? ([SLOMFP-CPUC 2023], pdf page 12)

### 3.5.4.2. Consultant's Evaluation

A thin layer of weld-deposited austenitic stainless steel, generally between 4-9 mm in thickness, is placed on the inner diameter of the RPV wall to protect the low-alloy ferritic vessel steel from both general corrosion and stress corrosion cracking. The welding process used to deposit the cladding is a potential source of defects which could contribute to vessel failure, however there is no empirical evidence that such through-cladding defects exist [NUREG/CR-6817]. Surface defects were conservatively included in the NRC's probabilistic model used to inform both PTS rules [10CFR50.61, 10CFR50.61a]. However, these cladding flaws were not found to be responsible for a significant part of the vessel failure probability associated with PTS [NUREG-1806, NUREG-1874]. This potential for flaws notwithstanding, the much higher fracture toughness of austenitic stainless steels coupled with the structural benefits of the thin clad layer should, if anything, reduce the vessel failure probability. Again, the NRC took a conservative approach in its probabilistic modeling and ignored these benefits [NUREG-1806, NUREG-1807] in the development of both PTS rules [10CFR50.61, 10CFR50.61a]. It may be noted that it is also common practice in ASME to ignore the possible structural benefits of the cladding when assessing the suitability of nuclear pressure vessels for routine service loadings.

## 3.5.5. Treatment of low-temperature thermal annealing of irradiation damage

### 3.5.5.1. SLOMFP and FOE Concerns

Why has no attention been given to low temperature thermal annealing of radiation damage when credible literature on the phenomenon exists and demonstrates that it is an important factor in determining the ultimate hardening of the steel for a given neutron fluence? Low temperature annealing of radiation damage in embrittled RPVs is a little-known but nevertheless recognized phenomenon in the literature on RPV embrittlement. This phenomenon is important because it appears to limit the ultimate level of embrittlement that might occur. That level is determined by the equality of the rate of formation of defects and hence is a function of the fluence and the rate of recombination of the Frenkel defects that are produced by neutron bombardment. The rate of recombination of Frenkel defects is primarily a function of time and temperature because it is a first-order kinetic process. If the rate of recombination increases for a given fluence then the ultimate extent of embrittlement will be lower but if the fluence increases for a given temperature the extent of embrittlement will be higher. I do not believe that the NRC has recognized this phenomenon in their regulations, however, it is possible that the NRC may align potential updated regulations with the advances in science that have been made. ([SLOMFP-CPUC 2023], pdf page 13)

### 3.5.5.2. Consultant's Evaluation

It is correct to note that the temperature at which a steel is exposed to neutron irradiation affects the magnitude of embrittlement. A 1961 report by the US Naval Research Laboratory [Steele 1961] concluded that:

> *At an irradiation temperature of 550 °F (288 °C), the process of self-annealing of neutron-induced changes in notch ductility is a concurrent factor, reducing the transition-temperature shift to approximately one-hundred degrees less than that observed for materials irradiated at temperatures less than 450 °F (232 °C). … Shifts in the ductile-to-brittle transition temperature of these materials after irradiation at 400 and 450 °F (204 and 232 °C) are not significantly different from those observed for the same materials irradiated at 260 °F (238 °C). It is concluded that no appreciable annealing of radiation effects occur during irradiation at temperatures under 450 °F (232 °C).*

In 1967 an extensive report from the Oak Ridge National Laboratory [Whitman 1967] stated the following:

> *Radiation-induced mechanical property changes are temperature dependent. At moderately elevated temperatures many of the defects caused by radiation are mobile, and annealing of the radiation effects may occur. Vacancies are mobile at the operating temperatures of nuclear power reactor pressure vessels and will diffuse to dislocations, grain boundaries, and inclusions or will coalesce to form larger lattice defects. In addition, the larger defects will grow by vacancy diffusion at the expense of the smaller defects, and the total effect can be strikingly similar to aging and over-aging in a precipitation-hardenable alloy. … In fact, the temperature range for the recovery of the major portion of radiation effects is also the temperature range in which most commercial water-cooled nuclear pressure vessels operate. Thus, radiation damage and at least partial recovery of the effects occur simultaneously in the nuclear pressure vessel.*

The surveillance specimens, and the resultant $\Delta T_{41J}$ and $\Delta USE$ data determined from the tested specimens, measure both the radiation damage and annealing effects that, as stated in [Whitman 1967] "occur simultaneously." Since the NRC's ETC [RG1.99R2] and more current

ETCs [ASTM E900-15] are calibrated to surveillance data (see [Randall 1986] and [ASTM Adjunct], respectively), these equations capture the effect of annealing at reactor temperatures.

## 3.5.6. Evaluation of the Extended Beltline

### 3.5.6.1. SLOMFP and FOE Concerns

Despite PG&E's 68% increase in the projected life of Unit 1 in addition to a 15% shift in the brittleness estimates of the most compromised plates and welds, there were still concerns expressed by PG&E as late as October 2015 regarding the nozzle shell welds. A statement on page 36 of DCL-12-124 admits that the nozzle shell welds and related components may not meet fracture toughness limits through the entire 20-year extension, even after the fluence calculations were used to justify an approximately 80% shift in the data. ([SLOMFP-NRC 2023], pdf page 191).

### 3.5.6.2. Consultant's Evaluation

The following text appears on page 36 of DCL-12-124 and appears to be the origin of this concern (***emphasis*** added) [DCL-12-124]:

> *For license renewal, Westinghouse performed additional calculations to define which materials in the DCPP pressure vessels, other than beltline materials, are projected to exceed the threshold neutron fluence of 1x10$^{17}$ n/cm$^{2}$ at 54 EFPY (extended beltline materials). The results of these calculations are documented in WCAP-17299-NP, for Units 1 and 2, through EOLE. For both units,* ***although the nozzle shell course and the associated nozzle shell to intermediate shell weld are projected to exceed the 1x10$^{17}$ n/cm$^{2}$ threshold****, the nozzles themselves as well as the nozzle-to-nozzle shell welds remain below the 1x10$^{17}$ n/cm$^{2}$ threshold through 54 EFPY. Likewise, the lower shell to lower head weld remains below 1x10$^{17}$ n/cm$^{2}$ through 54 EFPY for both units. Table 4.2-3 [of the 2009 PG&E License Renewal Application, see [DiabloLRA 2009] shows the EOLE fluence values for all beltline and extended beltline materials for both Units 1 and 2.*

The stated fluence threshold of 1x10$^{17}$ n/cm$^{2}$ is not a fracture toughness screening criterion. Data has shown that measurable effects of irradiation on the fracture toughness properties of RPV steels begins to occur around a fluence of 1x10$^{17}$ n/cm$^{2}$. During the original 40-year license period the 1x10$^{17}$ n/cm$^{2}$ threshold limited the materials considered in design of the surveillance program to those in the RPV shell adjacent to the reactor's active core (the so-called "beltline" region). However, as licenses have been extended to 60 and 80 years the region of the RPV experiencing fluences above 1x10$^{17}$ n/cm$^{2}$ has extended to include regions above and below the active core. This region is now commonly referred to as the "extended beltline" and can include portions of the nozzle shell course. In [NRC 2014] the NRC clarified that these extended beltline regions need to be considered in PTS and P-T limits assessments. The ***emphasized statement*** from [DCL-12-124] indicates only that the 1x10$^{17}$ n/cm$^{2}$ threshold fluence will be exceeded for some extended beltline materials, it does not say or imply that these materials do not meet the NRC's fracture toughness requirements. [DiabloLRA 2009] gives fluence values as high as 5.2x$10^{17}$ n/$cm^2$ for extended beltline materials. Figure 22 copies Table 6.1-2 from [WCAP-17315]. This image shows that while the extended beltline materials exceed the fluence threshold (red box) as reported in [DiabloLRA 2009], none of the extended beltline materials has a $RT_{PTS}$ value (that is: estimated toughness value, blue box) that comes close to the $RT_{PTS}$ values for the conventional beltline materials (green box) or to any NRC screening criteria.

In summary, while the extended beltline materials in Diablo Canton Unit 1, exceed the 1x$10^{17}$ n/$cm^2$ threshold fluence, none of these materials has a toughness transition temperature predicted to exceed any NRC regulatory screening criteria, even after 60 years of service.

## 3.6. Questions Regarding Deficient Materials

On this topic, the SLOMFP and the FOE comments asserted that the Diablo Canyon Unit 1 RPV is made from a deficient material. The following sub-sections contains a summary of the SLOMFP and FOE comments; this is followed by the consultant's evaluation of the comments.

### 3.6.1. SLOMFP and FOE Concerns

It is common knowledge that there are known metallurgical flaws in the Unit 1 reactor vessel, excessive copper and nickel impurities in welds and plate metals that were discovered only after the Unit 1 RPV was delivered to DCPP. It is well documented that there were engineering errors made in the metallurgical specifications of Unit 1 plate and weld alloys and that Westinghouse, the manufacturer, realized their errors and corrected them prior to the second reactor vessel being installed at DCPP. As stated in a Fairewinds and Associates report filed with the CPUC in 2016 ([SLOMFP-NRC 2023], pdf page 192):

> *"Diablo Canyon Unit 1 was one of the first US atomic reactors ever designed and manufactured by the nuclear power industry, therefore, unusual, and consequential errors were made in the design and engineering. The wrong material was used to weld the atomic reactor vessel introducing impurities in the weld material that have caused significant and accelerated radiation damage in the form of embrittlement...Diablo Canyon now ranks as one of the five worst reactors out of the 99 remaining operational reactors in the US."*
>
> <u>*Quoted From*</u>*: Neutron Embrittlement at Diablo Canyon Unit 1 Nuclear Reactor, A. Gundersen, Fairewinds Associates Inc., 2016, page 2.*

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

| Reactor Vessel Material | R.G. 1.99, Rev. 2 Position | CF[(b)] (°F) | Fluence[(c)] ($x10^{19}$ n/cm², E > 1.0 MeV) | FF[(c)] | $IRT_{NDT}$ [(d)] (°F) | $\Delta RT_{NDT}$ (°F) | $\sigma_U$[(d)] (°F) | $\sigma_\Delta$[(f)] (°F) | Margin (°F) | $RT_{PTS}$ (°F) |
|---|---|---|---|---|---|---|---|---|---|---|
| | | | | *Beltline Materials* | | | | | | |
| IS Plate B4106-1 | 1.1 | 85.3 | 2.02 | 1.1917 | -10 | 101.7 | 0 | 17 | 34 | 126 |
| IS Plate B4106-2 | 1.1 | 81.0 | 2.02 | 1.1917 | -3 | 96.5 | 0 | 17 | 34 | 128 |
| IS Plate B4106-3 | 1.1 | 55.2 | 2.02 | 1.1917 | 30 | 65.8 | 17[(e)] | 17 | 48.1 | 144 |
| → Using non-credible surveillance data | 2.1 | 37.4 | 2.02 | 1.1917 | 30 | 44.6 | 17[(e)] | 17 | 48.1 | 123 |
| LS Plate B4107-1 | 1.1 | 89.8 | 2.01 | 1.1904 | 15 | 106.9 | 0 | 17 | 34 | 156 |
| LS Plate B4107-2 | 1.1 | 82.2 | 2.01 | 1.1904 | 20 | 97.9 | 0 | 17 | 34 | 152 |
| LS Plate B4107-3 | 1.1 | 81.4 | 2.01 | 1.1904 | -22 | 96.9 | 0 | 17 | 34 | 109 |
| IS Longitudinal Weld 2-442A & B | 1.1 | 226.8 | 1.49 | 1.1104 | -56[(e)] | 251.8 | 17[(e)] | 28 | 65.5 | 261 |
| → Using credible surveillance data | 2.1 | 214.1 | 1.49 | 1.1104 | -56[(e)] | 237.7 | 17[(e)] | 14 | 44.0 | 226 |
| IS Longitudinal Weld 2-442C | 1.1 | 226.8 | 0.768 | 0.9259 | -56[(e)] | 210.0 | 17[(e)] | 28 | 65.5 | 220 |
| → Using credible surveillance data | 2.1 | 214.1 | 0.768 | 0.9259 | -56[(e)] | 198.2 | 17[(e)] | 14 | 44.0 | 186 |
| LS Longitudinal Weld 3-442A & B | 1.1 | 226.8 | 1.19 | 1.0485 | -56[(e)] | 237.8 | 17[(e)] | 28 | 65.5 | 247 |
| → Using credible surveillance data | 2.1 | 214.1 | 1.19 | 1.0485 | -56[(e)] | 224.5 | 17[(e)] | 14 | 44.0 | 213 |
| LS Longitudinal Weld 3-442C | 1.1 | 226.8 | 2.01 | 1.1904 | -56[(e)] | 270.0 | 17[(e)] | 28 | 65.5 | 280 |
| → Using credible surveillance data | 2.1 | 214.1 | 2.01 | 1.1904 | -56[(e)] | 254.9 | 17[(e)] | 14 | 44.0 | **243** |
| IS to LS Circ. Weld 9-442 | 1.1 | 172.2 | 2.01 | 1.1904 | -56[(e)] | 205.0 | 17[(e)] | 28 | 65.5 | 215 |

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

| Reactor Vessel Material | R.G. 1.99, Rev. 2 Position | CF[(b)] (°F) | Fluence[(c)] ($x10^{19}$ n/cm², E > 1.0 MeV) | FF[(c)] | $IRT_{NDT}$ [(d)] (°F) | $\Delta RT_{NDT}$ (°F) | $\sigma_U$[(d)] (°F) | $\sigma_\Delta$[(f)] (°F) | Margin (°F) | $RT_{PTS}$ (°F) |
|---|---|---|---|---|---|---|---|---|---|---|
| | | | *Extended Beltline Materials* | | | | | | | |
| US Plate B4105-1 | 1.1 | 82.2 | 0.0341 | 0.2365 | 28 | 19.4 | 17[(e)] | 9.7 | 39.2 | 87 |
| US Plate B4105-2 | 1.1 | 82.4 | 0.0341 | 0.2365 | 9 | 19.5 | 17[(e)] | 9.7 | 39.2 | 68 |
| US Plate B4105-3 | 1.1 | 98.2 | 0.0341 | 0.2365 | 14 | 23.2 | 17[(e)] | 11.6 | 41.2 | 78 |
| US Longitudinal Weld 1-442A | 1.1 | 215.7 | 0.0245 | 0.1948 | -20 | 42.0 | 0 | 21.0 | 42.0 | 64 |
| US Longitudinal Weld 1-442B | 1.1 | 215.7 | 0.0149 | 0.1428 | -20 | 30.8 | 0 | 15.4 | 30.8 | 42 |
| US Longitudinal Weld 1-442C | 1.1 | 215.7 | 0.0306 | 0.2222 | -20 | 47.9 | 0 | 24.0 | 47.9 | 76 |
| US to IS Circ. Weld 8-442 | 1.1 | 197.5 | 0.0341 | 0.2365 | -56[(e)] | 46.7 | 17[(e)] | 23.4 | 57.8 | 48 |

**Notes:**

(a) The 10 CFR 50.61 methodology was utilized in the calculation of the $RT_{PTS}$ values. See Section 1 of this report for details.

(b) Taken from Table 5.1-2 of this report.

(c) Taken from Table 2.1-1 of this report.

(d) Initial $RT_{NDT}$ values are taken from Table 3.1-1 of this report and are measured values, unless otherwise noted.

(e) Initial $RT_{NDT}$ values are either generic or estimated; therefore, $\sigma_U$ = 17°F.

(f) Per Appendix A of this report, the surveillance plate data was deemed non-credible. Per the guidance of 10 CFR 50.61, the base metal $\sigma_\Delta$ = 17°F for Position 1.1 and for Position 2.1 with non-credible surveillance data. Per Appendix A, the surveillance weld data was deemed credible. Per the guidance of 10 CFR 50.61, the weld metal $\sigma_\Delta$ = 28°F for Position 1.1 and with credible surveillance data $\sigma_\Delta$ = 14°F for Position 2.1. However, $\sigma_\Delta$ need not exceed $0.5*\Delta RT_{NDT}$.

Figure 22. Table 6.1-2 from [WCAP-17315].

### 3.6.2. Consultant's Evaluation

A decision cannot be called an "error" when information revealing the error was unavailable at the time the decision was made. The NRC issued a construction permit for Diablo Canyon Unit 1 in 1968. At that time there was limited knowledge regarding what aspects of a steel's metallurgy most elevates its sensitivity to irradiation damage. It was not until two years later, in 1970, that information on the deleterious effects of copper was generally available [Steele 1970]. As revealed by the data in Figure 23, by 1973-1975 this information had influenced the steel specifications adopted by nuclear RPV manufacturers. Before 1973 copper contents in RPV base metals and welds as high as 0.35 weight percent were common, however after 1975 almost every plant has a maximum copper content no more than 0.1 weight percent. In 1975 the ASTM specification for RPV steel was modified to include an upper limit on copper of 0.12 weight percent [ASTM A533-75].

Figure 23 also reveals that many plants have beltline materials of higher copper content (i.e., greater embrittlement sensitivity) than Diablo Canyon Unit 1. These plants have remained compliant with all NRC and ASME standards. All have operated safely and in no case have their operating lives been shortened by embrittlement. At least 25 units having a copper content equal to or exceeding that of the Diablo Canyon Unit 1 weld remain in operation today.

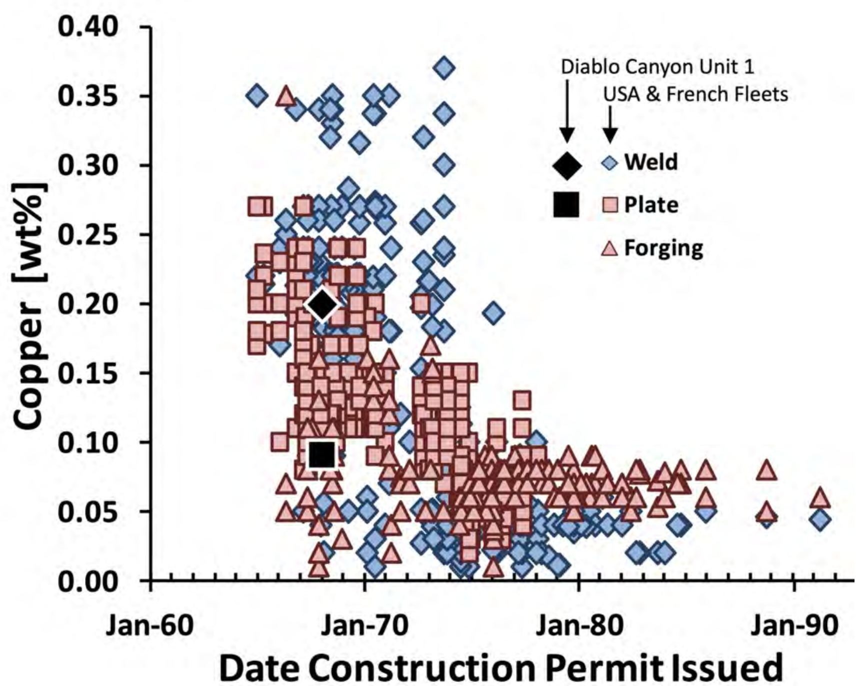


Figure 23. Variation of copper in RPV beltline materials versus plant construction date, after [Kirk 2018b].

# 4. Summary and Conclusions

From 2009 to 2018 the Pacific Gas and Electric Company (PG&E) had been pursuing a 20-year license extension for the nuclear power plant at Diablo Canyon with the Nuclear Regulatory Commission (NRC). In 2018 PG&E informed the NRC it wished to withdrawal that application due to then-projected energy demands and economic factors in California. However, in 2022 the State of California directed the California Public Utilities Commission (CPUC) to direct PG&E to again pursue license extension with the NRC for Diablo Canyon, whose NRC licenses currently expire in 2024 (Unit 1) and 2025 (Unit 2). Following this decision members of two organizations, the San Luis Obispo Mothers for Peace (SLOMFP) and the Friends of the Earth (FOE), placed before the Diablo Canyon Independent Safety Committee (DCISC) their concerns regarding the embrittlement of the in Unit 1 reactor pressure vessel (RPV) and, consequently, its continued operating safety. SLOMFP and FOE have expressed similar concerns to both the CPUC and the NRC.

This is one of two reports prepared for the DCISC. This report addresses the concerns expressed by SLOMFP and FOE while the companion report evaluates the current state of knowledge concerning the embrittlement of the Diablo Canyon Unit 1 RPV and reviews that unit's current RPV safety analyses. This report also includes introductory material that describes in general terms the procedures used for nuclear RPV surveillance, for embrittlement and fracture toughness forecasting, and for RPV safety evaluation.

The SLOMFP and FOE concerns fall into five categories. This consultant's evaluation of each set of concerns is summarized below:

1. **Questions concerning the analysis of surveillance data**: These included concerns regarding the "credibility" of available surveillance data, the capsule testing schedule,

the use of surveillance data from other plants as part of the Diablo Canyon Unit 1 evaluation, and concerns about the analysis methodology itself.

a. Credibility: Many of the concerns expressed by SLOMFP and FOE result from a misunderstanding of the NRC's requirements and guidelines for analysis of the plant-specific embrittlement data collected as part of a surveillance program; this report explains these requirements. The NRC requires an evaluation of the "credibility" of surveillance data, which is an assessment of how well the surveillance data set from a specific plant matches the embrittlement trends expected based on the embrittlement trend curve (ETC) in Regulatory Guide 1.99 Revision 2. The plant's data are always used in this evaluation, they are neither discredited nor discarded. If data are determined to be not-credible (as was the case for Diablo Canyon Unit 1 between 2003 and 2011) the NRC's required procedure mandates more conservatism in estimation of the embrittlement trends than would otherwise be the case. Since 2011, when more data for the plant's limiting weld became available the data set has been judged to be credible, leading to the use of NRC guidelines that credit the greater state of knowledge. Concerns about PG&E not correctly interpreting the NRC's credibility guidelines are no longer relevant because since 2011 the plant's analysis has been based on credible data.
b. Capsule Testing Schedule: The extensions requested by PG&E and granted by the NRC to the Capsule B removal and testing schedule are consistent with the NRC's guidance and with extensions routinely granted to other plants. By design the specimens in surveillance capsules accumulate irradiation dose (damage) at a rate greater than the vessel being surveilled. Because of this, large intervals may exist between capsule withdrawals; these intervals do not compromise the integrity of the surveillance condition monitoring program, nor do they compromise plant safety. Relevant data for Diablo Canyon Unit 1 includes the Capsule V measurement made in 2003 at a fluence of $1.36 \times 10^{19}$ n/cm$^2$ and data from the same weld wire heat obtained from Capsule SA240 irradiated in the Palisades plant at a fluence of $2.38 \times 10^{19}$ n/cm$^2$. Both measurements exceed the current fluence of Diablo Canyon Unit 1; at the end of Cycle 23 in 2023 the vessel fluence at the inner diameter was $1.27 \times 10^{19}$ n/cm$^2$. The fluence of Capsule B, when withdrawn, will be at least $3.7 \times 10^{19}$ n/cm$^2$ while at the end of 60 operating years the vessel fluence at the inner diameter is projected at $2.07 \times 10^{19}$ n/cm$^2$. Thus, the currently available data for the limiting weld in Diablo Canyon Unit 1 bounds the vessel fluence expected at 60 years. When the Capsule B data becomes available it can be used to further improve the estimated embrittlement magnitude during extended plant operations.
c. Data from Other Plants
    i. PG&E's use of similar data from another plant's surveillance program is not a "*stop-gap measure,*" but rather fulfils a NRC regulatory

requirement expressed in [10CFR50.61]. The so-called sister plant data from two capsules irradiated in the Palisades plant in Michigan are from the same weld wire heat as the limiting weld in Diablo Canyon Unit 1. The reported copper and nickel values from the Palisades weld compare well (within measurement uncertainty) to the Diablo Canyon Unit 1 weld. As copper and nickel are primarily responsible for a RPV steel's sensitivity to neutron irradiation, the Palisades weld provides a good match to the Diablo Canyon Unit 1 weld. Use of these data together is appropriate and should provide improved quantification of the irradiation sensitivity of the Diablo Canyon Unit 1 weld.

ii. Related concerns were raised as to the similarity of irradiation environments in different plants, particularly with respect to irradiation temperature and irradiation temperature history. SLOMFP and FOE contended that irradiation damage is "*idiosyncratic to individual reactors*" and that "*the accumulation of damage depends upon the temperature history of the component.*" Evidence presented herein shows the magnitude of such history effects, if existent, are small and, in any event, are accounted for by the uncertainty margins adopted in the required embrittlement evaluation. A comparison of the treatment of temperature adopted by the NRC in Revision 2 of Regulatory Guide 1.99 to that of a more modern trend curve calibrated to an up-to-date database (ASTM E900-15) showed that the Regulatory Guide's approach may underestimate the embrittlement of the limiting weld in Diablo Canyon Unit 1 by at most 2 °C. This potential non-conservativism will be considered in Part 2 of this report.

d. <u>Analysis Methodology</u>: The Extent of Embrittlement (EoE) metric proposed for embrittlement trending of surveillance capsule Charpy data was compared to the metric used currently ($T_{41J}$). The analysis was based on a collection of data from RPV steels spanning a wide range of embrittlement. The analysis showed that the EoE metric does not correlate as well with the true fracture toughness transition temperature ($T_0$) as does $T_{41J}$. Additionally, the analysis determined that the EoE metric is 2½ times less accurate in predicting $T_0$ than is $T_{41J}$.

2. **Questions concerning inspections of the RPV beltline**: These included concerns that the small number of indications found by non-destructive evaluation of the Diablo Canyon Unit 1 RPV are not plausible, that the time interval permitted between inspections is too long, and that important time-dependent cracking phenomena have been ignored.

   a. <u>The small number of indications in Diablo Canyon Unit 1 is not plausible</u>: Evidence was presented from other RPV beltline weld inspections of Beaver Valley Unit 2 and at Palisades that the small number of indications reported for Diablo Canyon Unit 1 is not uncommon. Evidence cited by SLOMFP and FOE for a much larger numbers of flaws resulted from a misunderstanding of the capabilities of the FAVOR and GRIZZLY computer codes; these were explained. Finally, SLOMFP and FOE expressed a concern that Diablo Canyon Unit 1 may experience a large number of hydrogen flakes similar to those revealed in 2012 by inspection of two reactors in Belgium. There is no evidence that these types of defects, which occurred due to unusual aspects of the manufacturing process of the Belgian RPVs, could plausibly exist for Diablo Canyon Unit 1. Postulations that such defects, if present, could increase in size over time were rejected by a national and international panel of scientific experts convened by the Belgian regulatory authority. Finally, the Electric Power Research Institute (EPRI) concluded, and the NRC concurred, that such flakes, even if present, do not create an undue risk of RPV failure.

b. <u>Time interval between inspections</u>: SLOMFP and FOE expressed a concern that the 10-year interval between ultrasonic (UT) inspection of the RPV beltline is too large, especially in view of uncertainties associated with the surveillance data for Diablo Canyon Unit 1. However, there is no relationship between the timing of the surveillance capsule withdrawals, which monitor embrittlement, and the timing of reactor vessel inspections, which monitor cracking. The mechanism causing embrittlement does not cause the sub-critical cracking that is monitored by UT. NRC regulations and the ASME Code require a once every 10 years in-service inspection. These inspections monitor flaws that may exist within the RPV beltline to determine if they increase in size over time. The 10-year inspection interval was established at the beginning of electricity production by nuclear power when there was very little experience concerning crack growth rates. Now operational experience demonstrates that there are not any sub-critical cracking mechanisms active. Thus, the 10-year UT exams have effectively become a re-examination of a static condition every 10 years. Occasionally UT will find a "new" flaw in a location where none was previously recorded; however, this is typically caused by the increase of inspection quality and accuracy over time.

c. <u>Time dependent cracking phenomena</u>: SLOMFP and FOE expressed a concern that environmental cracking caused by hydrogen embrittlement may be occurring and is not being effectively monitored. However, environmentally assisted crack growth (hydrogen cracking, stress corrosion cracking) of the low-alloy ferritic structural steel from which the RPV is made as well as the austenitic stainless steel that is weld-deposited on the RPV inner diameter has not been observed in the extensive operating experience that now exists for PWRs like those at Diablo Canyon. Control of the chemistry of the primary coolant to limit the oxygen content is the factor most responsible for this lack of environmentally assisted cracking.

3. **Suggestions on alternative testing methods to characterize irradiation damage**: The SLOMFP and FOE suggested that "nano-indentation hardness can provide additional data on the irradiation damage experienced by Diablo Canyon Unit 1 materials, and

with greater certainty than the Charpy impact test." The nano-indentation technique permits investigation of samples exposed to low-energy ion-irradiation in which the material properties are only altered by irradiation near the surface of the sample. The Charpy samples from Diablo Canyon Unit 1 were exposed to high-energy neutron irradiation, which changed the properties of the entire sample, not just the surface layer. While nano-indentation could be used it would also be possible to use more conventional techniques such as Rockwell or Vickers hardness. The greater sampling/averaging of material properties by the conventional hardness tests may offer some advantage relative to the proposed nano technique in terms of reduced data scatter. In principle a hardness value can be correlated to yield strength, and yield strength can then be correlated to either fracture toughness or Charpy toughness. However, there is no regulatory precedent for using fracture toughness values that have been estimated from hardness tests and a sequence of correlative relationships. The uncertainties introduced to a measurement – however precisely and repeatedly made – by the sequence of empirical correlations will degrade the ability of such data to illuminate the embrittlement trends of the Diablo Canyon Unit 1 surveillance materials. If re-testing of previously irradiated samples is desired other techniques that use mini compact tension specimens to directly measure the fracture toughness transition temperature of the material are available. There is both regulatory precedent and standard ASME procedures for using such approaches to support RPV integrity assessment. The possible need for additional testing to establish an embrittlement status will be made in the Part 2 report.

4. Questions regarding the methodology for RPV safety assessment: These included concerns on the following five topics:
    a. Apparent fluence reduction from $3\times10^{19}$ to $1\times10^{17}$ n/cm$^2$: Representatives of SLOMFP and FOE were unable to document this reduction in fluence. If such documentation is located in the future this question can then be evaluated. It is believed that their statement may be based on based on a misreading of the minimum fluence level at which evaluation of the extended beltline is required by the NRC.
    b. Requirements of the alternate PTS rule (10CFR50.61a) as compared with the original PTS rule (10CFR50.61): SLOMFP and FOE expressed concerns that PG&E, were "*unable to meet the required criteria of 10 CFR 50.61a*" and were then *"allowed to invalidate their own test data to fall back to the more generous RG1.99 Position 1.2 under 10 CFR 50.61 which enabled them to ignore the credible 2003 test data.*" This concern reveals a misunderstanding regarding the current regulatory process. Compliance with the alternate PTS rule [10CFR50.61a] is not a requirement, but rather a means of complying with [10CFR50.61] that a licensee may elect. PG&E has not currently elected to use the alternate rule. Also, as described in item 1a above, PG&E followed the required regulatory process for data credibility evaluation. This evaluation did not ignore data but rather used all then-available data to establish a more

conservative $T_{41J}$ estimate because the data set was determined to be not credible.

c. Treatment of the stainless-steel liner: SLOMFP and FOE expressed a concern that the effects of the stainless-steel cladding in determining the susceptibility for brittle failure of the RPV had not been considered. This cladding, a thin layer of weld-deposited austenitic stainless steel, is placed on the inner diameter of the RPV wall to protect the low-alloy ferritic vessel steel from both general corrosion and stress corrosion cracking. The welding process used to deposit the cladding is a potential source of defects which could contribute to vessel failure. These defects were conservatively included in the NRC's probabilistic model that informed the alternate PTS rule [10CFR50.61a], however they were found to not be responsible for a significant part of the vessel failure probability associated with PTS. Also, the much higher fracture toughness of austenitic stainless steel coupled with the structural benefits of the thin clad layer should, if anything, reduce the vessel failure probability. Again, the NRC took a conservative approach in its probabilistic modeling and ignored these benefits. Thus, the effects of the stainless-steel cladding have been accounted for.

d. Treatment of low-temperature thermal annealing of irradiation damage: The SLOMFP and FOE expressed concerns that "*no attention been given to low temperature thermal annealing of radiation damage*." In this context, "low temperature" means "RPV operating temperatures." The surveillance specimens, and resultant $\Delta T_{41J}$ and $\Delta$USE data determined from the tested specimens, measure both the radiation damage and annealing effects that occur simultaneously. Since the functional forms of ETCs are calibrated to these surveillance data these equations capture implicitly the effect of annealing at reactor temperatures.

e. Evaluation of the extended beltline: The SLOMFP and FOE expressed concerns that PG&E "admits that the nozzle shell welds and related components may not meet fracture toughness limits through the entire 20-year extension." This concern resulted from a misinterpretation of guidance for consideration of so-called "extended beltline materials," which may include the nozzle course. NRC has directed that all materials forecast to experience a fluence above $1 \times 10^{17}$ n/cm$^2$ during license extension must be evaluated for both P-T limits and PTS. PG&E has acknowledged that several of their nozzle course materials exceed the $1 \times 10^{17}$ n/cm$^2$ limit and has therefore included them in their P-T limits and PTS assessments. This analysis demonstrated that the nozzle course materials still have a transition temperature much lower than the beltline weld and, therefore, do not limit plant operations.

5. **Questions regarding deficient materials**: SLOMFP and the FOE expressed concerns that the Diablo Canyon Unit 1 RPV was made from a deficient material that was selected in error. The origin of this concern is the high copper content (approximately 0.2 weight percent) of the limiting weld. Copper is precipitated from the ferrite matrix

by irradiation and, as such, is the element primarily responsible for the irradiation sensitivity of RPV steels. Evidence was presented to show that at the time of the Diablo Canyon Unit 1 construction in 1968 the significant role played by copper in enhancing the irradiation damage sensitivity of steel was not yet well recognized. Plants having construction permits issued through 1973 had copper contents as high as 0.35 weight percent and the applicable ASTM standard was not modified to include a limitation on copper content until 1976. Thus, the RPV steels and welds used to construct the Diablo Canyon Unit 1 RPV were selected consistent with the state of knowledge at the time. Of the early-date construction plants, at least 25 units having a copper content equal to or exceeding that of the Diablo Canyon Unit 1 weld remain in operation today. These plants have remained compliant with all NRC and ASME standards. All have operated safely and in no case have their operating lives been shortened by embrittlement.

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# 6. Mark Kirk – Resume

## Education

| Degree | Year Granted | University | Discipline |
|---|---|---|---|
| Ph.D. | 1992 | University of Illinois at Urbana-Champaign | Civil Engineering |
| M.S. | 1989 | University of Maryland at College Park | Mechanical Engineering |
| B.S. | 1984 | Virginia Polytechnic Institute and State University | Engineering Science and Mechanics |

## Employment

| Dates | Company Name & Location | Concluding Position Title |
|---|---|---|
| 2018-date | Central Research Institute of Electric Power Industry (CRIEPI)<br>*Yokosuka, Kanagawa, Japan* | Guest Research Fellow |
| 2018-date | Phoenix Engineering Associates, Inc. (PEAI)<br>*Unity, New Hampshire, USA* | Principal Engineer |
| 1999-2018 | Nuclear Regulatory Commission<br>*Rockville, Maryland, USA* | Senior Materials Engineer |
| 1997-1999 | Westinghouse Electric Corporation<br>*Pittsburgh, Pennsylvania, USA* | Senior Engineer |
| 1992-1997 | Edison Welding Institute<br>*Columbus, Ohio, USA* | Energy & Chemical Industry Team Leader |
| 1981-1992 | David Taylor Research Center (U.S. Navy)<br>*Annapolis, Maryland, USA* | Senior Project Engineer |

## Nuclear Power Skills and Experience

I began my work in nuclear power in 1997 with two years at the Westinghouse hot cells followed by nearly 20 years with the Nuclear Regulatory Commission (NRC). Since 2018 i have held an appointment as a Guest Research Fellow at the Central Research Institute of Electric Power Industry (CRIEPI) in Japan and I serve as a Principal Engineer at Phoenix Engineering Associates, Inc (PEAI).

At Westinghouse I participated in the initial development of industry plans for Master Curve (direct fracture toughness) implementation. Elements completed while at Westinghouse included development of ASME Code Cases, and their technical basis, which allowed use of Master Curve to estimate the index temperature ($RT_{To}$) for the $K_{IC}$ and $K_{IR}$ curves, and also development of the Kewaunee lead plant submittal.

At the NRC I continued to focus on RPV integrity issues, including the following:

- Led the government and industry team responsible for developing of technical basis for the alternate pressurized thermal shock rule (10 CFR 50.61a). Also worked as part of a team to develop guidance for application of 10 CFR 50.61a (Regulatory Guide 1.230).
- Led the team responsible for structural assessment and residual life prediction of the corroded head at the Davis-Besse nuclear power plant.
- Led and oversaw the contract that developed the probabilistic fracture mechanics (PFM) Code called FAVOR (Fracture Analysis of Vessels, Oak Ridge) and an on-line database of nuclear RPV surveillance data called REAP (Reactor Embrittlement Archive Project).
- Identified the need to re-assess the NRC's procedures to estimate $RT_{NDT}$ for early-construction plant steels (Branch Technical Position 5.3).
- Led the NRC's assessment of the continued adequacy of regulatory guidance on embrittlement prediction (Regulatory Guide 1.99).
- Addressed citizens' concerns of embrittlement and vessel failure risk at the Palisades nuclear plant via a webinar and a series of public meetings.
- Provided NRC support to several international partners in the aftermath of unexpected findings during made during inspections (Doel and Tihange in Belgium from 2012-2015, Beznau in Switzerland from 2015-2017), in response to significant public interest (Kori in South Korea), and as part of educational or development missions (taught a PFM course for IAEA in China, gave invited speech at a PFM symposium in Japan).

On non-RPV topics I worked on assessment of external hazards (postulated pipeline explosions near nuclear plants) and worked as part of a team developing regulatory guidance on the use of PFM in licensing actions (Regulatory Guide 1.254).

At CRIEPI I am working on projects focused on RPV integrity issues, including the following:

- Development of embrittlement trend curves and ETC modeling procedures, including machine learning techniques such as k-nearest neighbor (kNN). Application of the kNN method to develop advanced methods for surveillance during long term operation is now being evaluated.

- Developed a justification to eliminate the need for HAZ testing as part of RPV surveillance monitoring.
- Support of various CRIEPI projects including efforts to gain acceptance for using mini compact tension (mini-CT) specimens to determine $T_0$ and efforts to develop and gain acceptance of PFM techniques in Japan.
- Participating in the European Commission project ENTENTE, which concerns embrittlement modeling and database development.

At PEAI I am working on projects focused on RPV integrity issues, including the following:

- Development of an ASME Code case designated N-830 that allows the use of Master Curve and extended Master Curve models in ASME Code assessments.
- Development of an ASME Code case designated N-914 that provides a comprehensive methodology to account for neutron irradiation embrittlement in ASME Code assessments and includes parallel paths for both traditional (meaning Charpy and NDT-based) as well as Master Curve approaches.
- Removal of HAZ requirements for RPV beltline materials from the ASME Code.
- Assessed the impact of potential changes to US Regulatory Guide 1.99 on operating plants in the USA.
- Development of practical plant guidelines for addressing RPV integrity issues.
- Development of embrittlement prediction models applicable at the low reactor operating temperatures anticipated for future small modular reactor operations (SMRs).

## Professional Organizations

### American Society for Testing and Materials (ASTM)

I have been active in ASTM since the beginning of my career with the US Navy. My early activities focused on Committee E08 on Fatigue and Fracture where I contributed to the development of standards E1820 (J-R and $J_{Ic}$ testing) and E1921 (Master Curve $T_0$ testing). Upon joining the nuclear industry my focus shifted to Subcommittee E10.02 on the Behavior and Use of Nuclear Structural Materials. Within E10.02 I led a five-year effort that produced the first consensus embrittlement trend curve (E900-15) applicable to all western-grade light water reactor steels. I am currently responsible for coordinating the continued evaluation of the adequacy of the E900-15 predictive model as new data becomes available.

### American Society of Mechanical Engineers (ASME)

I am active in the ASME Working Group on Flaw Evaluation (WGFE) and in the Working Group on Operating Plant Criteria (WGOPC), which i have chaired since 2023. While with Westinghouse I supported efforts that produced the first Code Cases to use Master Curve (N-629, N-631). More recently I have worked as part of a team to develop a Code Case revision (N-830) and technical basis demonstrating the applicability of direct fracture toughness ("Master Curve") models for use in Section XI Appendices A, G, and K. In 2021 this revision to N-830 was adopted as part of the ASME Code. Since 2019 i have been developing a Code Case designated N-914 that provides a comprehensive methodology to account for neutron irradiation embrittlement in ASME Code assessments and includes parallel paths for both traditional (meaning Charpy and NDT-based) as well as Master Curve approaches. This Code Case is currently in the balloting process. On-going activities also focus on removal of HAZ analysis requirements from Section XI of the Code.

## Publications

I have authored or co-authored over 120 refereed journal articles, technical papers in conference proceedings, and technical reports, and have also served as editor for four ASTM Special Technical Publications. Key publications relevant to my work in nuclear structural integrity include the following:

- Lott, R.G., Kirk, M.T., and Kim, C.C., "Master Curve Strategies for RPV Assessment," Westinghouse Electric Corporation, WCAP-15075, November 1998.
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- *Materials Reliability Program: Assessment of the Need to Consider Heat Affected Zone (HAZ) Properties in RPV Integrity Assessments (MRP-475).*EPRI, Palo Alto, CA: 2022. 3002023870.
- Yoshinori Hashimoto, Akiyoshi Nomoto, Mark Kirk, Kenji Nishida, "Development of new embrittlement trend curve based on Japanese surveillance and atom probe tomography data," *Journal of Nuclear Materials*, Volume 553, 2021, 153007, ISSN 0022-3115, https://doi.org/10.1016/j.jnucmat.2021.153007.
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A more complete citation list can be found on GOOGLE Scholar at https://scholar.google.com/citations?user=tVlFcrsAAAAJ&hl=en

# Appendix

## Embrittlement Trend Curve for $\Delta T_{41J}$

The embrittlement trend curve from [ASTM E900-15] is as follows:

$$\Delta T_{41J:E900(15)} = \frac{5}{9}[max\{min(\mathrm{Cu},\ 0.28) - \ 0.053, 0\}M + B]$$

$$M = \begin{bmatrix} \mathrm{W: 0.968} \\ \mathrm{P: 0.819} \\ \mathrm{F: 0.738} \end{bmatrix} max\{min[113.87\,(ln(\Phi) - ln(4.5 \times 10^{16})), 612.6], 0\}\left(\frac{1.8\ \mathrm{T} + 32}{550}\right)^{-5.45}\left(0.1 + \frac{P}{0.012}\right)^{-0.098}\left(0.168 + \frac{Ni^{0.58}}{0.63}\right)^{0.73}$$

$$B = \begin{bmatrix} \mathrm{W: 0.919} \\ \mathrm{P:1.080} \\ \mathrm{F:1.011} \end{bmatrix} 3.593 \times 10^{-10}\Phi^{0.5695}\left(\frac{1.8\ \mathrm{T} + 32}{550}\right)^{-5.47}\left(0.09 + \frac{P}{0.012}\right)^{0.216}\left(1.66 + \frac{Ni^{8.54}}{0.63}\right)^{0.39}\left(\frac{Mn}{1.36}\right)^{0.3}$$

$$SD = \begin{bmatrix} \mathrm{W: 7.681} \\ \mathrm{P: 6.593} \\ \mathrm{F: 6.972} \end{bmatrix} \times (\Delta T_{41J})^{[\mathrm{W:0.181}\quad \mathrm{P:0.163}\quad \mathrm{F:0.199}]}$$

In this equation W, P, and F mean weld, plate, and forging (respectively); standard reference materials (SRMs) are classified as plates. SD stands for "standard deviation." Composition variables have units of weight percent, temperatures are expressed in °C, and fluence (E > 1MeV) is expressed in n/cm$^2$.

Full details on the development and basis for this equation appear in [ASTM Adjunct]. Use of this equation should follow the requirements and limitations of [ASTM E900-15].

# An Evaluation of the Status of Diablo Canyon Unit 1 With Respect to Reactor Pressure Vessel Condition Monitoring and Prediction

## Part 2: Evaluation of Diablo Canyon Unit 1 Embrittlement

Consultant's report to the Diablo Canyon Independent Safety Committee

Submitted by
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Date: 26 January 2024

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**Disclaimer**

This report summarizes the consultant's evaluation of the state of embrittlement in the Diablo Canyon Unit 1 reactor pressure vessel and various questions related to this topic. The report is provided as information to the Diablo Canyon Independent Safety Committee (DCISC). The consultant has no responsibility for decisions made by the DCISC, or by any other body, based on the information in this report.

# Executive Summary

From 2009 to 2018 the Pacific Gas and Electric Company (PG&E) pursued a 20-year license renewal for the nuclear power plant at Diablo Canyon with the Nuclear Regulatory Commission (NRC), an effort terminated in 2018 due to then-projected energy demands and economic factors. In 2022 the State of California directed the California Public Utilities Commission (CPUC) to direct PG&E to again pursue license renewal to the year 2030. Subsequently, members of the San Luis Obispo Mothers for Peace (SLOMFP), the Friends of the Earth (FOE), and Mr. Bruce Severance (a member of the public) placed before the Diablo Canyon Independent Safety Committee (DCISC) concerns regarding embrittlement of the Unit 1 reactor pressure vessel (RPV) and its continued operating safety. SLOMFP and FOE have expressed similar concerns to both the CPUC and NRC.

This is one of two reports prepared for the DCISC. The objective of this report is to evaluate the state of knowledge concerning the embrittlement of the Diablo Canyon Unit 1 reactor pressure vessel (RPV) and to review current safety evaluations performed by PG&E. The objectives of the companion report are to explain the current process for predicting material embrittlement and for establishing operating limits, and to address concerns raised by SLOMFP and FOE as well as by Mr. Bruce Severance.

This report reviews documents concerning PG&E's surveillance program, its embrittlement predictions for Unit 1, and the safety evaluations performed for both pressurized thermal shock and upper shelf energy. Supplemental analysis using more data than is required by the NRC and using recently proposed analysis techniques are performed to gain additional insights concerning the embrittlement condition of the Unit 1 RPV. Collectively these evaluations provide a basis to evaluate the need for and potential benefit of additional testing.

The information contained herein supports the following conclusions:

<u>Concerning Surveillance Testing</u>

- Surveillance requirements for the original 40-year license required testing of three capsules. This was completed in 2003 when Capsule V was withdrawn and tested. No further capsule testing was required by the 40-year license.
- Surveillance guidance during license renewal is established by the NRC. For the situation of Unit 1 testing of a fourth capsule is recommended between 40 and 60 years of operation. This will be achieved by PG&E's plan to test Capsule B, which is documented in its license renewal application. Withdrawal of Capsule B has been planned for the next refueling outage and should be completed before 2028 to be consistent with NRC guidelines.

- Several deferrals of Capsule B testing, which was originally planned for 2009, were all acceptable because, as stated, the 40-year requirements were for three capsules.
- While it does not affect surveillance capsule testing guidance it is nevertheless reassuring to note that Unit 1 has, since 2011, had data for its limiting weld to a fluence exceeding that projected for 60 years of operation. When Capsule B is withdrawn the new data, which must be reported to the NRC within 18 months of the capsule withdrawal date, will have a fluence well beyond that of the RPV after 60 years. Once obtained, these new data may change the outcome of the structural integrity estimates for long term operation (i.e., to 60 years), which are discussed next.

<u>Concerning Safety Evaluations Performed to NRC Requirements</u>

- Embrittlement predictions made by PG&E for Unit 1 are accurate and compliant with NRC procedures, including the characterization of credibility of the Charpy impact toughness transition temperature shift ($\Delta T_{41J}$) data throughout Unit 1's operation.
- Based on data currently available, Unit 1 is not forecast to exceed the NRC's PTS screening criteria or the NRC's 68J screening criteria on Charpy upper shelf energy until sometime after 60-years of operation. Thus, Unit 1 currently satisfies NRC criteria associated with pressurized thermal shock and upper shelf energy through 60-years of operation.

<u>Concerning Safety Evaluations Performed Using Supplemental Techniques</u>

- Analyses were performed by this consultant using techniques supplemental to those now in regulatory and Code use to gain additional insights concerning the embrittlement condition of the Diablo Canyon Unit 1 RPV. These techniques make use of more data from other plants and more recently developed analytical techniques than now required by NRC. This information may inform DCISC and public judgments concerning the confidence that can be placed in existing techniques. Since these techniques are neither required nor currently endorsed by the NRC, this information has no impact on the licensing basis of Diablo Canyon Unit 1.
- For $\Delta T_{41J}$ and pressurized thermal shock, the supplemental analysis demonstrated that there is little likelihood that the plate material will become limiting during future operations. Using available direct fracture toughness (Master Curve) data for the Unit 1 limiting weld material demonstrated a conservatism of 20-30 °C in the current approaches adopted by the NRC.
- For upper shelf energy and the NRC's 68J screening criteria, the analysis forecasts that the Unit 1 RPV could fall below the 68J USE screening criteria during its license renewal period, likely sometime in 2029 or 2030. An equivalent margins analysis following Regulatory Guide 1.161 could be performed to demonstrate adequate safety margins for the RPV were it deemed necessary. Equivalent margins analyses performed on other reactors have, without exception, demonstrated that continued operation at USE values considerably below 68J is acceptable. In any event, the Unit 1 USE values

remain acceptable for the 5-year license extension period proposed by the State of California.

<u>Concerning Additional Testing</u>

- All analyses performed herein that follow current NRC requirements show PG&E has correctly assessed the Unit 1 RPV, and it meets all current regulatory requirements to 60 years of operation. As such, there is no need for testing to collect additional data at the current time. When Capsule B is tested, which is currently planned to occur after its withdrawal during the next refueling outage, those data will be considered with existing data could alter predictions for long term operation (i.e., to 60 years). If the new analysis suggests a degree of embrittlement that exceeds regulatory screening criteria for pressurized thermal shock or upper shelf energy before 60 years, compensatory actions would be required by NRC regulations. These actions may include changes to plant operating practices, performance of plant-specific analyses (for example using the alternate pressurized thermal shock rule and/or the NRC's guidance on assessment of low upper shelf steels) to demonstrate the adequacy of existing margins, collection of additional data, or some combination of all approaches. If additional data are collected, direct measurement of fracture toughness would be advisable as such data can be most clearly interpreted using existing regulatory and ASME Code procedures.

It should be recognized that NRC screening criteria do not represent failure conditions, but rather situations of very low failure probability in which further analysis, plant modifications, or additional data are used to demonstrate the maintenance of adequate safety margins, with high confidence. As such, an assessment that forecasts a screening criterion will be passed in the future is not a cause for alarm but, rather, indicates that additional analyses and actions are needed. The NRC requires these analyses and actions be completed three years before the screening criteria are passed.

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# Glossary & Acronyms

| Abbreviation | Definition |
|---|---|
| ASME | American Society of Mechanical Engineers |
| ASTM | American Society for Testing and Materials |
| ART | Adjusted Reference Temperature |
| CFR | Code of Federal Regulations |
| CMM or SRM | Correlation monitor material or standard reference material. These are samples of a common steel that is placed in the surveillance capsules of many plants as a quality control measure. |
| CPUC | California Public Utilities Commission |
| CVN | Charpy V-notch |
| DCISC | Diablo Canyon Independent Safety Committee |
| EMA | Equivalent margins analysis |
| ETC | Embrittlement trend curve |
| FOE | Friends of the Earth |
| NRC | Nuclear Regulatory Commission |
| P-T | Pressure-Temperature Limit |
| PTS | Pressurized thermal shock |
| RPV | Reactor pressure vessel |
| SLOMFP | San Luis Obispo Mothers for Peace |
| WOL | Wedge Opening Load fracture toughness specimen |

| Symbol | Definition |
|---|---|
| ¼-T | Quarter thickness: a location in the RPV wall one-quarter of the distance from the inner diameter toward the outer diameter |
| CF | Chemistry factor: quantified the radiation sensitivity of RPV steel. Based on either tables in [RG1.99R2] or a fit to surveillance data |
| EFPY | Effective full power years. This value counts only the time that a nuclear power plant is operating, excluding time during which the plant is down for outages and scheduled maintenance. |
| FF | Fluence factor: use in [RG1.99R2] |
| LF | Lead factor: defined in [ASTM E185] as "*the ratio of the average neutron fluence of the specimens in a surveillance capsule to the peak neutron fluence of the corresponding material at the ferritic steel reactor pressure vessel inside surface calculated over the same time period.*" |
| %drop | Percentage drop of USE from the unirradiated value |

| Symbol | Definition |
|---|---|
| $T_{41J}$ | The temperature at which the average Charpy energy is 41J (30 ft-lbs) |
| $RT_{NDT}$ | Index temperature that is based on Charpy and nil-ductility temperature tests that is used with the ASME $K_{Ic}$ curve. |
| $RT_{NDT(u)}$ | An unirradiated value of $RT_{NDT}$ |
| $T_0$ | Index temperature of the $K_{Jc}$ fracture toughness curve determined according to ASTM testing standard E1921. At $T_0$ the median fracture toughness is 100 MPa$\sqrt{m}$. |
| USE | Average upper shelf energy of the Charpy transition curve |
| 68J | USE screening criteria from [10CFR50-AppG] |

# 1. Background and Objective

In 2009 the Pacific Gas and Electric Company (PG&E) submitted its plans for license renewal of both units at the Diablo Canyon site [Diablo LRA 2009] with the Nuclear Regulatory Commission (NRC) under 10 CFR Part 54 [10CFR54]. If approved by the NRC a license renewal authorizes a 20-year renewal to the existing operating license[1]. In 2018 PG&E informed the NRC it wished to withdrawal that license renewal application due to the then-projected energy demands and other economic factors in California [DCL-18-015]. The California Public Utilities Commission (CPUC) approved this decision to terminate the license renewal application [CPUC 2018]. However, in 2022 Senate Bill No. 846 was passed in the State of California [C-Senate 2022]. This bill invalidated the decision documented in [CPUC 2018] and directed the CPUC "*to set new retirement dates for the Diablo Canyon powerplant … conditioned upon the United States Nuclear Regulatory Commission extending the powerplant's operating licenses.*" Consequently, in October 2022 PG&E informed the NRC that it wished to re-initiate the license renewal application it withdrew four years earlier [DCL-22-085]. Currently the original 40-year operating licenses for Unit 1 and Unit 2 expire in 2024 and 2025, respectively. In November 2023 PG&E submitted a new license renewal application to the NRC, which was accepted by the NRC as sufficient in December of that year [NRC 2023a]. The NRC's determination allows PG&E to operate past the end of its current operating licenses while its license renewal application is under review.

Following the 2022 decision to pursue a license renewal, members of two organizations, the San Luis Obispo Mothers for Peace (SLOMFP) and the Friends of the Earth (FOE), have placed before the Diablo Canyon Independent Safety Committee (DCISC) their concerns regarding the state of embrittlement in the Unit 1 reactor pressure vessel (RPV) and, consequently, its continued operating safety. Mr. Bruce Severance has also provided commentary and analysis

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[1] In the USA there is no legal limit to the time over which a nuclear power plant may be licensed to operate; six units have already extended their licenses from 60 to 80 years [NRC Renewals 80] and some discussions have been held concerning 100-year licenses [NRC 2021].

to the DCISC on numerous occasions. SLOMFP and FOE have participated in CPUC rulemaking addressing the extended operation of Diablo Canyon Unit 1 [SLOMFP-CPUC 2023] and have initiated legal proceedings with the NRC [SLOMFP-NRC 2023], expressing their concerns with the embrittlement condition of the Unit 1 RPV.

This report is one of two reports prepared for the DCISC. The purpose of these reports is to address the concerns expressed by SLOMFP and FOE in their presentations before the DCISC and in their legal filings with the CPUC and the NRC. The objective of this report (Part 2 of 2) is to independently evaluate the state of knowledge concerning the embrittlement of the Diablo Canyon Unit 1 reactor pressure vessel (RPV) and to review current safety evaluations performed by PG&E. The objectives of the companion report (Part 1 of 2) are to explain the current NRC process for predicting material embrittlement and for establishing operating limits, and to address concerns raised by SLOMFP and FOE in two documents [SLOMFP-CPUC 2023] and [SLOMFP-NRC 2023].

The remainder of this report is structured as follows:

- Chapter 2 summarizes the data now available that quantifies the embrittlement of the RPV in Diablo Canyon Unit 1, discusses future data collection plans, and reviews deferral of the withdrawal of Capsule B.
- Chapter 3 reviews the RPV safety evaluations of the Diablo Canyon Unit 1 RPV.
- Chapter 4 provides a supplementary analysis of Unit 1 embrittlement trends using new techniques; some are being considered by the American Society of Mechanical Engineers, but none have yet gained regulatory approval. These analyses provide additional insights concerning the embrittlement condition of the Diablo Canyon Unit 1 RPV and help inform judgments concerning the need, or not, for additional testing.
- Chapter 5 provides conclusions and recommendations.
- Chapter 6 provides a list of documents cited in this report.
- Chapter 7 provides the professional resume of Dr. Mark Kirk, DCISC's consultant and author of this report.

Upon its completion this report was reviewed by PG&E for factual correctness.

# 2. Embrittlement Data for the Limiting Weld Material in the Diablo Canyon Unit 1 RPV

This Chapter provides the following information:

- Section 2.1 describes the Diablo Canyon Unit 1 surveillance program.
- Section 2.2 describes the data obtained for the limiting material from the Diablo Canyon Unit 1 surveillance program and from surveillance programs conducted at other plants pertinent to evaluation of the continued operating safety of the Diablo Canyon Unit 1 RPV.
- Section 2.3 describes the capsule withdrawal schedule with a particular focus on the multiple modifications to the withdrawal plan for Capsule B since 2003.

## 2.1. Diablo Canyon Unit 1 Reactor Vessel Surveillance Program

### 2.1.1. Original Program

Construction on Diablo Canyon Unit 1 began in 1968. The reactor vessel surveillance program was designed to fulfill the requirements of ASTM E185-70 (the -70 designates the publication year: 1970) [ASTM E185-70]. The ASTM requirements in that standard, which remain in force as part of the Diablo Canyon Unit 1 license for the first 40 years of operation, can be summarized as follows (quotations, which appear in *italics*, are from [ASTM E185-70]):

- <u>Materials Sampled</u>: *"A minimum test program shall consist of specimens taken from the following locations: (1) base metal of one heat, incorporated in the highest flux location of the reactor vessel, that has the highest initial ductile-brittle transition temperature, (2) weld metal, fully representative of fabrication practice used for the welds in the highest flux location of the reactor vessel, (weld wire or rod, and flux must come from one of the heats used in the highest flux region of the reactor vessel), and (3) the heat-affected zone of the weldments noted above. Where possible, weld, weld heat-affected zone, and base-metal specimens shall be from the same test coupon."*

- Types of Specimens: *"Each set of test specimens from each different neutron exposure level shall consist of eight or more impact and two or more tension specimens from the base metal and the weld metal and eight or more impact specimens from the heat-affected zone."* Fracture toughness specimens were not required by ASTM until the 1979 revision of ASTM E185, nevertheless they were included in some early surveillance programs.
- Capsule Withdrawal Schedule: *"It is recommended that sets of specimens be withdrawn at three or more separate times. One of the data points obtained shall correspond to the neutron exposure of the reactor vessel at no greater than 30 percent of its design life. One other data point obtained shall correspond to the neutron exposure of the reactor vessel near the end of its design life."*

A Westinghouse report published in 1975 describes the design of the original surveillance program, consisting of five "Type I" and three "Type II" capsules [WCAP-8465]. Table 1 summarizes the specimen and material loading of both capsule types. The Type I capsules include samples from all intermediate shell plates used in fabrication of the RPV beltline but omit the weld whereas the Type II capsules monitor one of the intermediate shell plates along with the weld. The program includes eight capsules, far exceeding the minimum requirement of three. Fracture toughness (WOL) specimens and samples of a correlation monitor material are also included in the capsules even though these were not required by [ASTM E185-70].

Table 1. Original Diablo Canyon Unit 1 surveillance program [WCAP-8465].

| **Capsule Type** | Type I | Type II | | |
|---|---|---|---|---|
| **Capsule Designations** | T, U, X, W, Z | S, Y, V | | |
| **Lead Factors** | 3.57 for T & Z<br>1.17 for others | 3.57 for S & Y<br>1.17 for V | **Copper (wt%)** | **Nickel (wt%)** |
| **Intermediate Shell Plate B4106-1** | 8 CVN, 1 Tensile, 2 WOL | --- | 0.11 | 0.53 |
| **Intermediate Shell Plate B4106-2** | 8 CVN, 1 Tensile, 2 WOL | --- | 0.11 | 0.50 |
| **Intermediate Shell Plate B4106-3** | 8 CVN, 1 Tensile, 2 WOL | 8 CVN, 2 Tensile, 2 WOL | 0.077 | 0.46 |
| **Weld Heat 27204, Linde 1092 flux** | --- | 8 CVN, 2 Tensile, 2 WOL | 0.21 | 0.98 |
| **Heat Affected Zone** | --- | 8 CVN | --- | --- |
| **Correlation Monitor Material** | 8 CVN | 8 CVN | 0.14 | 0.68 |

**Notes**

- Table entries in the yellow shaded cells provide the numbers of specimens of each material in each capsule, including the following:
  - Charpy V-notch specimens (CVN)
  - Wedge Opening Load specimens (WOL), a type of fracture toughness specimen
  - Tensile specimens
- Lead factor information comes from [WCAP-11567].
- The practice of including a correlation monitor material, which was identified by [ASTM E185-70] as "desirable," provides for collection of embrittlement data using the same material in a wide variety of different operating reactors. All correlation monitor samples are taken from a single plate of well characterized RPV steel. Having a well-established reference material helps when assessing potentially anomalous results.

### 2.1.2. Supplemental Program

Diablo Canyon Unit 1 began commercial operation in 1985. The first surveillance capsule (Capsule S) was withdrawn, tested, and reported in 1987 after 1.25 effective full power years (EFPY) at a fluence of $2.83 \times 10^{18}$ n/cm$^2$ [WCAP-11567][2]. By the early 1990s it had become clear that the original surveillance program should be enhanced to better meet future operational goals and challenges. As explained by the 1992 Westinghouse report [WCAP-13440]:

> *"The original surveillance program for Diablo Canyon Unit 1 is adequate to monitor vessel embrittlement through 40 years of operation; however, its design does not accommodate operational periods significantly beyond 40 years and cannot supply all of the embrittlement data necessary to support a longer period of operation. PG&E determined that the existing surveillance program should be augmented through a supplemental surveillance program, which consists of additional capsules. The supplemental surveillance program will provide sufficient embrittlement data on the limiting materials to permit effective management of vessel embrittlement during the entire operating life of the vessel.*
>
> *The new surveillance program for Diablo Canyon Unit 1 incorporates both the existing surveillance capsules and the supplemental capsules and meets three goals: First, it provides embrittlement data through 48 effective full power years (EFPY) or approximately 60 years of operation. Second, it provides a "standby" capsule that will reside in a low lead factor location and be held in reserve should it be needed in the future. Third, it provides data that may be used to help demonstrate the effectiveness of thermal annealing, should that process be used at some future time."*

The report goes on to explain that the supplemental program included four new surveillance capsules designed A, B, C, and D. These capsules were installed at the end of fuel cycle 5 (5.86 EFPY) as part of the refueling outage when the second planned capsule, Capsule Y, was removed and tested. At that time two surveillance locations were vacant, the former locations of Capsule S that had been removed at the end of cycle 1 and of Capsule Y that had just been removed. Two of the supplemental capsules were placed in the former holders of Capsules S and Y. To make space for the other two supplemental capsules the original program Capsules T and Z were removed and stored. As explained in [WCAP-13440] it was considered appropriate to remove and not test Capsules T and Z because those capsules did "*not contain the limiting beltline materials (weld metal).*"

Table 2 lists the different specimen types and materials contained in the four supplemental capsules, which included the following:

- The plate material monitored in these capsules corresponds to the lower-shell of the Diablo Canyon Unit 1 beltline rather than samples taken from the intermediate shell used in the original program.

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[2] These values of EFPY and fluence are current best estimates based on all currently available information. They differ slightly from the values reported in [WCAP-11567] in 1985, which were 1.26 EFPY and $2.98 \times 10^{18}$ n/cm$^2$.

- Samples of the original surveillance weld were included in Capsule B and Capsule D. These samples are the broken halves of tested weld of HAZ samples from Capsule S, which was withdrawn in 1987. The broken halves that were made of weld metal were used because there was no remaining supply of the original surveillance weld material. Upon removal these samples can be "reconstituted" by welding tabs onto the end of the broken halves and then machining a notch. The reconstitution process is standardized and demonstrated to produce results comparable to specimens that have not been reconstituted [ASTM E1253-21].
- Samples from a so-called "surrogate" weld made from the same weld wire heat, same flux type, and same flux lot as the surveillance weld were included. As they were removed from a nozzle drop-out these welds were made using the same practices as the surveillance weld. A comparison of the copper and nickel contents of the "surrogate" weld and the surveillance weld (see Table 2) shows that the values lie within an uncertainty band typical of similar materials. The embrittlement sensitivity of the surveillance weld and the "surrogate" weld is therefore comparable.
- Samples from four welds (designated weld 8B, 9B, W7, and 72W) that had been supplied by the Electric Power Research Institute. These samples were included for research purposes and have no relationship to the materials used to fabricate the Diablo Canyon Unit 1 RPV.

Table 3 provides the status of all capsules that constitute the Diablo Canyon Unit 1 reactor vessel surveillance program [WCAP-18655].

Table 2. Diablo Canyon Unit 1 supplemental surveillance capsules [WCAP-13440].

| Material | Capsule A | Capsule B | Capsule C | Capsule D | Cu (wt%) | Ni (wt%) |
|---|---|---|---|---|---|---|
| Surrogate Weld from Nozzle Drop-Out. Weld heat (27204) and flux type (Linde 1092) same and flux lot (3714) as surveillance weld. | 15 CVN, 3 Tensile | 15 CVN, 3 Tensile | 30 CVN, 3 Tensile | 15 CVN, 3 Tensile | 0.22 | 1 |
| Lower Shell Plate B4107-1 | 15 CVN, 3 Tensile | 15 CVN, 3 Tensile | 15 CVN, 3 Tensile | 15 CVN, 3 Tensile | 0.13 | 0.56 |
| Correlation Monitor Material | 12 CVN | 8 CVN | --- | --- | 0.14 | 0.68 |
| Surveillance Weld (from Capsule S), Heat 27204, Longitudinal Beltline Weld | --- | 2 WOL, 9 Charpy Inserts | --- | 8 Charpy Inserts | 0.21 | 0.98 |
| Weld 8B (Linde 80) | --- | 8 CVN | 9 CVN | 12 CVN | --- | --- |
| Weld 9B (Linde 0091) | --- | 7 CVN | 9 CVN | 11 CVN | --- | --- |
| Weld W7 (Linde 80) | --- | 7 CVN | 8 CVN | 11 CVN | --- | --- |
| Weld 72W (Linde 0124) | --- | 8 CVN | 9 CVN | 12 CVN | --- | --- |

Table 3. Current status of Diablo Canyon Unit 1 surveillance capsules.

| Capsule | Program | Lead Factor | Installed before Fuel Cycle | Removed after Fuel Cycle | Removal Year | Status |
|---|---|---|---|---|---|---|
| S | Original | 3.46 | 0 | 1 | 1986 | Tested |
| Y | | 3.44 | 0 | 5 | 1992 | |
| V | | 2.26 | 0 | 11 | 2002 | |
| T | | 3.44 | 0 | 5 | 1992 | Stored |
| Z | | 3.44 | 0 | 5 | 1992 | |
| U | | 1.28 | 0 | TBD | TBD | Standby |
| X | | 1.28 | 0 | TBD | TBD | |
| W | | 1.28 | 0 | TBD | TBD | |
| A | Supplemental | 1.31 | 6 | TBD | TBD | |
| B | | 3.46 | 6 | 24 or 25 | 2025 | Planned Removal |
| C | | 3.46 | 6 | 12 | 2004 | Stored |
| D | | 3.46 | 6 | 12 | 2004 | |

**Notes**

- Stored capsules have been removed from the reactor. Samples in these capsules were not tested.
- Testing of standby capsules is not required by current regulatory requirements. They remain in the reactor for potential future use.

## 2.2. Surveillance Data Summary

The weld metal samples used for the Diablo Canyon Unit 1 surveillance program are taken from weld wire heat 27204. This weld wire heat was used to fabricate the longitudinal seam welds in the reactor vessel beltline. The surveillance weld samples have a copper and nickel contents of approximately 0.20 and 1.0 weight percent, respectively. As summarized in Table 4, these contents of the elements primarily responsible for the embrittlement sensitivity of RPV steels well represent the welds in the RPV. The circumferential weld, which is also in the RPV beltline, is less sensitive to irradiation damage as evidenced by its lower copper and nickel content (again see Table 4). Table 4 also shows that the copper and nickel contents of the various plates used to fabricate the RPV beltline are lower than that of welds, again indicating a lower sensitivity to irradiation damage. However, some of the plates have inferior unirradiated properties compared to the welds as evidenced by higher values of $RT_{NDT}$ and lower values of unirradiated USE. Using information from Table 4 along with data from the Diablo Canyon Unit 1 surveillance program [WCAP-15958], Figure 1 compares the embrittlement response of the plate and weld materials. In terms of transition temperature ($RT_{NDT}$), which is used in PTS and P-T limits assessment, the upper graphic panel shows the weld to be limiting (that is: has the highest $RT_{NDT}$ after irradiation) except at very low fluence values not relevant to current operations. For USE, which is used to assess compliance with the USE screening criteria of [10CFR50-AppG], the value after irradiation is much closer between plate and weld materials. Nevertheless, the weld is limiting (i.e., has a lower USE after irradiation) at the highest fluence for which surveillance data exists and is likely to remain so as greater fluence exposure should reduce the USE of the weld to a greater extent than the USE of the plate[3]. Likewise, materials

[3] Which material is limiting during the license renewal period, plate or weld, will be re-evaluated by PG&E as new surveillance data becomes available when Capsule B is tested.

in the extended beltline[4] are not limiting. These materials are of comparable chemistry to the beltline materials but are exposed to at most 20% of the beltline fluence after 54 EFPY [WCAP-17315]. The NRC requires evaluation of these materials, which PG&E has performed [WCAP-17315]. No extended beltline material comes close to being limiting and, consequently, are not evaluated here.

**Table 4. Information on the Diablo Canyon Unit 1 beltline materials taken from Tables 2.1-1 and 3.1-1 of [WCAP-17315].**

| Identifier | Copper (wt%) | Nickel (wt%) | Unirradiated $RT_{NDT}$ (°C) | $RT_{NDT}$ Type | Unirradiated USE (J) | Maximum Inner Diameter Fluence ÷ $1\times10^{19}$ n/cm² | | Maximum ¼T Fluence ÷ $1\times10^{19}$ n/cm² | |
|---|---|---|---|---|---|---|---|---|---|
| | | | | | | 32 EFPY | 54 EFPY | 32 EFPY | 54 EFPY |
| **PLATE MATERIALS** | | | | | | | | | |
| Intermediate Shell B4106-1 | 0.125 | 0.53 | -23 | NB-2331 | 157 | 1.23 | 2.02 | 0.73 | 1.20 |
| Intermediate Shell B4106-2 | 0.12 | 0.5 | -19 | NB-2331 | 155 | 1.23 | 2.02 | 0.73 | 1.20 |
| Intermediate Shell B4106-3 | 0.086 | 0.476 | -1 | Generic | 104 | 1.23 | 2.02 | 0.73 | 1.20 |
| Lower Shell B4107-1 | 0.13 | 0.56 | -9 | NB-2331 | 149 | 1.22 | 2.01 | 0.73 | 1.20 |
| Lower Shell B4107-2 | 0.12 | 0.56 | -7 | NB-2331 | 140 | 1.22 | 2.01 | 0.73 | 1.20 |
| Lower Shell B4107-3 | 0.12 | 0.52 | -30 | NB-2331 | 157 | 1.22 | 2.01 | 0.73 | 1.20 |
| Surveillance (B4106-3) | 0.086 | 0.476 | -- | -- | 160 | -- | -- | -- | -- |
| **WELD MATERIALS** | | | | | | | | | |
| Intermediate Shell Longitudinal 2-442 A, B, and C (Heat 27204) | 0.203 | 1.018 | -49 | Generic | 123 | A,B: 0.905 C: 0.463 | A, B: 1.49 C: 0.768 | A, B: 0.539 C: 0.267 | A, B: 0.888, C: 0.458 |
| Lower Shell Longitudinal 3-442 A, B, and C (Heat 27204) | 0.203 | 1.018 | -49 | Generic | 123 | A, B: 0.721 C: 1.22 | A, B: 1.19 C: 2.01 | A, B: 0.43 C: 0.727 | A, B: 0.709 C: 1.198 |
| Intermediate to Lower Shell Circumferential 9-442 (Heat 21935) | 0.183 | 0.704 | -49 | Generic | 148 | 1.22 | 2.01 | 0.73 | 1.20 |
| Surveillance (Heat 27204) | 0.198 | 0.999 | -- | -- | 123 | -- | -- | -- | -- |

**Notes**

- NB-2331 means that the unirradiated $RT_{NDT}$ values were determined from nil-ductility tests and Charpy V-notch tests according to the requirements of Section XI of the ASME Code, Article NB-2331.
- Generic $RT_{NDT}$ values are averages estimated by the NRC for welds made with different weld wire flux types [Vagins 1982]. [10CFR50.61] permits the use of generic values, along with an appropriate margin term, when there are no direct measurements of the unirradiated $RT_{NDT}$ values.

[4] The "extended beltline" is the region of the RPV above and below the active core where the neutron fluence is projected to exceed $1\times10^{17}$ n/cm² [NRC 2014]. Embrittlement measurable by the Charpy parameters $\Delta T_{41J}$ and USE can begin to be detected above this fluence.

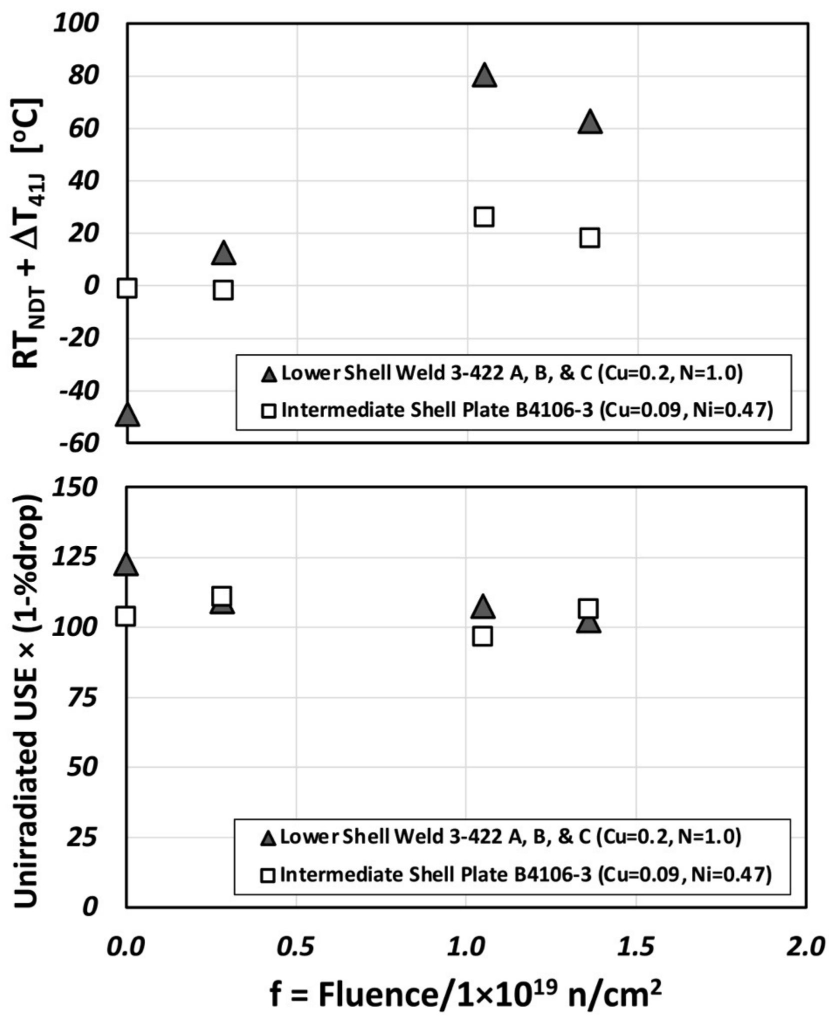


Figure 1. Comparison of the response to embrittlement of the plate and weld materials used to fabricate the Diablo Canyon Unit 1 RPV.

The evaluations performed in Chapters 3 and 4 of this report focus on assessment of the weld material which, as just explained, most constrains the continued safe operability of Diablo Canyon Unit 1. Table 5 summarizes surveillance data pertinent to this material, which includes data collected as part of the Diablo Canyon Unit 1 RPV surveillance program as well as information on the same weld wire heat from supplemental surveillance capsules irradiated at the Palisades Nuclear Power Plant in Michigan. As explained in the Part 1 companion report [Kirk 2024], in the PTS rule [10CFR50.61] the NRC requires consideration of all surveillance data available from a particular heat of material in evaluations of the plant's continued operating safety regardless of the plant in which such materials were exposed to radiation. This NRC requirement treats such "sister plant" materials in the same way as surveillance data obtained directly from the plant in question for analysis of transition temperature shift ($\Delta T_{41J}$) data. The information in Table 5 demonstrates that the material samples from the Palisades surveillance program have nearly identical copper and nickel content to the samples from Diablo Canyon

Unit 1. It is therefore appropriate to consider the Palisades data as part of the Diablo Canyon Unit 1 safety evaluation. Questions concerning the potential effect of somewhat different irradiation environments in Palisades versus Diablo Canyon Unit 1 on the embrittlement experienced by these samples is addressed in the Part 1 report, see Section 3.2.4 of [Kirk 2024].

Information beyond that summarized in Table 4 and Table 5 needed to support the analyses performed in Chapters 3 and 4 includes the operating temperature for Diablo Canyon Unit 1 and the so-called "best estimate[5]" chemistry for the limiting weld. [WCAP-17315] gives the operating temperature as 281 °C and the best-estimate chemistry as 0.203 weight percent copper and 1.018 weight percent nickel.

**Table 5. Summary of surveillance data for the Diablo Canyon Unit 1 limiting weld material, weld wire heat 27204 [WCAP-15958, BAW-2341].**

| Plant | Capsule | Fluence ÷ $1\times10^{19}$ n/cm$^2$ | Time Averaged Cold Leg Temperature (°C) | $T_{41J}$ (°C) | USE (J) | Cu (wt%) | Ni (wt%) |
|---|---|---|---|---|---|---|---|
| Diablo 1 | -- | 0 | -- | -54.2 | 123.4 | 0.21 | 0.98 |
| Diablo 1 | S | 0.283 | 284.4 | 7.3 | 109.8 | 0.21 | 0.98 |
| Diablo 1 | Y | 1.05 | 283.3 | 75.0 | 81.3 | 0.21 | 0.98 |
| Diablo 1 | V | 1.36 | 282.8 | 57.5 | 89.5 | 0.21 | 0.98 |
| Palisades | -- | 0 | -- | -40.7 | 147.0 | 0.194 | 1.067 |
| Palisades | SA-60 | 1.50 | 279.4 | 99.9 | 71.9 | 0.194 | 1.067 |
| Palisades | SA-240 | 2.38 | 280 | 108.1 | 72.9 | 0.194 | 1.067 |

## 2.3. Capsule Withdrawal Schedule

The third surveillance capsule, Capsule V, was withdrawn from Diablo Canyon Unit 1 and tested in 2002. This testing completed the surveillance requirements for the first 40 years of operation of the Unit 1 reactor, which was designed to an ASTM E185-70 surveillance program (see Section 2.1.1). Since 2003, the scheduled date for the next planned withdrawal (Capsule B) has been deferred on several occasions for different reasons. This section reviews these deferrals and evaluates their appropriateness.

### 2.3.1. Summary of Withdrawal Schedule Changes since 2003

The original schedule for Capsule B withdrawal was established in 2003 by Table 7-1 of the Capsule V surveillance report [WCAP-15958]; indicating Capsule B removal and testing after 20.7 EFPY at a fluence of approximately $2.91\times10^{19}$ n/cm$^2$, this corresponding to the end of fuel cycle 15 which was projected to occur in 2009. However, in 2008 PG&E requested approval to

[5] The "best estimate" chemistry is defined in [Wichman 1998]; it is an average of all chemistry data available for the material in question.

instead withdraw and test Capsule B at the end of fuel cycle 16, which would occur in 2010 after 21.9 EFPY of operation. In [DCL-08-021] PG&E stated the following:

> *NUREG-1801 requires that a licensee pursuing license renewal, and not crediting alternative dosimetry, must have a reactor vessel surveillance program consisting of a vessel material coupon that has fluence exposure equivalent to 60 years of operation. ... The current withdrawal schedule for Unit 1 does not meet the NUREG-1801 requirements for license renewal. A change is requested in the removal time for Capsule B to accommodate NUREG-1801 compliance.*

These statements say that Capsule B needed slightly more fluence to reach the then-forecast vessel fluence after 60 years of operation, which would occur if license renewal from 40 to 60 years was approved by the NRC. The NRC found this acceptable, stating in [NRC 2008] the following:

> *The withdrawal and testing of Capsule V … fulfilled the third and final recommendation of ASTM E 185-70 for the current Diablo Canton Unit 1 operating license. Capsule V, upon removal and testing at 32.3 EFPY, had an accumulated neutron fluence of $1.37\times10^{19}$ n/cm$^2$, which is representative of the RPV fluence value at end of license (EOL). Therefore, the proposed delayed removal of Capsule B does not deviate from the licensee's current RPV materials surveillance program requirements. The change in withdrawal schedule for Capsule B to be withdrawn at a fluence approximately equivalent to 75.8 EFPY for the RPV will provide high fluence data of the RPV useful for license renewal application.*

In these statements the NRC affirms that Diablo Canyon Unit 1 needed to test three surveillance capsules to satisfy the terms of its original 40-year license, and that the removal and testing of Capsule V in 2003 satisfied this requirement. The NRC validated PG&E's claim that deferring withdrawal of Capsule B for one more fuel cycle would provide data representative of an operating time somewhat beyond that needed for a 40- to 60-year license renewal.

In 2010 PG&E requested another one fuel cycle extension for Capsule B withdrawal because during refueling outage 16 PG&E learned that Capsule B was stuck. In [DCL-10-141] PG&E stated the following:

> *During 1R16, refueling personnel have not been able to remove the Capsule B access plug on the reactor core barrel flange. Removal of the access plug is required to gain access to the specimen capsule. Normally the plug is held in place by its own weight (approximately five pounds). Refueling personnel have applied over 2,000 pounds of force in attempts to remove the plug. The application of additional extraction force may result in damage and prevent the plug from being reinserted after the capsule is removed. If the plug itself is damaged or the hole is deformed, the vendor does not have a spare access plug and is not prepared to machine the hole in the flange of the core barrel during the current refueling outage. There is also a concern for introducing foreign material to the reactor vessel if personnel damage the plug or tool while attempting to remove the plug. Therefore, PG&E requests revision to the Unit 1 reactor vessel material surveillance program withdrawal schedule to allow withdrawal of Capsule B during the Unit 1 Seventeenth Refueling Outage (1R17). Removal of Capsule B during*

> *1R17 will ensure adequate time to allow for the appropriate tooling, materials, and contingency plans to be in place to remove and reinsert or replace the Capsule B access plug.*

The NRC granted this extension. In their reply they stated that PG&E had provided "*technically sufficient justifications for the delay*" [NRC 2010]. Thus, the revised plan was to withdrawal Capsule B during the 17th refueling outage, then scheduled for 2012 at which time the capsule would have been in the reactor for 23.2 EFPY. However, by 2011 PG&E had been asked to participate in an industry-wide coordinated surveillance program being organized by EPRI [MRP-326, Server 2014]. The coordinated program recognized the critical industry need to collect surveillance data at higher fluences to better inform embrittlement predictions for both first (60-year) and second (80-year) license renewals that were then being considered by some plants. As part of the coordinated program many plants were asked to defer capsule withdrawals when those deferrals did not interfere with licensing commitments. EPRI determined, and PG&E confirmed this to be the case for Capsule B in Unit 1. Consequently, in [DCL-11-122] PG&E requested a schedule change with the NRC, stating in part the following:

> *EPRI … has recommended that Diablo Canyon Unit 1 delay the removal and testing of Capsule B until approximately twice the 60-year fluence. This is estimated to occur during the Unit 1 23rd refueling outage (1R23), which is scheduled for May 2022. The recommended delay has been proposed to support data acquisition for the EPRI Coordinated Reactor Vessel Surveillance Program. … Diablo Canyon Unit 1 has withdrawn and tested three capsules from Unit 1 that meet the three recommendations of ASTM E 185-70. ... The change in withdrawal schedule allows Capsule B to be withdrawn at a fluence of approximately 93.9 EFPY for the reactor pressure vessel.*

In their response the NRC stated that "*the revised surveillance capsule withdrawal date for Surveillance Capsule B for DCPP, Unit 1, is acceptable because withdrawal and testing of capsule V during the Unit 1 11th refueling outage fulfilled the third and final recommendation of ASTM E185-70 for the current DCPP Unit 1 operating license*" and that "*removing Surveillance Capsule B during the 23rd refueling outage is in accordance with 10 CFR Part 50, Appendix H, and will meet the recommendations of NUREG-1801, Revision 2, "Generic Aging Lessons Learned (GALL) Report.*"

The 23rd refueling outage occurred in early 2022. At that time Diablo Canyon Unit 1 had suspended its pursuit of license renewal and was planning to shut down permanently in 2024. Consequently, the withdrawal and testing of Capsule B would no longer be needed, so the Capsule was not removed during the outage. However later in 2022 (September) the State of California reversed its previous position and the Public Utilities Commission directed PG&E to seek license renewal for Diablo Canyon [C-Senate 2022]. Consequently, in [DCL-23-038] PG&E requested a schedule change with the NRC, stating in part the following:

> *The Unit 1 reactor pressure vessel fluence data is now needed for license renewal and PG&E requests revision to the Unit 1 reactor vessel material surveillance program withdrawal schedule to allow withdrawal of Capsule B during the Unit 1 Twenty-Fourth Refueling Outage (Fall 2023) or Unit 1 Twenty-Fifth Refueling Outage (Spring 2025).*

The NRC approved this request, again stating that the licensee had fulfilled its surveillance commitments for the original 40-year licensing period [NRC 2023b]. During the 24th refueling outage, which occurred late in 2023, PG&E's contractor, Westinghouse, again was unable to removed Capsule B due to the problems first identified in 2010 [DCL-10-141]. Current plans are therefore to prepare for removal of the stuck capsule during the 25th refueling outage in the Spring of 2025. During this outage PG&E has indicated that the reactor will be defueled and the core barrel removed, which will allow better access and more options for removal of Capsule B.

### 2.3.2. Evaluation

Testing of Capsule B to satisfy the NRC's recommendations for surveillance monitoring depends on if Unit 1 holds only its original 40-year license or if it is pursuing or has been granted a license renewal to operate for 60 years.

Concerning surveillance requirements for the 40-year license, the key point is that the surveillance program requirements associated with the original 40-year license of Diablo Canyon Unit 1 were established in [ASTM E185-70] and were fulfilled by the testing of three capsules with one capsule having a fluence exceeding that expected after 40-years of operation. These requirements were satisfied by testing the third capsule, Capsule V, in 2003. No additional testing, of Capsule B or any other capsule, was ever required before the end of the 40-year license in 2024.

Surveillance recommendations during license renewal are expressed in [NUREG-1801][6], the NRC's guidelines for long-term aging management. These guidelines state, in part, the following:

> *Reactor vessel beltline materials will be monitored by a surveillance program in which surveillance capsules are withdrawn from the reactor vessel and tested in accordance with ASTM E 185-82. … The surveillance program shall have at least one capsule with a projected neutron fluence equal to or exceeding the 60-year peak reactor vessel wall neutron fluence prior to the end of the period of extended operation. The program withdraws one capsule at an outage in which the capsule receives a neutron fluence of between one and two times the peak reactor vessel wall neutron fluence at the end of the period of extended operation.*

The recommendations of [ASTM E185-82] differ from those of [ASTM E185-70]. Table 1 in [ASTM E185-82] recommends testing of a total of four surveillance capsules if the predicted $\Delta T_{41J}$ is between 56-111 °C at the vessel inner diameter surface at the end of license, which in this context is 60-years. Further [ASTM E185-82] recommends testing this capsule at a fluence between one- and two-times the estimated 60-year fluence. Figure 2, which appears on page 28, shows that the limiting weld in Diablo Canyon Unit 1 has a forecast $\Delta T_{41J}$ in this range. Since Diablo Canyon Unit 1 is now pursuing license renewal this recommendation suggests

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6 NUREG-1801 is an NRC report; it is not a law. As such the information in NUREG-1801 constitutes recommendations and guidance, not requirements.

that PG&E should test a fourth capsule prior to the end of the license renewal period and before that capsule reaches two-times the estimated 60-year fluence.

PG&E has stated its plan to remove and test Capsule B between 40 and 60 years, as documented on page B.2-95 of their 2023 license renewal application [DCL-23-038]. The target fluence for Capsule B at its planned withdraw in 2025 is $3.7 \times 10^{19}$ n/cm$^2$, which is between 1-2 times the 60-year fluence. Since the 60-year fluence is $2.01 \times 10^{19}$ n/cm$^2$, Capsule B should be tested before it reaches $4.02 \times 10^{19}$ n/cm$^2$ to be consistent with NRC guidance. Capsule B will close this fluence gap in about 2.7 EFPY, which is approximately two 18-month refueling cycles. Thus, to stay below the 2-times 60-year fluence recommendation of [ASTM E185-82] Capsule B should be tested by 2028 if it cannot be removed earlier.

As an alternative to testing Capsule B, it may be noted that Capsule D also contains samples of the limiting weld (see Table 2 and Table 3). Thus, if Capsule B remains stuck it may be possible to reinsert Capsule D, which was removed during the 12$^{th}$ refueling outage [WCAP-18655] and test it after it has reached a target fluence beyond 60 years. For example, if Capsule D can be reinserted in its old location, then it should exceed the 60-year fluence after ≈7-8 more EFPY, which would take at most 9 calendar years following the date of re-insertion.

The withdrawal and testing of Capsule B or Capsule D would satisfy [NUREG-1801] guidance, providing additional information to further refine embrittlement estimates.

While it does not affect surveillance capsule testing guidelines, it is nevertheless reassuring to note that Diablo Canyon Unit 1 has, since 2011, had data for its limiting weld to a fluence of $2.38 \times 10^{19}$ n/cm$^2$ (see Table 5). This exceeds the fluence estimated at both the RPV inner diameter ($2.01 \times 10^{19}$ n/cm$^2$) and at the ¼-T location ($1.20 \times 10^{19}$ n/cm$^2$) associated with 60 years (54 EFPY) of operation (see Table 4). When Capsule B is withdrawn the new data may change estimates of USE and $\Delta T_{41J}$. The magnitude and direction of these changes may affect the outcomes of the various RPV safety evaluations that will be discussed in Chapter 3.

# 3. RPV Safety Evaluations

As described in the Part 1 companion of this report [Kirk 2024], the NRC requires plants to perform three types of safety evaluations for the RPV that use as input information on embrittlement obtained by surveillance program testing: a pressurized thermal shock (PTS) evaluation to ensure adequate fracture toughness during certain severe operating events, an evaluation of pressure-temperature (P-T) operating limits to ensure adequate fracture toughness during routine operation, and an evaluation of upper-shelf energy for its compliance with the screening criteria set forth in [10CFR50-AppG]. PTS and P-T limits analyses both rely on a forecast of the adjusted reference temperature (ART) after irradiation, while the [10CFR50-AppG] screening criterion requires a forecast of the effects of irradiation damage on upper shelf energy (USE).

Forecasts of ART and USE both evolve with time as more information becomes available from a plant's surveillance program. The following sections provide this consultant's evaluation of how the ART and USE estimates have changed over time. Additionally, this consultant's estimates of ART and USE are compared with those of PG&E since 2003. The PG&E estimates between 2003-2011 have since been superseded due to the availability of additional data in 2011. However, the estimates from the 2003-2011 timeframe are evaluated here due to concerns expressed by SLOMFP, FOE, and Mr. Bruce Severance

regarding the appropriate treatment of "not credible" data during that time.

## 3.1. Forecast of Adjusted Reference Temperature (ART)

Section 2.4.2 of the Part 1 companion to this report [Kirk 2024] described the current procedure for estimating the effect of irradiation damage on $\Delta T_{41J}$, including consideration of plant-specific surveillance data. That estimate of $\Delta T_{41J}$ is used in the following equation to estimate ART:

$$ART = RT_{NDT(U)} + \Delta T_{41J} + 2\sqrt{\sigma_U^2 + \sigma_\Delta^2} \tag{3-1}$$

where

$RT_{NDT(u)}$ is the unirradiated value of $RT_{NDT}$. $RT_{NDT}$ may either be evaluated from Charpy and drop-weight test data as described in ASME SC-XI NB-2331 or it may be estimated using generic mean values prescribed by the NRC for certain classes of materials. [10CFR50.61] provides the following generic mean values: -17.8 °C for welds made with Linde 80 flux, and –48.9 °C for welds made with Linde 0091, 1092 and 124 and ARCOS B–5 weld fluxes [Vagins 1982].

$\Delta T_{41J}$ is the shift caused by irradiation damage in the Charpy V-notch transition temperature determined at a mean absorbed energy of 41 Joules (J). $\Delta T_{41J}$ is the product of a "chemistry factor" (CF) and a "fluence factor" (FF) and is estimated as described in Section 2.4.2 of the Part 1 companion of this report [Kirk 2024]. In this section ART values are calculated for comparison with the NRC's PTS screening criteria of 132 °C for axial welds. Consequently $\Delta T_{41J}$ is estimated using the fluence at the inner-diameter of the RPV.

$\sigma_U$ is the standard deviation for $RT_{NDT(u)}$. When an ASME SC-XI NB-2331 $RT_{NDT(u)}$ value is used then $\sigma_U$ is set to 0. If a generic mean $RT_{NDT(u)}$ value is used, then $\sigma_U$ = 9.4 °C.

$\sigma_\Delta$ is the standard deviation of residuals reported for the [RG1.99R2] ETC in [Randall 1986]. $\sigma_\Delta$ Has a value of 15.6 °C for welds and 9.4 °C for base materials.

In equation (3-1) the term $2\sqrt{\sigma_U^2 + \sigma_\Delta^2}$ in equation (2-1) is typically referred to as the "Margin" term.

### 3.1.1. Consultant's Evaluation

Figure 2 summarizes the estimated values of ART and how they change with fluence for various times during Unit 1's operating lifetime. These estimates are based on the surveillance data and other information summarized in Table 4 and Table 5, the procedures outlined in Section 2.4.2 of the Part 1 companion to this report [Kirk 2024] for treatment of credible *vs.* not credible data, and on equation (3-1) as just described. Working from the top to the bottom of the figure the different graphic panels show how the ART estimates have changed over time as more surveillance data have become available. On each panel:

- The solid black curve is the mean estimate of ART, i.e., equation (3-1) without the margin term.
- The dashed black curve is the upper bound estimate of ART, i.e., equation (3-1), which is used for comparison to the PTS screening criteria and in estimation of P-T limits.

- The circles are surveillance data from Table 5 for weld wire heat 27204.
- The magenta horizontal line is the [10CFR50.61] PTS screening criterion for a longitudinal weld, which is 132 °C (270 °F)
- The colorful vertical lines give the fluence on the RPV inner diameter at different times in the plant's lifetime and for the expected fluence of Capsule B when it is withdrawn in 2025.

Overall, the trend exhibited by the plant-specific surveillance data is well represented by the mean estimate of ART determined using the [RG1.99R2] ETC, which is expected for a steel with a higher copper content like weld wire heat 27204 [Widrevitz 2019]. Table 6 illustrates the changes in chemistry factor (CF) and Margin with time; CF and Margin are terms in equation (3-1) and are needed to estimate the curves on Figure 2. As required by the NRC's procedures, the CF and Margin values are higher when the data were judged not credible before 1993 and again between 2003-2011. These higher values ensure greater conservatism in the assessment. Figure 3 superimposes all the ART curves on a single graph to permit easier comparisons. Several features bear note:

- The highest fluence surveillance data now available correspond to a vessel fluence well beyond 60 years.
- The ART curves, which are compared to the PTS screening criteria and are used to estimate P-T limits, are always a conservative representation (i.e., over-estimate) of the data for weld wire heat 27204.
- The ART curves for periods when the data were judged not credible, before 1993 and again between 2003-2011, are identical and are the most conservative.
- The ART curves for when the data were judged credible, as they were between 1993-2003 and again after 2011, while closer to the surveillance data still provide a conservative representation.
- Irrespective of the data being judged credible or not credible, these predictions all indicate that weld wire heat 27204 is not forecast to exceed the PTS screening criteria until after 40 operating years.
- When the data are judged not credible, weld wire heat 27204 is predicted to exceed the PTS screening criteria between 40 and 60 operating years.
- When the data are judged credible, weld wire heat 27204 is predicted to exceed the PTS screening criteria after 60 operating years.

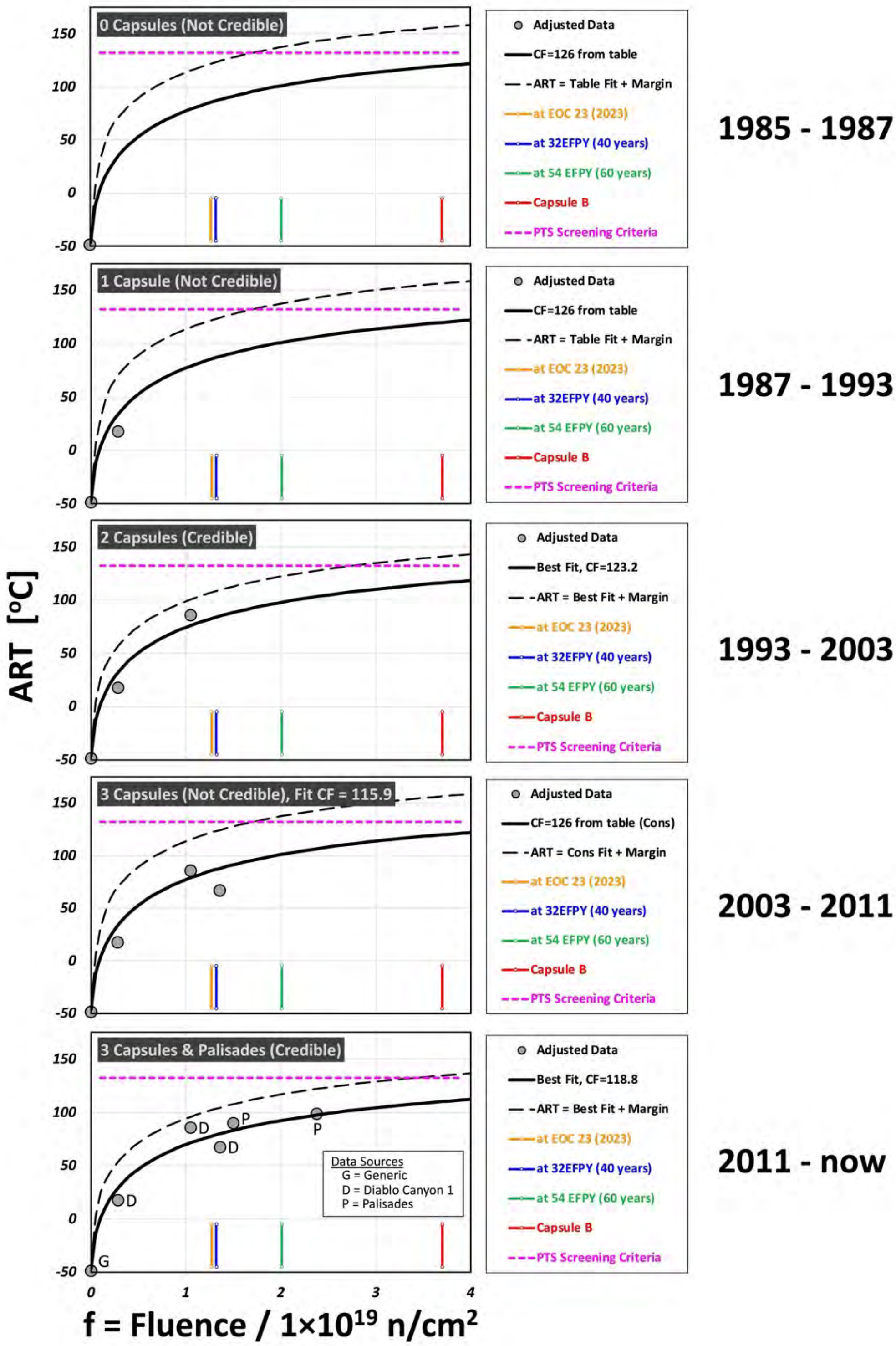


Figure 2. Evolution of ART predictions for the limiting weld in Diablo Canyon Unit 1 with time.

Table 6. Evolution of estimates of chemistry factor (CF) and margin for the limiting weld in Diablo Canyon Unit 1 with time.

| Year | Event | Number of $\Delta T_{41J}$ Data | Chemistry Factor (CF) in eq. (3-1)[°C] | Margin in eq. (3-1) [°C] | Are $\Delta T_{41J}$ Data Credible? |
|---|---|---|---|---|---|
| 1985 | Unit 1 Start | 0 | 126.0 | 36.5 | No |
| 1987 | Capsule S tested | 1 | 126.0 | 36.5 | No |
| 1993 | Capsule Y tested | 2 | 123.2 | 24.5 | Yes |
| 2003 | Capsule V tested | 3 | 126.0 | 36.5 | No |
| 2011 | Sister plant data from Palisades added to analysis | 5 | 118.8 | 24.5 | Yes |

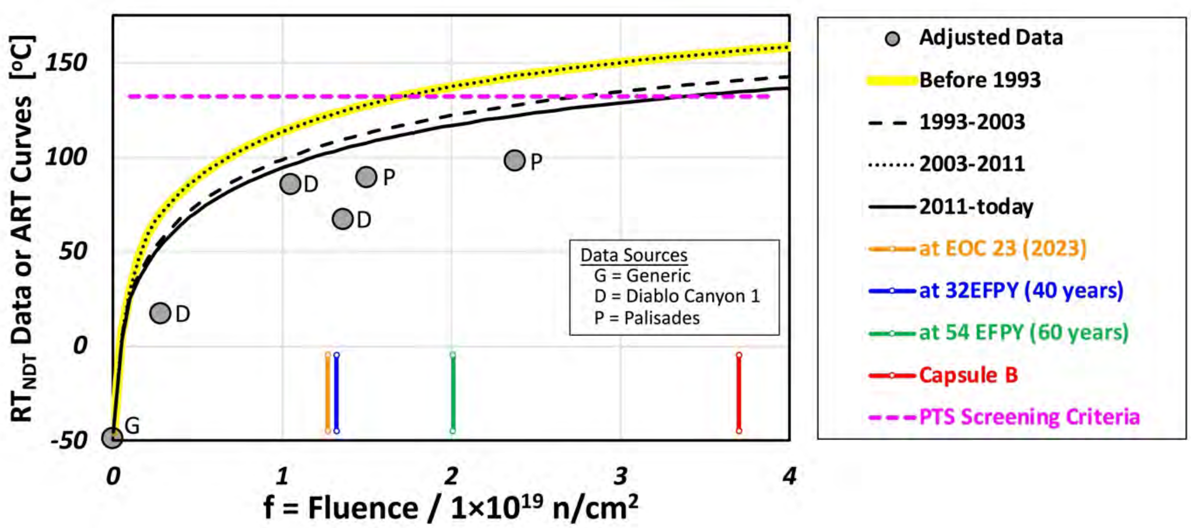


Figure 3. Summary of ART curves for different operating periods.

### 3.1.2. Comparison with PG&E Estimates Since 2003

Figure 4 compares the consultant's estimates of ART (the red and blue curves) presented previously as Figure 3 with those made by PG&E between the years 2003-2011 [Diablo LRA 2009] and after 2011 [WCAP-17315], these being represented by three diamonds. Between 2003-2011 the available surveillance data were deemed not credible while the addition of two data from the Palisades plant in 2011 made the data credible. This change in credibility status, which affects both the chemistry factor and the margin (see Table 6) is responsible for the decrease in estimated embrittlement that occurred in 2011 in both the consultant's and PGE's estimates. In both cases there is excellent agreement between the ART values calculated by PG&E and those calculated herein.

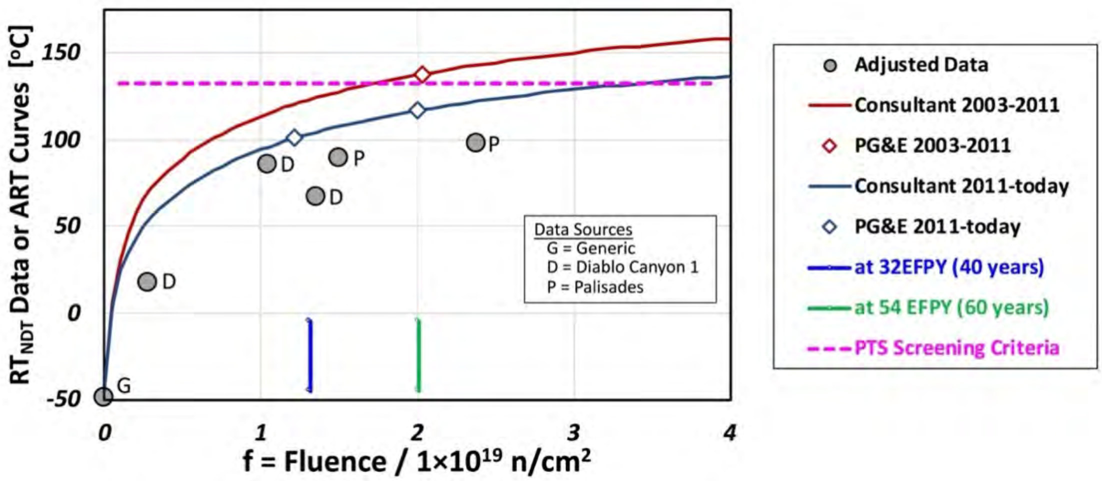


Figure 4. Comparison of consultant's ART estimates (curves) with those of PG&E (diamonds) for the timeframe 2003-2011 [Diablo LRA 2009] and 2011-today [WCAP-17315]. The NRC's required process for adjusting data is described in the Part 1 report, Section 2.4.2.2 [Kirk 2024].

Section 3.2.4.2 of the Part 1 companion report identified a potential non-conservatism of ≈ 2 °C in the NRC's temperature correction procedure that is used to adjust the data to a common temperature in diagrams like Figure 2, Figure 3, and Figure 4. The proximity of the current ART estimate (dark blue curve in Figure 4) and the PTS screening criteria (horizonal magenta line in Figure 4) permits an evaluation of importance of this potential non-conservatism. At 60 years the plant is estimated to have an ART value of 117 °C, which compares to the screening criteria of 132 °C, a 15 °C difference. Accounting for the potential non-conservatism would reduce this difference to 13 °C, but Diablo Canyon Unit 1 would still meet the PTS screening criteria to and beyond 60 years of operation. Thus, the non-conservatism does not alter any current conclusions concerning the licensable lifetime of Diablo Canyon Unit 1.

## 3.2. Forecast of Upper Shelf Energy (USE)

Section 2.4.3 of the Part 1 companion to this report [Kirk 2024] described the current procedure for estimating the effect of irradiation damage on USE, including consideration of plant-specific surveillance data. For USE there is no requirement to evaluate similar data from other ("sister") plants as was the case for $\Delta T_{41J}$ data (see Section 2.4.4.2 of the companion report). This is because [10CFR50-AppG] does not establish such a requirement. Additionally, when forecasting USE it is customary to assess compliance with the 68J screening criteria using the fluence at the quarter-thickness (¼T) location. This practice has been adopted by the industry for consistency with the flaw size used in the development of pressure-temperature (P-T) limits [10CFR50-App G] and for consistency with the flaw size used to demonstrate equivalent margins of safety if the USE is forecast to drop below the 68 J screening criteria [RG 1.161]. The NRC has not objected to this interpretation.

### 3.2.1. Consultant's Evaluation

Figure 5 compares the %drop values (circles) calculated from the Unit 1 surveillance data in Table 5 with the predictions of the [RG1.99R2] formula (see Part 1 report, Section 2.4.3.1) (black curve) [Kirk 2024]. The [RG1.99R2] prediction bounds all available data. Because these data are credible according to the [RG1.99R2] criteria, a value of α may be estimated from the data. As described in [Kirk 2024], the α value adjusts the NRC's generic USE prediction to ensure that all available %drop data are bounded (that is: over-estimated) by the adjusted prediction curve. The α value was determined to be -0.6 by forcing the [RG1.99R2] prediction to match the largest %drop result from Capsule V. The resultant prediction, accounting for the surveillance data, is shown by the red curve. In this case the surveillance data have very little impact on the prediction.

The history of the %drop prediction is as follows:

- 1985-1993, <2 surveillance data available: From the time of plant start (1985) to the time of Capsule Y withdrawal (1993) the black curve shows the prediction. Over this timeframe the data was not credible because there were less than 2 surveillance data available, so the NRC's formula is used without modification.
- 1993-2003, 2 surveillance data available: The USE data becomes credible in 1993 because two data are available. The USE prediction is that of the red curve, which is indexed to the Capsule Y datum (α=-0.6).
- 2003-today, 3 surveillance data available: The USE prediction is unaffected by the capsule V datum obtained in 2003 because the prediction remains indexed to the Capsule Y datum, as required by the NRC's [RG1.99R2] procedure. The red curve shows the USE prediction.

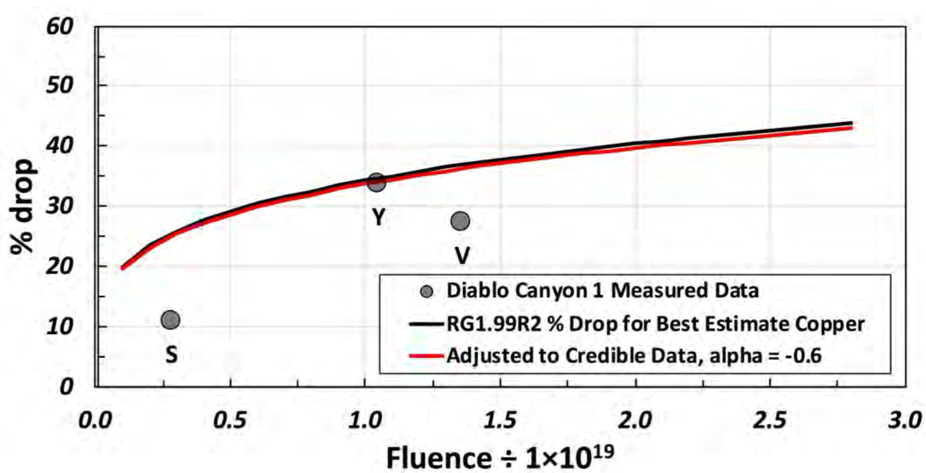


Figure 5. %drop values calculated from Diablo Canyon Unit 1 data for the surveillance weld.

### 3.2.2. Comparison with PG&E Estimates Since 2003

In 2003 Capsule V was withdrawn, however the Capsule V report [WCAP-15958] only contains a credibility analysis of the $\Delta T_{41J}$ data, not the ΔUSE data. That analysis determined the $\Delta T_{41J}$ data to be not credible. In the 2009 PG&E license renewal application [DiabloLRA 2009], which used the same data as reported in the Capsule V report, the ΔUSE data are also characterized as being not credible, which is incorrect. As described in the Part 1 report [Kirk 2024], [RG1.99R2] has different credibility criteria for ΔUSE data than it does for $\Delta T_{41J}$ data. The Regulatory Guide states "*even if the data fail this* [credibility] *criterion for use in shift calculations, they may be credible for determining decrease in upper-shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E 185-82.*" No statements were made in [DiabloLRA 2009] that the upper shelf could not be clearly determined, only that the ΔUSE data were not considered to be credible. This error was corrected in 2015 in response to a NRC request for additional information (see RAI 4.2.3-1 in [DCL-15-121]

In [DiabloLRA 2009] a value of USE for the limiting weld at 54 EFPY was given as 58.5 ft-lb (79 J) whereas the correct value should have been 59.2 ft-lb (80 J). This small error, which was in a conservative direction and, as noted, was corrected in 2015, had no effect on the assessment; both values are above the screening criteria of 50 ft-lb (68J). By 2011 this interpretation had been corrected; the USE data were characterized as being credible [WCAP-15958]. In the 2023 license renewal application [DCL-23-038] there is no change to the analysis from that provided in [WCAP-15958].

Figure 6 uses the %drop estimates for the limiting weld (red curve, Figure 5) to estimate the USE after irradiation and compares these estimates to the USE estimates at 32 and 54 EFPY provided in [WCAP-15958] and repeated in [DCL-23-038]. Agreement is excellent. These projections show that the USE of the limiting weld in Diablo Canyon Unit 1 is not forecast to fall below the 68 J screening criteria of [10CFR50-App G] until well after 60 years of plant operation.

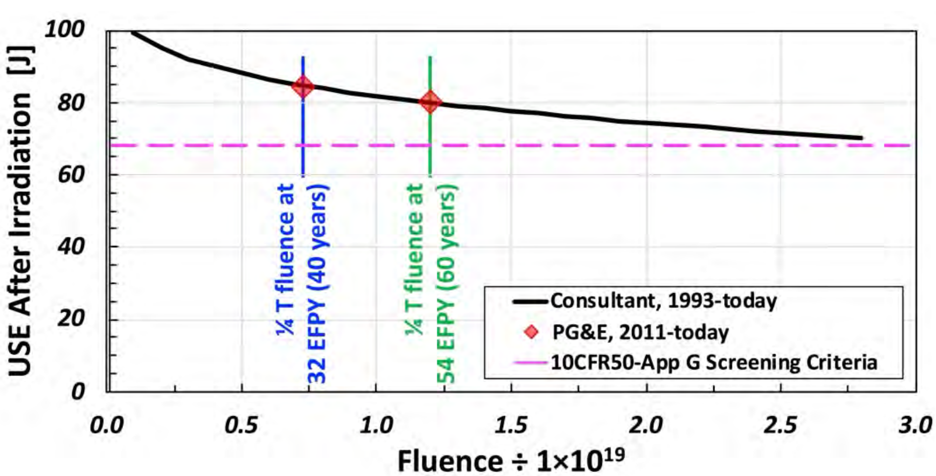


Figure 6. Comparison of consultant's and PG&E estimates of irradiated USE to the NRC's USE screening criteria.

## 3.3. Summary

The information in this Chapter shows excellent agreement between this consultant's estimates of ART and those of PG&E since 2011. ART is used to assess compliance with the NRC PTS screening criteria [10CFR50.61] and to calculate P-T limits following the requirements of [10CFR50-AppG]. The information in this Chapter also shows excellent agreement between the consultant's estimates of USE and those of PG&E since 2011. USE is used to assess compliance with the NRC USE screening criteria [10CFR50-AppG]. Using currently available information, these calculations forecast that Diablo Canyon Unit 1 will remain compliant with all evaluated NRC regulations through the end of a 60-year license.

# 4. Supplemental Analysis of Unit 1 Embrittlement Trends

This chapter uses techniques supplemental to those now in regulatory and Code use to gain additional insights concerning the embrittlement condition of the Diablo Canyon Unit 1 RPV. These techniques make use of more data and more recently developed analytical techniques than used in analyses performed to satisfy NRC regulatory requirements. It is hoped that this information may inform DCISC and public judgments concerning the confidence that can be placed in existing techniques and the need, or not, for additional testing, additional analysis, or plant modifications.

It should be emphasized that while the techniques used in this chapter evolved from currently accepted practices and in some cases are currently in codification review by ASME, none are currently endorsed or required by the NRC. Thus, the information presented in this chapter has no impact on the licensing basis of the Diablo Canyon Unit 1 nuclear power plant.

## 4.1. Analysis of Transition Temperature ($\Delta T_{41J}$) Data

Recently proposed techniques are used to characterize the embrittlement trends for the Diablo Canyon Unit 1 materials (Section 4.1.1) and to assess the impact of these trends on the continued operation of the Unit 1 RPV (Section 4.1.2).

### 4.1.1. Embrittlement Trends Exhibited by Similar Data

The SLOMFP and FOE concerns now before both the California Public Utilities Commission [SLOMFP-CPUC 2023] and the NRC [SLOMFP-NRC 2023] point out the value of obtaining additional data to better characterize the embrittlement experienced by the Diablo Canyon Unit 1 RPV and to estimate its impact on continued operations. As reviewed in Chapter 2, the amount of information plants are required by the NRC to collect as part of RPV surveillance programs is limited. Obtaining additional information is one way to assess the adequacy of existing data collection requirements and analytical methods.

As reviewed in this and the companion report [Kirk 2024], the NRC has long recognized the benefit of informing plant-specific embrittlement estimates using similar data obtained from other surveillance programs conducted at other plants (so-called “sister plants”). The NRC’s criteria for similarity of embrittlement sensitivity is that the “sister” data be obtained from the same heat of material irradiated in a similar neutron environment. This criterion is over three decades old [EPRI 1993, Wichman 1998] and dates from a time when the technical community had a more limited understanding of the causes of embrittlement. Recently a technique to identify data having similar embrittlement sensitivity using a large database of surveillance information collected by ASTM [ASTM Adjunct] has been proposed [Kirk 2022]. This technique, which is inspired by the machine learning method called “k-nearest neighbors” (kNN), ranks the similarity of materials in the ASTM database to a plant condition of interest based on the “distance” between a plant condition interest and other data in the database. As illustrated in Figure 7, the proposed distance is based on two chemistry variables (copper and nickel) and one environmental variable (irradiation temperature) that most significantly influence the irradiation damage sensitivity of RPV steels. [Kirk 2022] performed an extensive parametric study on different definitions of this similarity distance and concluded that this distance based on copper, nickel, and temperature provided a good means to identify similar data.

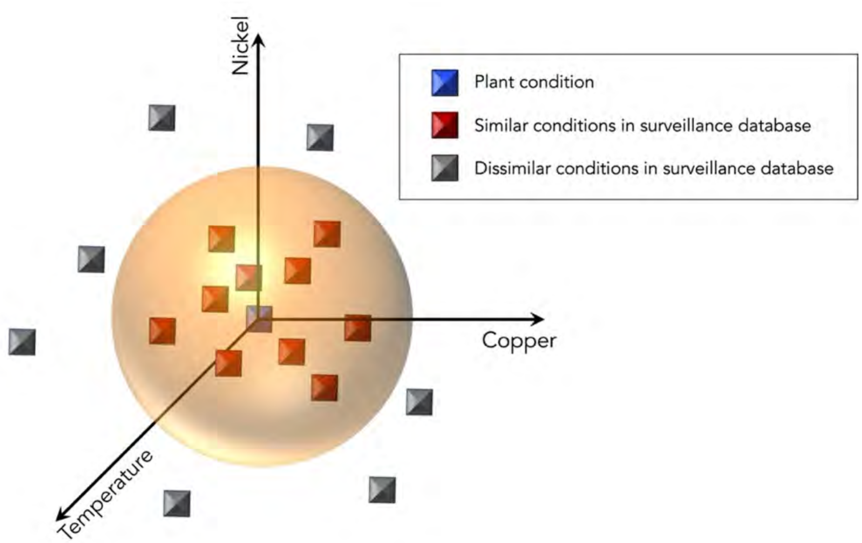


**Figure 7. Illustration of kNN approach for defining similar data, based on information from [Kirk 2022].**

In Figure 8 the technique described in [Kirk 2022] is used to identify data from the [ASTM Adjunct] database of similar irradiation sensitivity to the Diablo Canyon Unit 1 surveillance weld (left hand column) and surveillance plate (right hand column) materials. For the weld, similar data were identified in three plants other than Diablo Canyon (the Palisades plant and two

PWR plants from Germany); a total of 11 $\Delta T_{41J}$ data are available. For the plate, similar data were identified in 13 plants other than Diablo Canyon; a total of 51 $\Delta T_{41J}$ data are available. These data are available to fluences far above the value of $\approx 2\times10^{19}$ n/cm$^2$ that Diablo Canyon Unit 1 is expected to reach if it is licensed for 60-years and operates to the end of that term.

The top row of plots in Figure 8 compare these data to the predictions of the [RG1.99R2] Embrittlement Trend Curve (ETC) that the NRC requires while the bottom row of plots compare the data to the more recently developed ETC in [ASTM E900-15]. Comparison of the data to the curves supports the following conclusions for the Diablo Canyon Unit 1 materials:

- The embrittlement trend of the limiting weld material is slightly over-estimated by the [RG1.99R2] ETC while the [ASTM E900-15] ETC provides a better representation of the available weld data.
- The plate material continues to exhibit less embrittlement than the weld material to fluences much higher than Diablo Canyon Unit 1 would experience through 60-years of operation.
- The plate material is under-predicted by the [RG1.99R2] ETC beginning at a fluence of $\approx 2\times10^{19}$ n/cm$^2$. However, the degree of under prediction is not so significant that the plate material becomes more limiting than the weld material. The [ASTM E900-15] ETC provides a better representation of the available plate data.

These additional data provide increased confidence that the weld in Diablo Canyon will continue to be limiting to fluences much greater than the Unit 1 RPV is ever likely to experience. The data also provide confidence in the ability of the [ASTM E900-15] ETC to represent these data. The [ASTM E900-15] ETC is used as part of a draft assessment procedure now being considered by Section XI of the ASME Code, which is described in the next section.

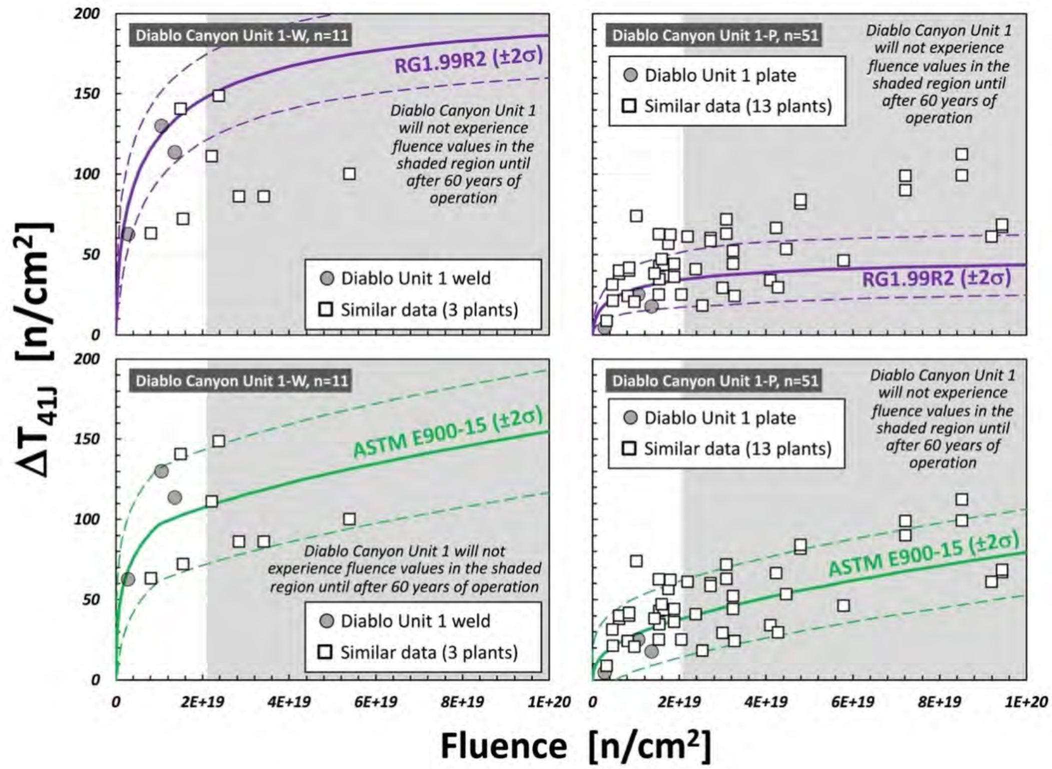


Figure 8. Data from the ASTM database of similar embrittlement sensitivity to the Diablo Canyon Unit 1 weld (left) and plate (right) compared to the predictions of the [RG1.99R2] ETC (top) and the [ASTM E900-15] ETC (bottom). For the weld Cu=0.196 and Ni=1.0 while for the plate Cu=0.081 and Ni=0.481 wt%.

### 4.1.2. Analysis following Draft ASME Code Case N-914

An effort is underway within the ASME Section XI Working Group on Operating Plant Criteria (WGOPC) to develop a comprehensive methodology for estimation of embrittlement for use in Code calculations [MPR-462]. Section XI includes methods to analyze the continued integrity of nuclear power plant components. For the RPV a means to account for the effects of embrittlement on the mechanical properties of vessel steels constitutes a key part of this analysis. Nevertheless, Code guidance on this topic remains imprecise.

In 2019 the WGOPC initiated an effort to develop a Code Case (CC N-914) to improve the clarity and comprehensiveness of current Code procedures, and to expand these procedures to address both conventional techniques based on Charpy and $RT_{NDT}$ [MRP-450] data as well as more current techniques based on the Master Curve and its fracture toughness transition temperature $T_0$ [MRP-418]. In 2021 EPRI produced a report providing a technical basis for the CC N-914 approach as well as draft Code Case language [MRP-462]. Further work by the WGOPC since 2021 has produced a revised draft to MRP-462, which recently went to ballot [MRP-462 Revision]. The WGOPC is currently working to improve the Code Case so it can be re-balloted sometime later in 2024. Once the Code Case is adopted by the ASME Code it will then be reviewed by the NRC as part of their recurring process to review the ASME Code Cases for inclusion in Regulatory Guide 1.147 [RG1.147]. Based on previous experience the Code Case balloting and NRC review process may take four to eight years to complete.

This section applies the draft method of CC N-914 described in [MRP-462 Revision] to estimate the variation of fracture toughness transition temperature with fluence for the Diablo Canyon limiting weld. To support this analysis, data from Table 4 and Table 5 are augmented by Master Curve data that is already available for the limiting weld (see Table 7). As reported in [MRP-127], unirradiated and irradiated $T_0$ values were determined using single edge notch bend, SE(B), specimens made from the Palisades weld wire heat 27204 for which Charpy data have been previously reported (see [BAW-2398] and [BAW-2341-2]).

Draft CC N-914 permits estimation of a value called $RT_{IRRAD}$. $RT_{IRRAD}$ and ART (see equation (3-1)) both represent an upper-bound estimate of the fracture toughness transition temperature for use in Code or regulatory calculations. $RT_{IRRAD}$ is estimated as follows:

$$RT_{IRRAD} = RT_{INPUT} + Shift + 2 \times Margin + Offset \tag{4-1}$$

CC N-914 is a data-driven approach that provides different ways to estimate the various terms in the equation depending on the data available (see [MRP-462] for full details). In this situation, Table 4, Table 5, and Table 7 summarize the available data. For this calculation the terms in the equation, consistent with the draft guidance of CC N-914, are as follows:

- $RT_{INPUT}$ is the unirradiated value of $RT_{To}$. $RT_{To}$ is estimated from the unirradiated value of $T_0$ from Table 7 plus a value of 19.4 °C. Thus, $RT_{INPUT}$ = -73.1 °C.
- Shift is the value of the ASTM E900-15 ETC (see Appendix) increased by a value $\eta$ = 14.7 °C. $\eta$ is the value needed to make the predicted shift values pass through the mean of the measured $\Delta T_{41J}$ data from Table 5 and the $\Delta T_0$ datum from Table 7. CC N-914 guidance says that for welds $\Delta T_{41J}$ and $\Delta T_0$ values are equivalent.
- Margin is a value that is 0.684 times the standard deviation of the ASTM E900-15 ETC (see Appendix). The value of 0.684 represents an uncertainty reduction allowable by the Code Case if there are three or more $\Delta T_{41J}$ and/or $\Delta T_0$ values. This is similar to the margin reduction allowed by [RG1.99R2] for credible data.
- Offset is a value of 10 °C that is added because the $RT_{INPUT}$ value was based on $T_0$ testing of SE(B) specimens.

Figure 9 shows the draft CC N-914 estimate of $RT_{IRRAD}$ and how it varies with fluence. This estimate is approximately 27 °C lower than the current ART estimate reported in Figure 3 that is based on currently accepted procedures. The primary source of this difference is the lower value of $RT_{INPUT}$ ($RT_{To}$ = -73 °C) compared with the conventional $RT_{NDT}$ value (-49 °C) used currently. This difference reflects the well-known over-conservatism associated with $RT_{NDT}$ [Kirk

2014] and demonstrates that Unit 1 is further from regulatory screening criteria than assessed using current NRC methods.

Table 7. Information on $T_0$ values for weld wire heat 27204 [MRP-127].

| Fluence ÷ 1× $10^{19}$ n/cm$^2$ | # specimens tested | # specimens meeting E1921 criteria | $T_0$ reported in [MRP-127] (°C) | $T_0$ from Consultant's re-calculation (°C) | Unirradiated $RT_{T_0}$ from Consultant's re-calculation [°C] | $\Delta T_0$ from Consultant's re-calculation (°C) |
|---|---|---|---|---|---|---|
| 0 | 9 | 9 | -93.0 | -92.5 | -73.1 | -- |
| 1.61 | 11 | 11 | -43.0 | -42.0 | -- | +134.5 |

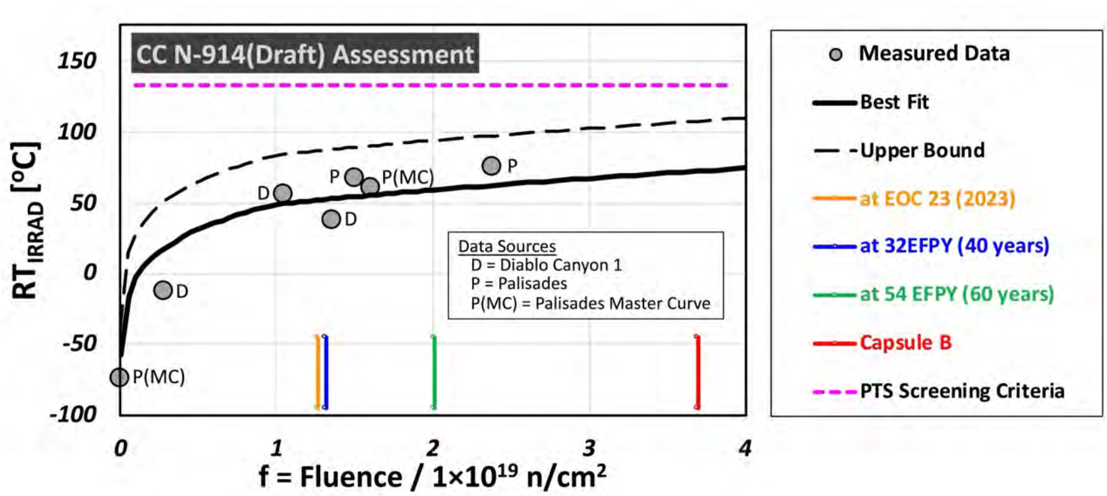


Figure 9. Estimate of the increase fracture toughness transition temperature with fluence for weld wire heat 27207 using ASME CC N-914 [MRP-462 Revision].

## 4.2. Analysis of Upper Shelf Energy (USE) Data

As discussed previously, [10CFR50-AppG] does not require the consideration of similar data from other ("sister") plants for ΔUSE data in the way that [10CFR50.61] requires such consideration for $\Delta T_{41J}$. Nevertheless, examination of similar data offers one way to assess the appropriateness of the embrittlement trends revealed by available plant-specific data for Diablo Canyon Unit 1. The similar data set identified in Section 4.1.1 using the kNN technique is again used to evaluate the embrittlement trends of steels similar in chemical content (copper and nickel) and exposure temperature to the limiting weld for the Diablo Canyon Unit 1 RPV. There are indications that the weld wire flux[7] may influence the unirradiated USE value [CEN-622]. Similar data identified by the kNN approach include that from Palisades, which is

[7] [CEN-622] describes welding fluxes as follows: "Fluxes used in submerged-arc welding are granular, fusible, mineral materials containing oxides of manganese, silicon, titanium, aluminum, calcium, zirconium, and magnesium, and other compounds. Some fluxes may contain intimately mixed metallic ingredients to deoxidize the weld pool or add alloying elements to the weld deposit, or both. The flux is deposited over the welding area and is melted by the heat of the arc. In the molten condition, the flux blankets the weld metal and shields the molten weld pool from atmospheric contamination."

identified as a sister plant for PTS analysis, as well as data from two German PWRs [May 2010, Hein 2010]. The Palisades weld has both the same weld wire heat (27204) as well as the same flux type (Linde 1092) as the Diablo Canyon Unit 1 data. The weld wire heat and flux type of the German PWR data is not reported in the literature. Nevertheless, the effect of weld wire flux is expected to be secondary to copper and nickel, which is why the German data is retained in this analysis.

Figure 10 uses these similar data to estimate percent USE drop and plots the data as a function of fluence. Also shown in blue is the prediction of a USE trend curve reported in 2010 [Kirk 2010], see the Appendix of this report for details about that trend curve. This trend curve was calibrated to a then-current set of USE surveillance data from the USA and, as such, is based on a much larger collection of data than is the NRC's RG1.99R2 formula (equations (2-2) to (2-4) in the Part 1 report [Kirk 2024]). In Figure 10 the heavy solid blue line represents the mean prediction from the 2010 work for the conditions of the Diablo Canyon weld and vessel (Cu=0.203, Ni=1.018, Mn=1.347, P=0.013, coolant temperature = 281.1 °C) while the dashed blue lines represent the $\pm 2\sigma$ tolerance bounds on this prediction. For normally distributed errors 95% of data should lie between these tolerance bounds. This trend provides a reasonable representation of the embrittlement trends revealed by the Diablo Canyon Unit 1 and similar data. Figure 10 illustrates that while the Palisades data may appear different than that from Diablo Canyon in this small dataset, the degree of difference is not uncharacteristic of the uncertainty in %drop data seen in a much larger population of data from many different RPV steels to which the [Kirk 2010] model was calibrated.

Based on the data in Figure 10 and the fact that the data similar in terms of chemical content to the Diablo Canyon Unit 1 weld is well represented by an embrittlement trend curve representative of a much larger data set, it is not unreasonable to assert that the Palisades data is representative of a similar steel under similar conditions to those experienced by the Diablo Canyon Unit 1 limiting weld.

To investigate the effect of considering the similar data, including that from Palisades, on the forecast of USE following the procedures of [RG1.99R2] the analysis of Section 3.2 was repeated, this time using the entire similar data set. Since the [RG1.99R2] procedure requires that its trend curve be adjusted to bound all available data, the [RG1.99R2] representation of this similar data set becomes the red curve shown in Figure 10. Figure 11 shows the resultant prediction, based on the red curve, of irradiated USE and how it decreases with increasing fluence. The blue and green vertical lines show the ¼T fluence values for the Unit 1 RPV at the end of 40 and 60 years of operation (32 and 54 EFPY, respectively). This analysis suggests that the Unit 1 RPV could fall below the [10CFR50-AppG] 68 J screening criteria sometime during its license renewal period. More specifically, the limiting weld in the Unit 1 RPV is forecast to fall below the 68J screening criteria at a fluence of $0.87\times10^{19}$ n/cm$^2$, which should occur at the ¼T after 38.5 EFPY of operation. Assuming a future capacity factor between 85-95%, 38.5 EFPY will be reached in the year 2029 or 2030.

Were it determined to be appropriate to consider all similar data as part of a USE analysis then an equivalent margins analysis performed following the guidance of Regulatory Guide 1.161

would be needed to demonstrate adequate safety margins for Diablo Canyon Unit 1 [RG1.161] after 2029-2030. The NRC would require such an analysis to be submitted three years before the plant is forecast to fall below the 68 J screening criteria [10CFR50-AppG]. Equivalent margins analyses performed on other reactors have, without exception, demonstrated that continued operation at USE values considerably below 68J is acceptable, see [WCAP-17651] and [ANP-3646] as examples.

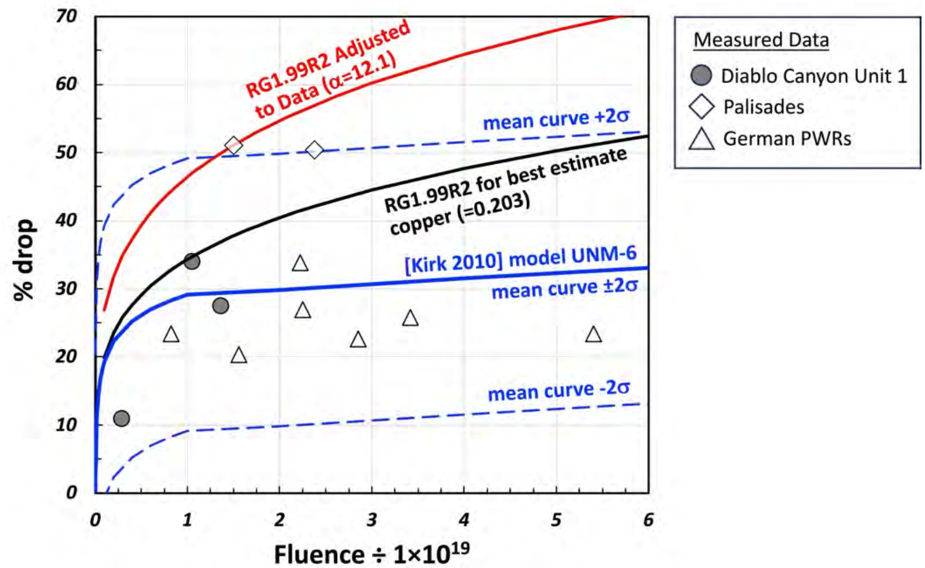


Figure 10. Estimates of percent USE drop determined from data having similar chemical composition and exposure temperature to the Diablo Canyon Unit 1 weld. The blue curves [Kirk 2010] was estimated for Cu=0.203, Ni=1.018, Mn=1.347, P=0.013, coolant temperature = 281.1 °C and illustrated the uncertainty in USE prediction typical of RPV steels. The black curve is based on the [RG1.99R2] formula whereas the red curve is adjusted to all data shown following the procedures of [RG1.99R2].

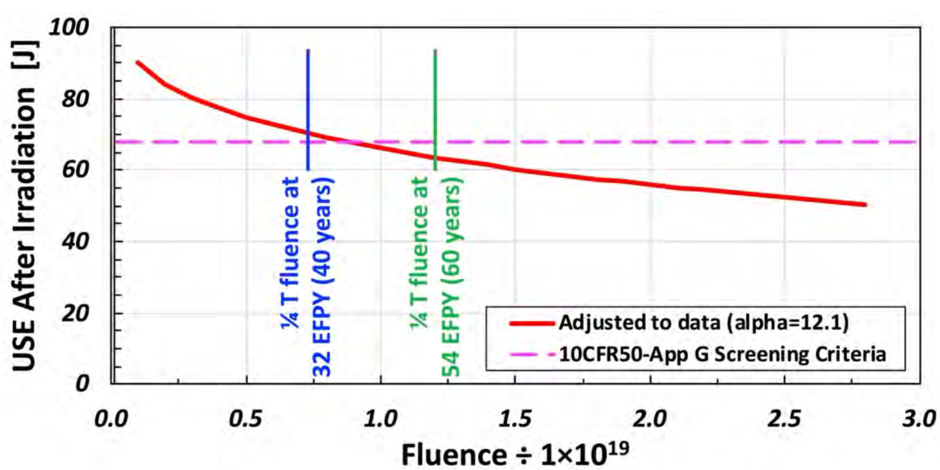


Figure 11. Supplemental USE analysis considering both the Diablo Canyon Unit 1 and Palisades data for weld wire heat 27204.

## 4.3. Summary and Evaluation of Need for Additional Data

This chapter used techniques supplemental to those now required to gain additional insights concerning the embrittlement condition of the Diablo Canyon Unit 1 RPV.

<u>Concerning analysis of transition temperature data</u>, which impacts compliance with the PTS screening criteria and evaluation of P-T limits, this chapter summarizes recently proposed techniques to characterize the embrittlement trends for the Diablo Canyon Unit 1 materials and to assess the impact of these trends on the Unit 1 RPV. Application of these techniques to data for the Diablo Canyon Unit 1 weld and similar materials demonstrated the following:

- The embrittlement trends for the limiting weld Diablo Canyon Unit 1 (Heat No. 27204, Flux Linde 1092) are, if anything, slightly over-estimated by the NRC's techniques, which is conservative.
- The plate material continues to exhibit less embrittlement than the weld material to fluences much higher than Diablo Canyon Unit 1 will experience during 60-years of operation. Thus, there is little likelihood that the plate material will become limiting during the plant's operational lifetime.
- Use of direct fracture toughness measurements demonstrates a conservatism of 20-30 °C in the current correlative approaches adopted by the NRC.

Collectively these observations provide increased confidence in the continued compliance of the Diablo Canyon Unit 1 RPV with the NRC's PTS screening criteria.

<u>Concerning the analysis of upper shelf energy data</u>, which impacts compliance with the USE screening criteria, a supplemental analysis was performed that considered the same set of surveillance data that are similar to the limiting weld wire heat in Diablo Canyon (including Palisades) as used for the transition temperature analysis. This analysis suggests that the Unit 1 RPV could fall below the 68J USE screening criteria sometime during its license renewal period, likely in 2029 or 2030. Were such an analysis determined to be appropriate, then an equivalent margins analysis following the guidance of Regulatory Guide 1.161 would need to be submitted to the NRC three years before the screening criteria is reached. Such analyses performed on other reactors have, without exception, shown that continued operation at a USE considerably below 68J is acceptable. For example, for RPVs of comparable thickness to Diablo Canyon Unit 1 adequate margins of safety have been demonstrated using conservative assumptions for loading and flaw size for USE values as low as 58J [WCAP-13587, NRC 1994]

The SLOMFP, FOE, and Mr. Bruce Severance have, in their various documents, frequently cited a preference for the collection of additional data to better inform decisions made concerning Diablo Canyon Unit 1. All analyses performed in Chapter 3, which follow currently required NRC practices, show that Diablo Canyon Unit 1 meets all current requirements to and beyond 60 years of operation. As such, there is no need for testing to collect additional data. The supplemental analyses presented in this Chapter, which use additional similar data and techniques not required by the NRC, provide even greater confidence in the ability of Diablo Canyon Unit 1 to satisfy the PTS screening criteria and have operable P-T limits to and beyond 60 years of operation. However, the supplemental analysis performed on USE suggests that

Diablo Canyon Unit 1 may fall below the USE screening criteria of 68J sometime in 2029 or 2030. This conclusion, if confirmed and validated by further analysis, would suggest the need to perform an "equivalent margins analysis" following the guidance of Regulatory Guide 1.161. New data could be collected to support such an analysis, but often it is not necessary to do so because correlative methods to estimate upper shelf fracture toughness from existing Charpy data are available and have regulatory acceptance.

Finally, as has been noted previously, when new data from Capsule B becomes available these data could affect existing predictions of both transition temperature and upper shelf. Depending on the degree to which the Capsule B data affect these predictions, particularly how they affect the years in which various screening criteria are forecast to be crossed, collection of additional data may be considered.

# 5. Conclusions

From 2009 to 2018 the Pacific Gas and Electric Company (PG&E) pursued a 20-year license renewal for the nuclear power plant at Diablo Canyon with the Nuclear Regulatory Commission (NRC), an effort terminated in 2018 due to then-projected energy demands and economic factors. In 2022 the State of California directed the California Public Utilities Commission (CPUC) to direct PG&E to again pursue license extension to the year 2030. Subsequently, members of the San Luis Obispo Mothers for Peace (SLOMFP), the Friends of the Earth (FOE), and Mr. Bruce Severance placed before the Diablo Canyon Independent Safety Committee (DCISC) concerns regarding embrittlement of the Unit 1 reactor pressure vessel (RPV) and its continued operating safety. SLOMFP and FOE have expressed similar concerns to both the CPUC and NRC.

This is one of two reports prepared for the DCISC. The objective of this report (Part 2) is to evaluate the state of knowledge concerning the embrittlement of the Diablo Canyon Unit 1 reactor pressure vessel (RPV) and to review current safety evaluations performed by PG&E. The objectives of the companion report (Part 1) are to explain the current process for predicting material embrittlement and for establishing operating limits, and to address concerns raised by SLOMFP and FOE.

This report reviews PG&E's embrittlement predictions, the surveillance capsule withdrawal schedule (including the several deferrals of withdrawal and testing of Capsule B), and the safety evaluations performed for both pressurized thermal shock and upper shelf energy. Also, supplemental analyses are performed using more data than is required by the NRC and using recently proposed analysis techniques, some of which are currently under codification review by the American Society of Mechanical Engineers. Collectively these evaluations provide a basis to evaluate the need for and benefit of additional testing.

The information contained herein supports the following conclusions:

## Concerning Surveillance Testing

- Surveillance requirements for the original 40-year license are established by the 1970 edition of ASTM surveillance standard E185, which requires testing of three capsules. This was completed in 2003 when the third surveillance capsule, Capsule V, was withdrawn from the reactor and tested. No further capsule testing was required by the 40-year license.
- Surveillance guidance during license renewal is established by the NRC in NUREG-1801, which references the 1982 edition of ASTM E185. For the situation of Diablo Canyon Unit 1, testing of a fourth capsule is recommended during the period of license renewal. This will be achieved by PG&E's plan to test Capsule B, which is documented in its license renewal application submitted to the NRC in 2023.
- The several deferrals of the testing date for Capsule B, which was originally planned for 2009, were all acceptable because, as stated, the 40-year surveillance program requirements were for three capsules, the testing of which was completed in 2003. Testing of Capsule B is recommended during the 40- to 60-year license renewal term. Withdrawal of Capsule B has been planned for the next refueling outage and should be completed before 2028 to be consistent with NRC guidelines.
- While it does not affect surveillance capsule testing guidance, it is nevertheless reassuring to note that Diablo Canyon Unit 1 has, since 2011, had data for its limiting weld to a fluence exceeding that projected for 60 years of operation. When Capsule B is withdrawn the new data obtained will have a fluence well beyond that of the RPV after 60 years. Once obtained these new data may change the outcome of the structural integrity estimates for long term operation (i.e., to 60 years), which are discussed next.

## Concerning Safety Evaluations Performed to NRC Requirements

- Embrittlement predictions made by PG&E for Unit 1 were checked and found to be accurate and compliant with NRC procedures, including the characterization of the $\Delta T_{41J}$ data available from 2003-2011 as being "not credible" while the $\Delta T_{41J}$ data available since 2011 is appropriately characterized as "credible."
- Based on data currently available, Unit 1 is not forecast to exceed the NRC's PTS screening criteria until sometime after 60-years of operation.
- Based on data currently available, Unit 1 is not forecast to fall below the NRC's 68J screening criteria on upper shelf energy until sometime after 60-years of operation.
- Thus, available data show that Unit 1 satisfies NRC criteria associated with pressurized thermal shock and upper shelf energy through 60-years of operation.

## Concerning Safety Evaluations Performed Using Supplemental Techniques

- Analyses were performed by this consultant using techniques supplemental to those now in regulatory and Code use to gain additional insights concerning the embrittlement condition of the Diablo Canyon Unit 1 RPV. These techniques, some of which are now being considered for inclusion in the ASME Code, make use of more data from other plants and more recently developed analytical techniques than now required by NRC. This information may inform DCISC and public judgments concerning the confidence that can be placed in existing techniques. Nevertheless, since these techniques are neither required nor currently endorsed by the NRC this information has no impact on the licensing basis of Diablo Canyon Unit 1.

- For $\Delta T_{41J}$ and pressurized thermal shock, the supplemental analysis demonstrated that the $\Delta T_{41J}$ embrittlement trend of the limiting weld is slightly over-estimated by the NRC's techniques, which is conservative. Additionally, the plate material should continue to exhibit less embrittlement than the weld material. Thus, there is little likelihood that the plate material will become limiting during the plant's operational lifetime. Using available direct fracture toughness data for the Unit 1 limiting weld material demonstrated a conservatism of 20-30 °C in the current approaches adopted by the NRC.
- For upper shelf energy and the NRC's 68J screening criteria the analysis shows that similar data from other plants are within expected uncertainty bounds, and forecasts that the Unit 1 RPV could fall below the 68J USE screening criteria during its license renewal period, likely sometime in 2029 or 2030. An equivalent margins analysis following the guidance of Regulatory Guide 1.161 could be performed to demonstrate the continued maintenance of adequate safety margins in this situation. Equivalent margins analyses performed on other reactors have, without exception, demonstrated that continued operation at USE values considerably below 68J is acceptable. Even considering this information, Diablo Canyon Unit 1's USE remains acceptable for the 5-year license extension period proposed by the State of California.

<u>Concerning Additional Testing</u>

- All analyses performed herein that follow current NRC requirements show that Diablo Canyon Unit 1 has been correctly assessed by PG&E and meets all current NRC requirements to and beyond 60 years of operation. As such, there is no need for testing to collect additional data at the current time. When Capsule B is tested, which is currently planned to occur after its removal during the next refueling outage, the data obtained will be combined with data already collected and analyzed following the NRC's procedures. This new data could alter existing predictions. If the new analyses suggest a degree of embrittlement that exceeds regulatory screening criteria before 60 years, compensatory actions would be required by NRC regulations. These actions may include changes to plant operating practices, performance of plant-specific analyses (for example using the alternate PTS rule and/or the NRC's guidance on assessment of low upper shelf steels) to demonstrate the adequacy of existing margins, collection of additional data, or some combination of all approaches. If additional data are collected, direct measurement of fracture toughness would be advisable as such data can be most clearly interpreted using existing regulatory and ASME Code procedures.

It should be recognized that NRC screening criteria do not represent failure conditions, but rather situations of very low failure probability in which further analysis, plant modifications, or additional data are used to demonstrate continued maintenance of adequate safety margins with high confidence. As such, an assessment that forecasts a screening criterion will be passed in the future is not a cause for alarm but, rather, indicates that additional analyses and actions are needed. The NRC requires these analyses and actions be completed three years before the screening criteria is passed.

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NRC 2008 NRC Letter to Diablo Canyon concerning Approval of Proposed Reactor Vessel Material Surveillance Capsule Withdrawal Schedule, 24 September 2008, ADAMS ML082380306.

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| NRC 2023a | "License Renewal Application; Acceptance for Docketing and Opportunity to Request a Hearing and to Petition for Leave to Intervene," December 14, 2023, NRC ADAMS Accession Number ML23341A003. |
| NRC 2023b | NRC Letter to Diablo Canyon Concerning Revision of the Reactor Vessel Material Surveillance Capsule Withdrawal Schedule, 20 July 2023. |
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| RG1.147 | Regulatory Guide 1.147, Revision 20, Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1, December 2021, https://www.nrc.gov/docs/ML2118/ML21181A222.pdf. |
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WCAP-13440 E. Terek, "Supplemental Reactor Vessel Radiation Surveillance Program for the Pacific Gas and Electric Company Diablo Canyon Unit No. 1," Westinghouse Electric Company, Report WCAP-13440, December 1992.

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| WCAP-18655 | E.T. Hayes, "Ex-Vessel Neutron Dosimetry Program for Diablo Canyon Unit 1, Cycle 22," Westinghouse Electric Company, WCAP-18655-NP Revision 0, August 2021. |
| Wichman 1998 | K.R., Wichman, M.A. Mitchell, and A.L. Hiser, "Generic Letter 92-01 and RPV Integrity Assessment: Status, Schedule, and Issues," Presentation to NRC/Industry Workshop on RPV Integrity Issues, February 12, 1998, ADAMS ML 110070570, https://www.nrc.gov/docs/ML1100/ML110070570.pdf. |
| Widrevitz 2019 | D. Widrevitz and M. Gordon, "Assessment of the Continued Adequacy of Revision 2 of Regulatory Guide 1.99," NRC Technical Letter Report TLR-RES/DE/CIB-2019-2, ADAMS ML19203A089, https://www.nrc.gov/docs/ML1920/ML19203A089.pdf. |

# 7. Mark Kirk – Resume

## Education

| Degree | Year Granted | University | Discipline |
|---|---|---|---|
| Ph.D. | 1992 | University of Illinois at Urbana-Champaign | Civil Engineering |
| M.S. | 1989 | University of Maryland at College Park | Mechanical Engineering |
| B.S. | 1984 | Virginia Polytechnic Institute and State University | Engineering Science and Mechanics |

## Employment

| Dates | Company Name & Location | Concluding Position Title |
|---|---|---|
| 2018-date | Central Research Institute of Electric Power Industry (CRIEPI)<br>*Yokosuka, Kanagawa, Japan* | Guest Research Fellow |
| 2018-date | Phoenix Engineering Associates, Inc. (PEAI)<br>*Unity, New Hampshire, USA* | Principal Engineer |
| 1999-2018 | Nuclear Regulatory Commission<br>*Rockville, Maryland, USA* | Senior Materials Engineer |
| 1997-1999 | Westinghouse Electric Corporation<br>*Pittsburgh, Pennsylvania, USA* | Senior Engineer |
| 1992-1997 | Edison Welding Institute<br>*Columbus, Ohio, USA* | Energy & Chemical Industry Team Leader |
| 1981-1992 | David Taylor Research Center (U.S. Navy)<br>*Annapolis, Maryland, USA* | Senior Project Engineer |

## Nuclear Power Skills and Experience

I began my work in nuclear power in 1997 with two years at the Westinghouse hot cells followed by nearly 20 years with the Nuclear Regulatory Commission (NRC). Since 2018 i have held an appointment as a Guest Research Fellow at the Central Research Institute of Electric Power Industry (CRIEPI) in Japan and I serve as a Principal Engineer at Phoenix Engineering Associates, Inc (PEAI).

<u>At Westinghouse</u> I participated in the initial development of industry plans for Master Curve (direct fracture toughness) implementation. Elements completed while at Westinghouse included development of ASME Code Cases, and their technical basis, which allowed use of Master Curve to estimate the index temperature ($RT_{To}$) for the $K_{IC}$ and $K_{IR}$ curves, and also development of the Kewaunee lead plant submittal.

<u>At the NRC</u> I continued to focus on RPV integrity issues, including the following:

- Led the government and industry team responsible for developing of technical basis for the alternate pressurized thermal shock rule (10 CFR 50.61a). Also worked as part of a team to develop guidance for application of 10 CFR 50.61a (Regulatory Guide 1.230).
- Led the team responsible for structural assessment and residual life prediction of the corroded head at the Davis-Besse nuclear power plant.
- Led and oversaw the contract that developed the probabilistic fracture mechanics (PFM) Code called FAVOR (Fracture Analysis of Vessels, Oak Ridge) and an on-line database of nuclear RPV surveillance data called REAP (Reactor Embrittlement Archive Project).
- Identified the need to re-assess the NRC's procedures to estimate $RT_{NDT}$ for early-construction plant steels (Branch Technical Position 5.3).
- Led the NRC's assessment of the continued adequacy of regulatory guidance on embrittlement prediction (Regulatory Guide 1.99).
- Addressed citizens' concerns of embrittlement and vessel failure risk at the Palisades nuclear plant via a webinar and a series of public meetings.
- Provided NRC support to several international partners in the aftermath of unexpected findings during made during inspections (Doel and Tihange in Belgium from 2012-2015, Beznau in Switzerland from 2015-2017), in response to significant public interest (Kori in South Korea), and as part of educational or development missions (taught a PFM course for IAEA in China, gave invited speech at a PFM symposium in Japan).

On non-RPV topics I worked on assessment of external hazards (postulated pipeline explosions near nuclear plants) and worked as part of a team developing regulatory guidance on the use of PFM in licensing actions (Regulatory Guide 1.254).

<u>At CRIEPI</u> I am working on projects focused on RPV integrity issues, including the following:

- Development of embrittlement trend curves and ETC modeling procedures, including machine learning techniques such as k-nearest neighbor (kNN). Application of the kNN method to develop advanced methods for surveillance during long term operation is now being evaluated.

- Developed a justification to eliminate the need for HAZ testing as part of RPV surveillance monitoring.
- Support of various CRIEPI projects including efforts to gain acceptance for using mini compact tension (mini-CT) specimens to determine $T_0$ and efforts to develop and gain acceptance of PFM techniques in Japan.
- Participating in the European Commission project ENTENTE, which concerns embrittlement modeling and database development.

At PEAI I am working on projects focused on RPV integrity issues, including the following:

- Development of an ASME Code case designated N-830 that allows the use of Master Curve and extended Master Curve models in ASME Code assessments.
- Development of an ASME Code case designated N-914 that provides a comprehensive methodology to account for neutron irradiation embrittlement in ASME Code assessments and includes parallel paths for both traditional (meaning Charpy and NDT-based) as well as Master Curve approaches.
- Removal of HAZ requirements for RPV beltline materials from the ASME Code.
- Assessed the impact of potential changes to US Regulatory Guide 1.99 on operating plants in the USA.
- Development of practical plant guidelines for addressing RPV integrity issues.
- Development of embrittlement prediction models applicable at the low reactor operating temperatures anticipated for future small modular reactor operations (SMRs).

## Professional Organizations

### American Society for Testing and Materials (ASTM)

I have been active in ASTM since the beginning of my career with the US Navy. My early activities focused on Committee E08 on Fatigue and Fracture where I contributed to the development of standards E1820 (J-R and $J_{Ic}$ testing) and E1921 (Master Curve $T_0$ testing). Upon joining the nuclear industry my focus shifted to Subcommittee E10.02 on the Behavior and Use of Nuclear Structural Materials. Within E10.02 I led a five-year effort that produced the first consensus embrittlement trend curve (E900-15) applicable to all western-grade light water reactor steels. I am currently responsible for coordinating the continued evaluation of the adequacy of the E900-15 predictive model as new data becomes available.

### American Society of Mechanical Engineers (ASME)

I am active in the ASME Working Group on Flaw Evaluation (WGFE) and in the Working Group on Operating Plant Criteria (WGOPC), which i have chaired since 2023. While with Westinghouse I supported efforts that produced the first Code Cases to use Master Curve (N-629, N-631). More recently I have worked as part of a team to develop a Code Case revision (N-830) and technical basis demonstrating the applicability of direct fracture toughness ("Master Curve") models for use in Section XI Appendices A, G, and K. In 2021 this revision to N-830 was adopted as part of the ASME Code. Since 2019 i have been developing a Code Case designated N-914 that provides a comprehensive methodology to account for neutron irradiation embrittlement in ASME Code assessments and includes parallel paths for both traditional (meaning Charpy and NDT-based) as well as Master Curve approaches. This Code Case is currently in the balloting process. On-going activities also focus on removal of HAZ analysis requirements from Section XI of the Code.

## Publications

I have authored or co-authored over 120 refereed journal articles, technical papers in conference proceedings, and technical reports, and have also served as editor for four ASTM Special Technical Publications. Key publications relevant to my work in nuclear structural integrity include the following:

- Lott, R.G., Kirk, M.T., and Kim, C.C., "Master Curve Strategies for RPV Assessment," Westinghouse Electric Corporation, WCAP-15075, November 1998.
- Application of Master Curve Fracture Toughness Methodology for Ferric Steels (PWRMRP-01): PWR Materials Reliability Project (PWRMRP); EPRI, Palo Alto, CA: 1999. TR-108390, Revision 1.
- Kirk, M.T., et al., "Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61)," NRC, NUREG-1806, August 2007, https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1806/v1/index.html
- Kirk, M.T., and Dickson, T.L., "Recommended Screening Limits for Pressurized Thermal Shock (PTS)," NRC, NUREG-1874, March 2010, https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1874/index.html
- Kirk, M., "Development of the Alternate Pressurized Thermal Shock Rule (10 CFR 50.61a) in the United States," *Nuclear Engineering and Technology*, Vol 45, No 3, June 2013.
- "Adjunct for ASTM E900-15: Technical Basis for the Equation used to Predict Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials," ASTM Subcommittee E10.02 Work Package WK34875, ASTM International, September 2015, https://www.astm.org/adje090015-ea.html.
- Kirk, M., Hardies, R., and Gordon, M., "Historical Review of the Development of ASTM Standard Practices E185 and E2215 from the 1960s to 2015," International Review of Nuclear Reactor Pressure Vessel Surveillance Programs, ASTM STP1603, W. L. Server and M. Brumovsky, Eds., ASTM International, West Conshohocken, PA, 2018, pp. 9–53, http://dx.doi.org/10.1520/STP160320170006
- Kirk, M., "Summary of Work to Develop the Transition Temperature Shift Equation Used in ASTM Standard Guide E900-15," International Review of Nuclear Reactor Pressure Vessel Surveillance Programs, ASTM STP1603, W. L. Server and M. Brumovsky, Eds., ASTM International, West Conshohocken, PA, 2018, pp. 432–456, http://dx.doi.org/10.1520/STP160320170009

- *Technical Basis for ASME Code Case N-830, Revision 1 (MRP-418): Direct Use of Master Curve Fracture Toughness Curve for Pressure-Retaining Materials of Class 1 Vessels, Section XI.* EPRI, Palo Alto, CA: 2017. 3002010332.
- *Methods to Address the Effects of Irradiation Embrittlement in Section XI of the ASME Code (MRP-462): Estimation of an Irradiated Reference Temperature using either Traditional Charpy Approaches or Master Curve Data.* EPRI, Palo Alto, CA: 2021. 3002020911.
- *Materials Reliability Program: Assessment of the Need to Consider Heat Affected Zone (HAZ) Properties in RPV Integrity Assessments (MRP-475).*EPRI, Palo Alto, CA: 2022. 3002023870.
- Yoshinori Hashimoto, Akiyoshi Nomoto, Mark Kirk, Kenji Nishida, "Development of new embrittlement trend curve based on Japanese surveillance and atom probe tomography data," *Journal of Nuclear Materials*, Volume 553, 2021, 153007, ISSN 0022-3115, https://doi.org/10.1016/j.jnucmat.2021.153007.
- M. Kirk, Y. Hashimoto, and A. Nomoto, "Application of a Machine Learning Approach Based on Nearest Neighbors to Extract Embrittlement Trends from RPV Surveillance Data," Journal of Nuclear Materials 568 (2022) 153886, https://www.sciencedirect.com/science/article/abs/pii/S0022311522003725?via%3Dihub.
- Diego Ferreño, Marta Serrano, Mark Kirk, and José A. Sainz-Aja. 2022. "Prediction of the Transition-Temperature Shift Using Machine Learning Algorithms and the Plotter Database" Metals 12, no. 2: 186. https://doi.org/10.3390/met12020186
- Ferreño, Diego, Mark Kirk, Marta Serrano, and José A. Sainz-Aja. 2022. "Assessment of the Generalization Ability of the ASTM E900-15 Embrittlement Trend Curve by Means of Monte Carlo Cross-Validation" Metals 12, no. 3: 481. https://doi.org/10.3390/met12030481
- M. Kirk, D. Ferreño Blanco, and J. A. Sainz-Aja Guerra, "Evaluation of the ASTM E900 $\Delta T_{41J}$ Prediction Equation in Light of New Data," in Radiation Embrittlement Trend Curves and Equations and Their Use for RPV Integrity Evaluations, ed. W. L. Server, M. Brumovsky, and M. Kirk (West Conshohocken, PA: ASTM International, 2023), 233–258, http://doi.org/10.1520/STP164720220063
- M. Erickson and M. Kirk, "Effects of Prior Hardening on Irradiation Embrittlement Predictions," in Radiation Embrittlement Trend Curves and Equations and Their Use for RPV Integrity Evaluations, ed. W. L. Server, M. Brumovsky, and M. Kirk (West Conshohocken, PA: ASTM International, 2023), 65–81, http://doi.org/10.1520/STP164720220075

A more complete citation list can be found on GOOGLE Scholar at https://scholar.google.com/citations?user=tVlFcrsAAAAJ&hl=en

# Appendix

## Embrittlement Trend Curves

### A Trend Curve for $\Delta T_{41J}$

The embrittlement trend curve from [ASTM E900-15] for $\Delta T_{41J}$is as follows:

$$\Delta T_{41J:E900(15)} = \frac{5}{9}[max\{min(\mathrm{Cu},\ 0.28) - \ 0.053, 0\}M + B]$$

$$M = \begin{bmatrix} \mathrm{W{:}\ 0.968} \\ \mathrm{P{:}\ 0.819} \\ \mathrm{F{:}\ 0.738} \end{bmatrix} max\{min[113.87\,(ln(\Phi) - ln(4.5 \times 10^{16})), 612.6], 0\}\left(\frac{1.8\,\mathrm{T} + 32}{550}\right)^{-5.45}\left(0.1 + \frac{P}{0.012}\right)^{-0.098}\left(0.168 + \frac{Ni^{0.58}}{0.63}\right)^{0.73}$$

$$B = \begin{bmatrix} \mathrm{W{:}\ 0.919} \\ \mathrm{P{:}1.080} \\ \mathrm{F{:}1.011} \end{bmatrix} 3.593 \times 10^{-10}\Phi^{0.5695}\left(\frac{1.8\,\mathrm{T} + 32}{550}\right)^{-5.47}\left(0.09 + \frac{P}{0.012}\right)^{0.216}\left(1.66 + \frac{Ni^{8.54}}{0.63}\right)^{0.39}\left(\frac{Mn}{1.36}\right)^{0.3}$$

$$SD = \begin{bmatrix} \mathrm{W{:}\ 7.681} \\ \mathrm{P{:}\ 6.593} \\ \mathrm{F{:}\ 6.972} \end{bmatrix} \times (\Delta T_{41J})^{[\mathrm{W{:}0.181}\quad \mathrm{P{:}0.163}\quad \mathrm{F{:}0.199}]}$$

In this equation W, P, and F mean weld, plate, and forging (respectively); standard reference materials (SRMs) are classified as plates. SD stands for “standard deviation.” Composition variables have units of weight percent, temperatures are expressed in °C, and fluence (E > 1MeV) is expressed in n/cm$^2$.

Full details on the development and basis for this equation appear in [ASTM Adjunct]. Use of this equation should follow the requirements and limitations of [ASTM E900-15]

### A Trend Curve for $\Delta T_{41J}$

The embrittlement trend curve designated as UNM-6 from [Kirk 2010] for ΔUSE is as follows. This equation appears differently than in the original publication; here it has been converted into SI units.

$$\%\ \boldsymbol{drop} = \frac{\Delta USE}{USE_{(U)}} = 0.00297 \cdot \Delta \mathrm{T_{41J}} \cdot \{\mathrm{M}_{\boldsymbol{\Phi}} \cdot \mathrm{M_T} \cdot \mathrm{M_P} \cdot \mathrm{M_{Cu}} \cdot \mathrm{M_{Ni}} \cdot \mathrm{M_{Mn}}\}$$

$$M_\Phi = \left[\frac{Log_{10}(\Phi)}{18.83}\right]^{-4.9} \qquad M_\mathrm{T} = \left[\frac{1.8 \times T + 32}{545}\right]^5 \qquad M_\mathrm{P} = \left[\frac{P}{0.012}\right]^{-0.32}$$

$$M_\mathrm{Cu} = \left[\frac{Cu}{0.138}\right]^{0.19} \qquad M_\mathrm{Ni} = \left[\frac{Ni}{0.57}\right]^{-0.03} \qquad M_\mathrm{Mn} = \left[\frac{Mn}{1.31}\right]^{-0.1}$$

The uncertainty in this relationship (standard error) is $\sigma = 0.1$. In this equation $\Delta T_{41J}$ is expressed in °C and is as predicted by the [ASTM E900-15] ETC, irradiation temperature is expressed in °C, neutron fluence is expressed in n/cm$^2$ (E > 1 MeV), and all composition variables are expressed in weight percent.

# ATTACHMENT C

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

By

Dr. Digby Macdonald[1]

June 14, 2024

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

  - All calculated and physically measured data must come with "experimental" uncertainties and preferable standard deviations, and these uncertainties must be displayed by error bars along with the data in any graphical form. The errors and standard deviations in the USE data for unirradiated steels have been determined to be 21-38 ft. lb and 18.1 to 23.8 ft. lb for observed and model-calculated data, respectively (Table 4.1 for $ART_{NDT}$, from (Eason, E D; Wright, J E; and Odette, MCS G R (1998)). For $ART_{41J}$, the standard deviation is 21-34 ºF and 25.9 ºF-38 ºF for measured and calculated data, respectively. The "uncertainty" in the fluence is estimated at 10-20 % (CAP V Report, Appendix D (2003)), which, presumably, incorporates the uncertainty in whether flux changes due to all power changes are accurately accounted for.

    Neither the reference temperature nor USE nor the fluence are adjustable parameters, so they should not be treated as such to achieve a favorable result.

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

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

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

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

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

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

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

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

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

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

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

and

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

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

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

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

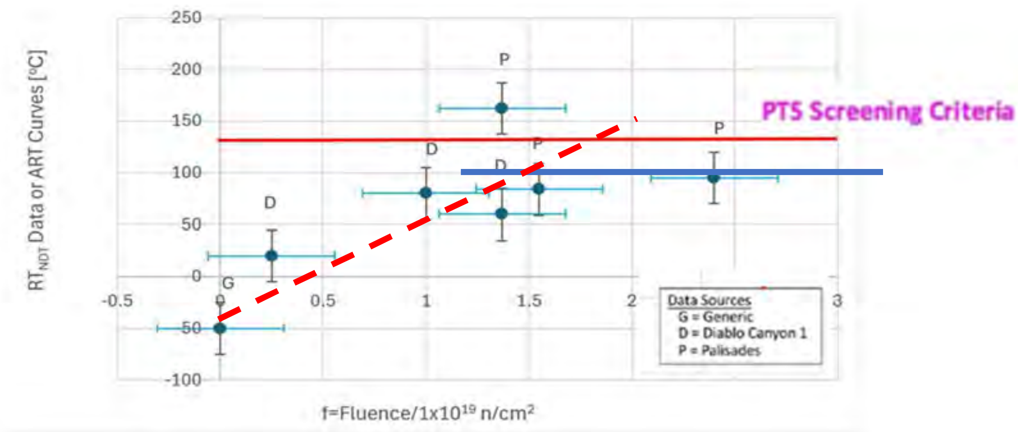


**Figure 1.** Summary of DCPP-1 $RT_{NDT}$ vs. fluence data along with similar data integrated without NRC authorization by Dr. Kirk from the PNP, in addition to a data point for PNP from and (RG 1.99, Revision 2 (1988)), with each point being assigned error bars as per (Eason, E D; Wright, J E; and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

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

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

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

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

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

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

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

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

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

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

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

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

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

or

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

where

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

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

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

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

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

and

$$h(Cu) = \left\{\frac{\frac{0,Cu\leq 0.72\ Wt.\%}{(Cu-0.072)^{0.678},0.072<Cu<0.300\ Wt.\%}}{0.367,Cu\geq 0.300\ Wt,\%}\right\} \quad (6)$$

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

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

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

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

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

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

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

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

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

    b. Each reactor must have an adequate dosimetry program.

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

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

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

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

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

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

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

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

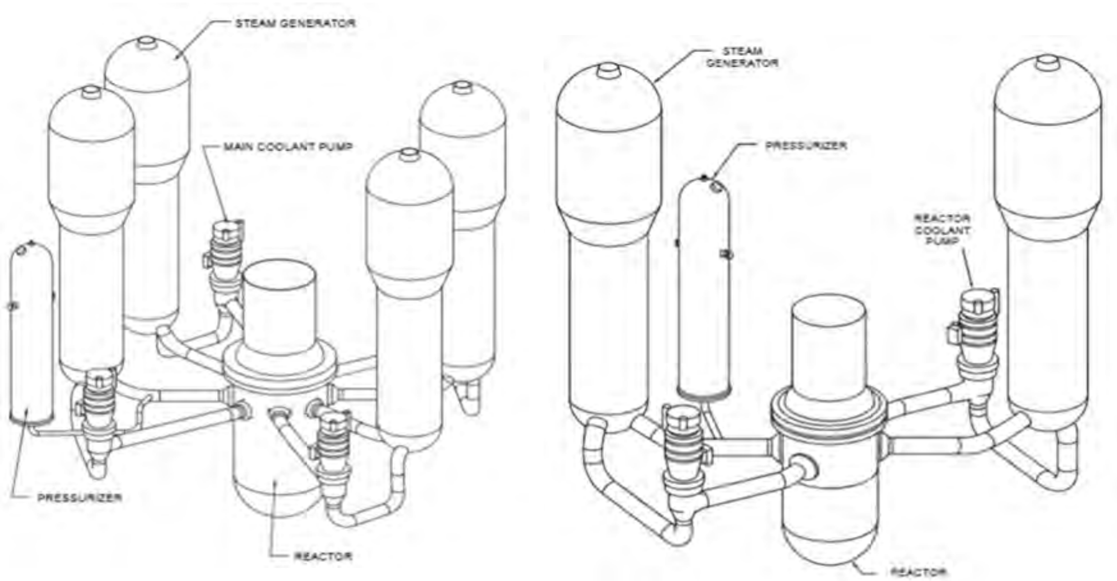


**Figure II.1.** Schematics of a Westinghouse-designed four-loop Pressurized Water Reactor (PWR, like DCPP-1(left), and a two-loop model like PNP (right)).

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

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

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

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

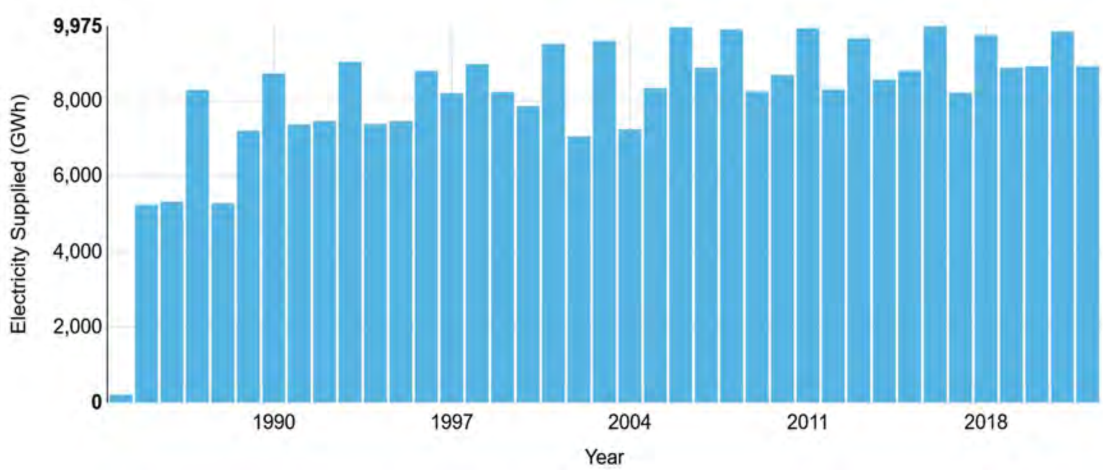


**Figure II.2.** Electricity supplied by DCPP-1 from start-up in 1984 to 2022.

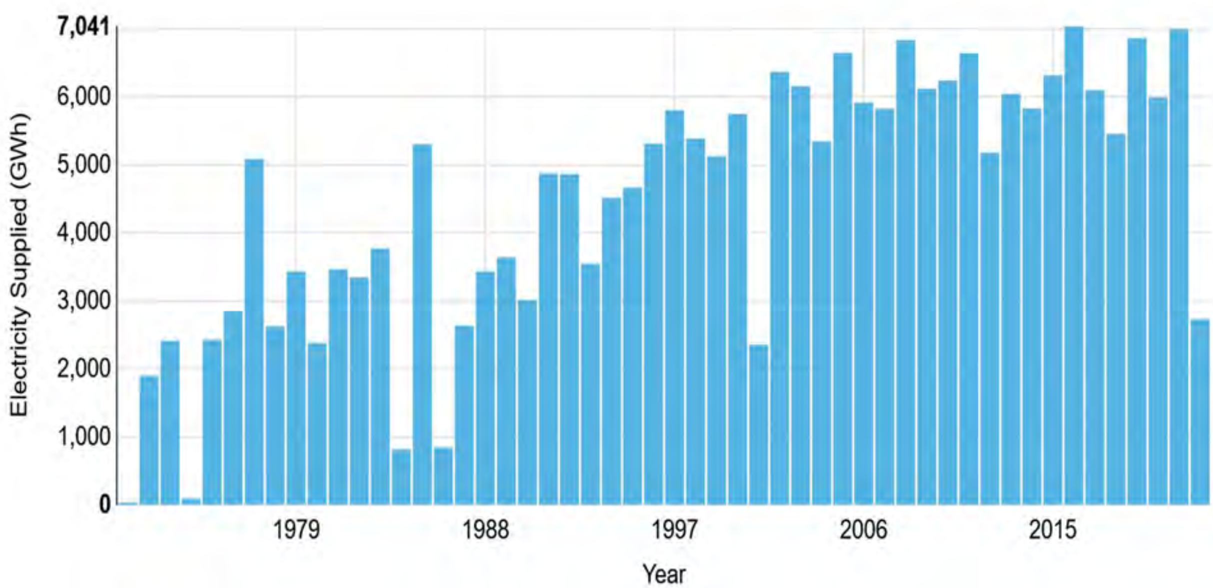


**Figure II.3.** Electricity supplied by PNP from a start-up in 1972 to shut down in 2020.

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

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

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

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

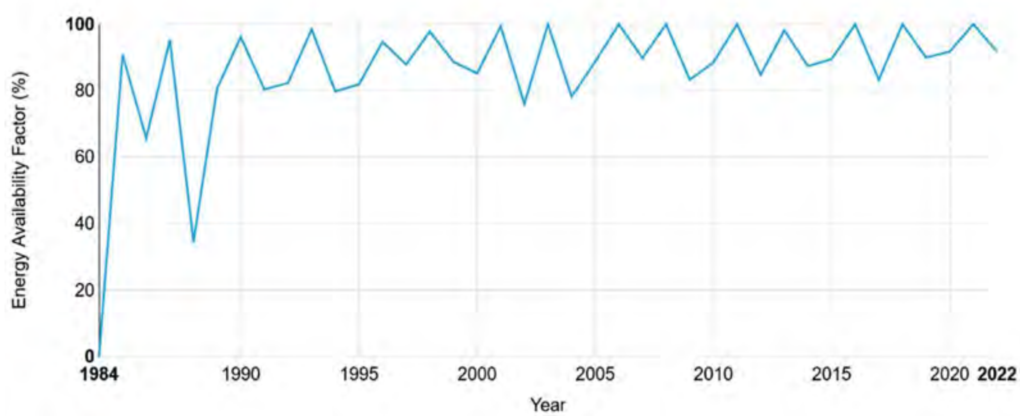


**Figure II.4.** Energy availability factor for DCPP-1 from startup to 2022.

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

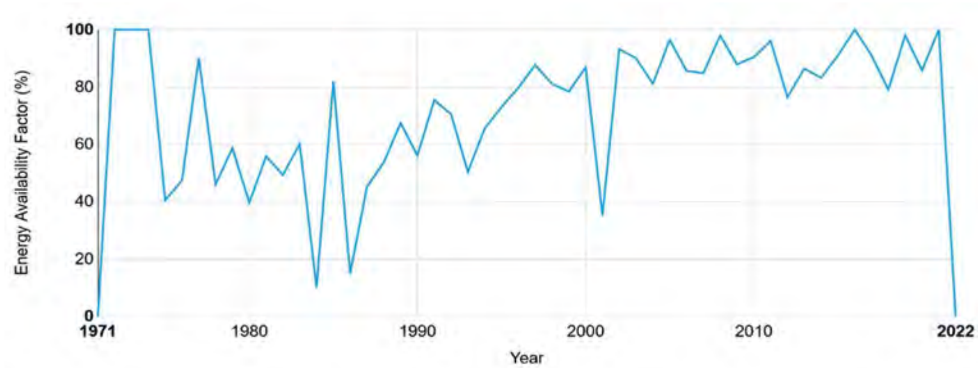


**Figure II.5.** Energy availability factor for PNP from startup to 2022.

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

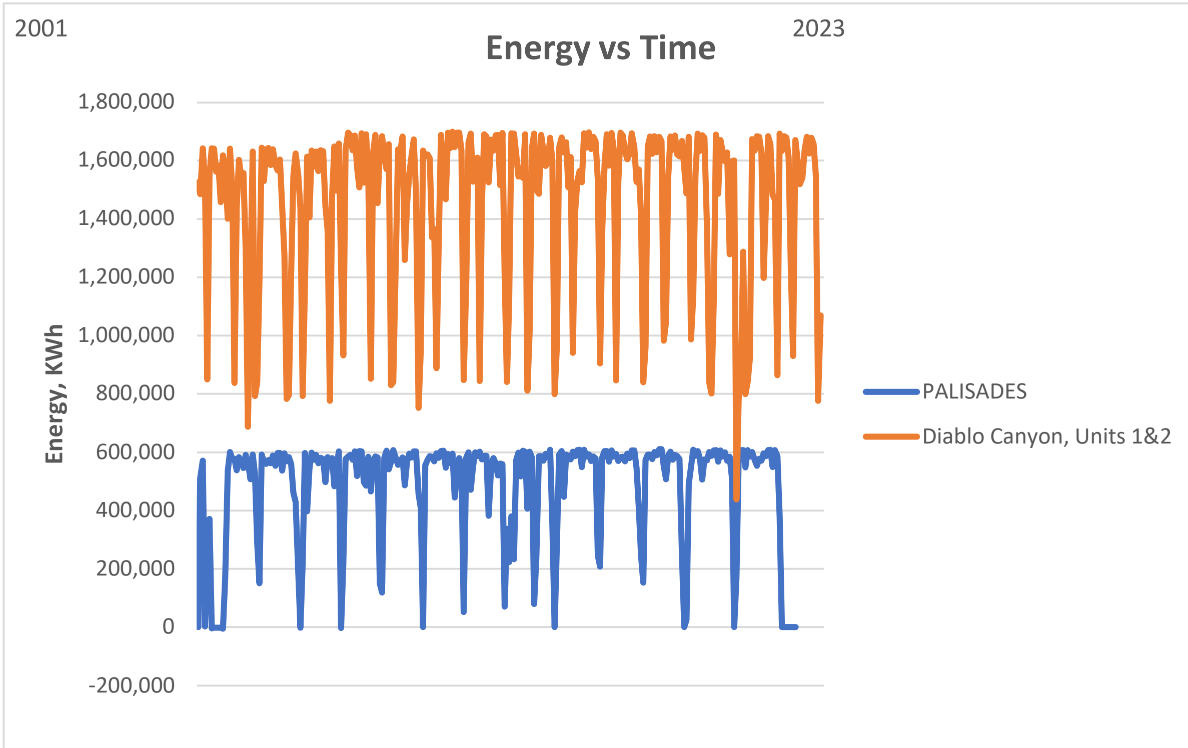


**Figure II.6.** Comparison of the monthly energy produced by DCPP-1 and PNP over the past ca. twenty-five years. Data from the US Energy Information Administration (EIA).

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

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

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

## Appendix III. Errors in Experimental and Calculated Data.

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

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

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

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

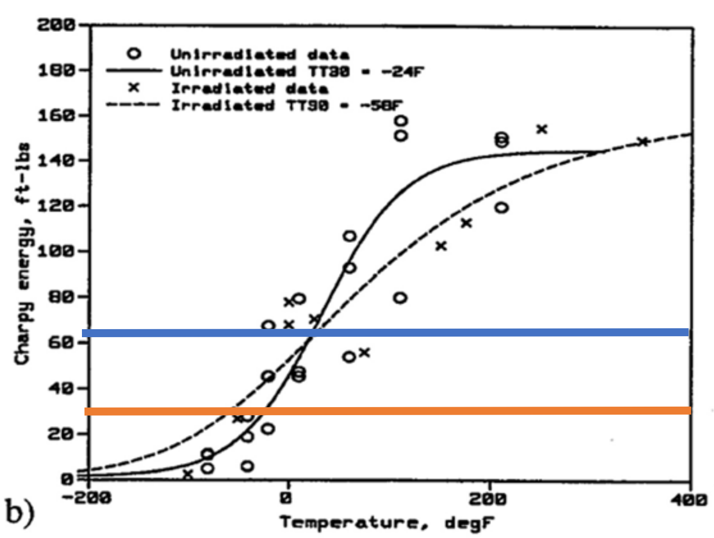


**Figure III.1.** Illustration of the effect of neutron irradiation on the mechanical properties of a pressure vessel steel as revealed by Charpy Impact Testing (CIT). The orange line represents $RT_{41J}$ and the blue line $RT_{POI}$. (Eason, E D; Wright, J E; and Odette, MCS G R (1998)).

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

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

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

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

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

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

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

## Appendix IV. Redefinition of $RT_{NDT}$.

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

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

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

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

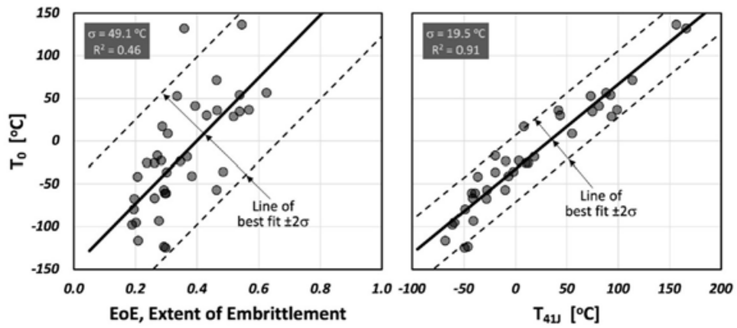


**Figure IV.1.** Graphs presented by Dr. Kirk support his assertion that my approach to calculating the embrittlement parameters is inferior to his (from (Kirk Report (2024)), Part 2, p. 57).

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

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

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

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

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

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

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

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

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

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

and

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

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

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

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

Examination of the data in Figures IV.2 to IV.4 shows that the embrittlement of the materials may be ranked as follows: Weld >> HAZ > Plate ≈ SRM. Figures IV.2 and IV.4 express these rankings more explicitly, especially in Figure IV.2. Finally, it is to be noted that the unirradiated materials are assigned fluence values of $1x10^{17}$ n/cm$^2$ because of the mathematical impossibility of assigning a value of zero fluence in a linear-log graphical format. The assigned value is too low to display any sign of embrittlement. Regardless of the arbitrary assignment of fluence for an unirradiated material, the results remain valid, as seen from the parameters for the three higher fluence values.

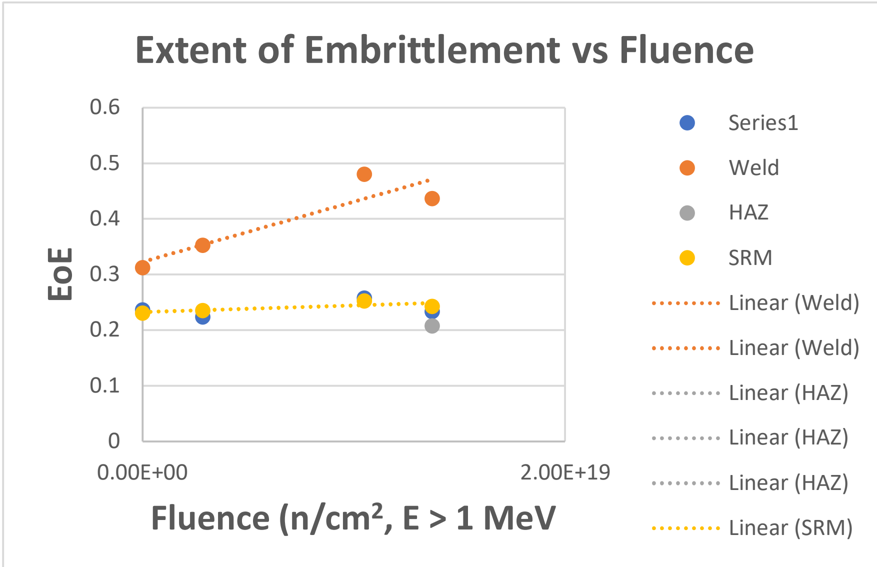


**Figure IV.2.** Plots *EoE* vs. Fluence for plate, weld, HAZ, and SRM specimens from Capsules S, Y, and V in Diablo Canyon, Unit 1 Nuclear Power Plant. Note that in some cases, "healing (reduction in the EoE)" occurs at high fluence (CAP V Report, Appendix D (2003)).

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

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

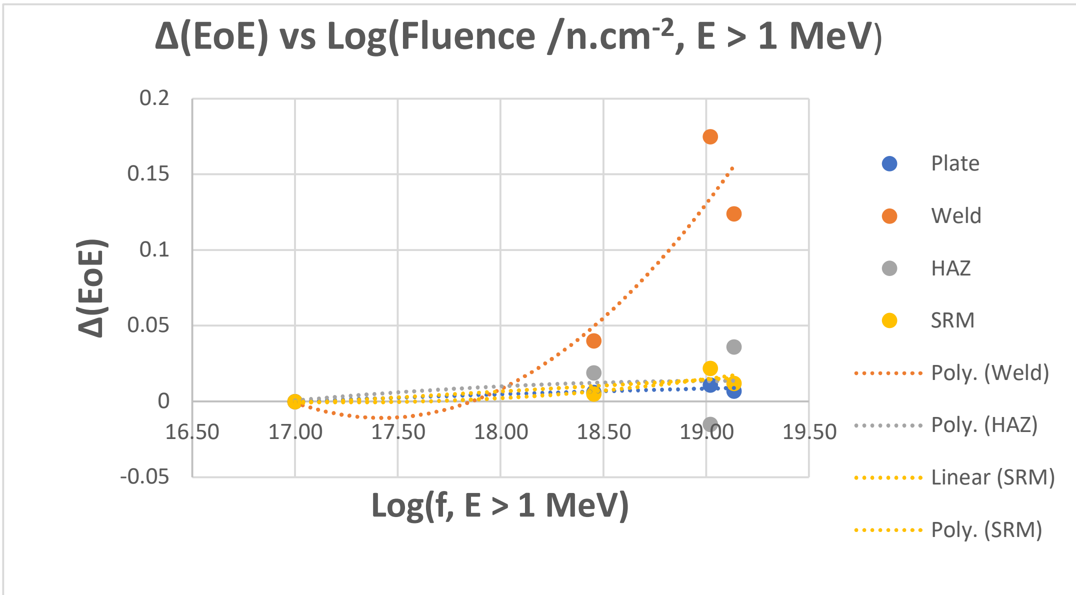


**Figure IV.3.** Plots ΔEoE vs. log(Fluence) for weld, plate, HAZ, and SRM specimens from Capsules S, Y, and V in Diablo Canyon, Unit 1 Nuclear Power Plant (Macdonald, D D (2023)). Note that the solid lines are included to emphasize that in some cases, "healing (reduction in the EoE)" apparently occurs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

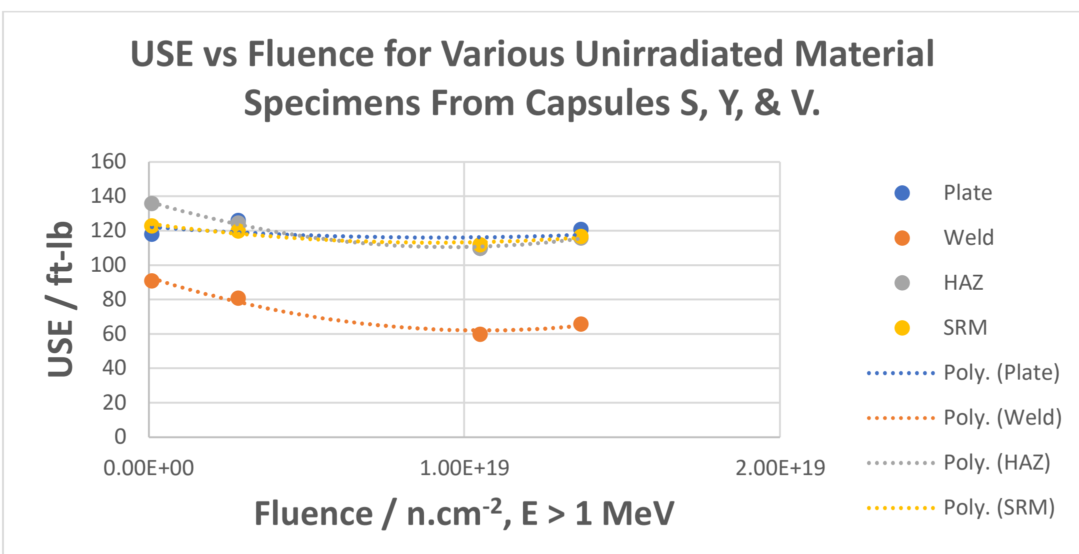


**Figure IV.4:** Plots USE vs Fluence for plate, weld, HAZ, and SRM specimens from Capsules S, Y, and V in Diablo Canyon, Unit 1 Nuclear Power Plant (CAP V Report, Appendix D (2003))

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

## Appendix VI. Why UT Examination is Essential.

### Introduction

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

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

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

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

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

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

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

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

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

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

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

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

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

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

### Reactors Doel-3 and Tihange-3 in Belgium

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

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

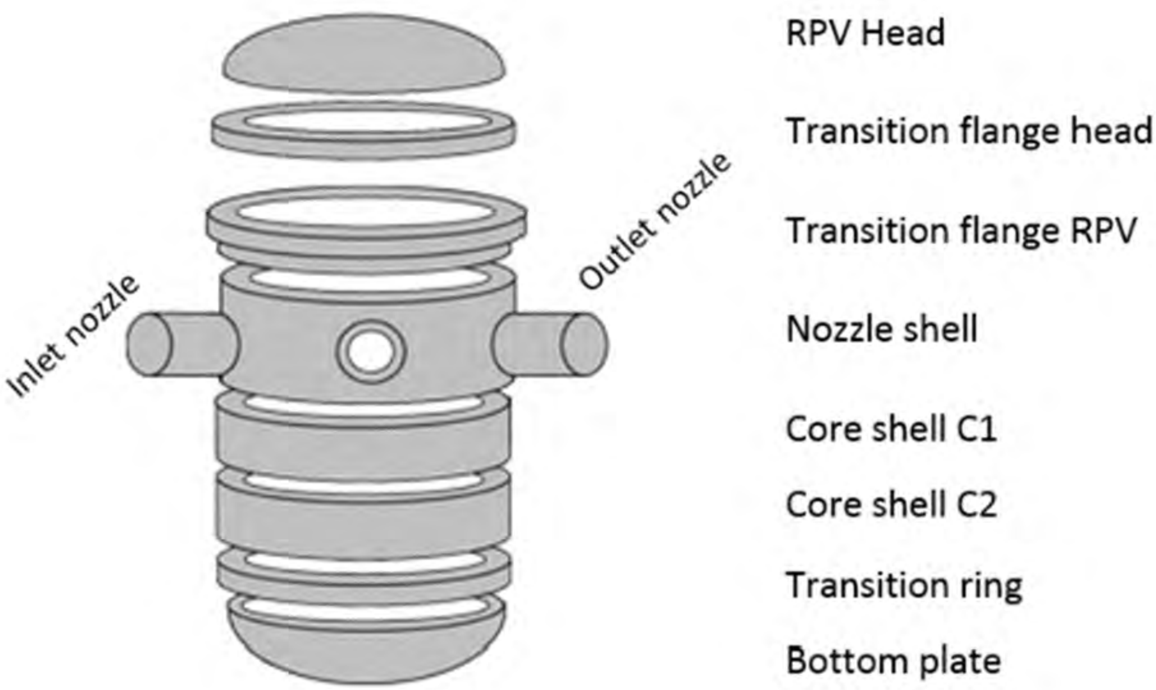


**Figure VI.1:** Illustration translated from ((FANC (2013)) and (FANC (2015))) showing the original forged steel ring sections of the RPV separated for clarity. These rings are welded and clad internally with a stainless-steel liner to form the reactor pressure vessel.

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

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

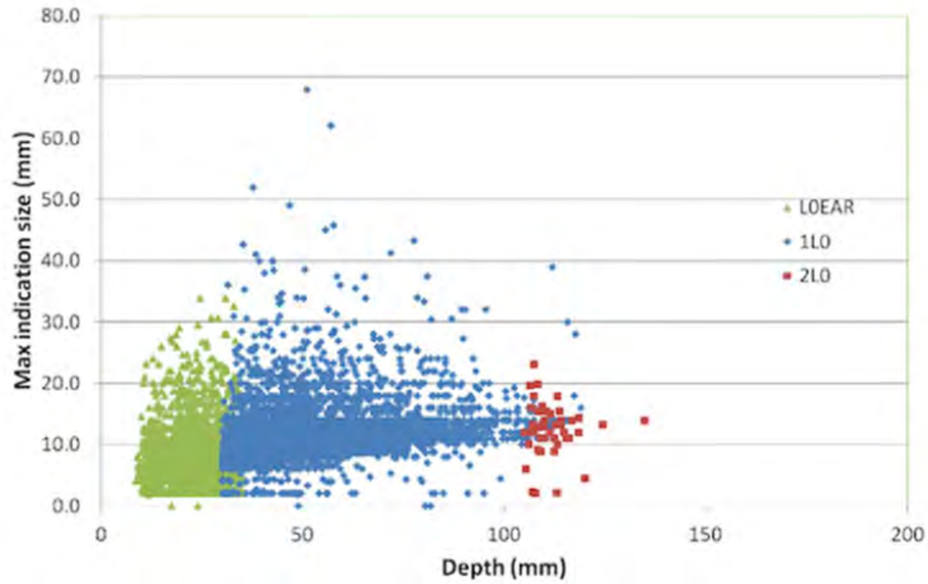


**Figure VI.2:** Doel 3 – Size of indications [max (x, y)] versus depth into the RPV steel wall (data 2012). **(**Electrabel (2013); The Greens (2013)**)**

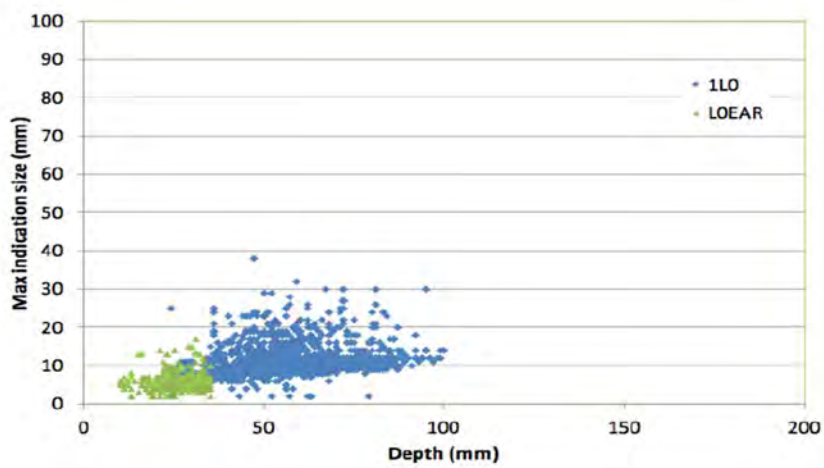


**Figure VI.3:** Tihange 2 – Size of indications [max (x, y)] versus depth into the RPV steel wall (Electrabel (2012).).

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

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

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

### Causes of the Cracks

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

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

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

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

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

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

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

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

**Water chemistry, corrosion effects, and hydrogen sources.**

**Hydrogen from cathodic partial corrosion reaction**

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

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

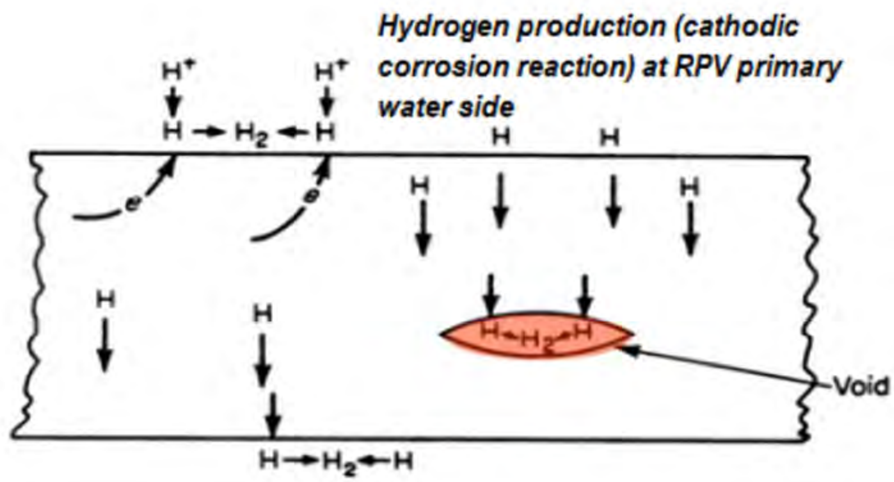


**Figure VI.4.** Schematic diagram of hydrogen diffusion and blister formation.

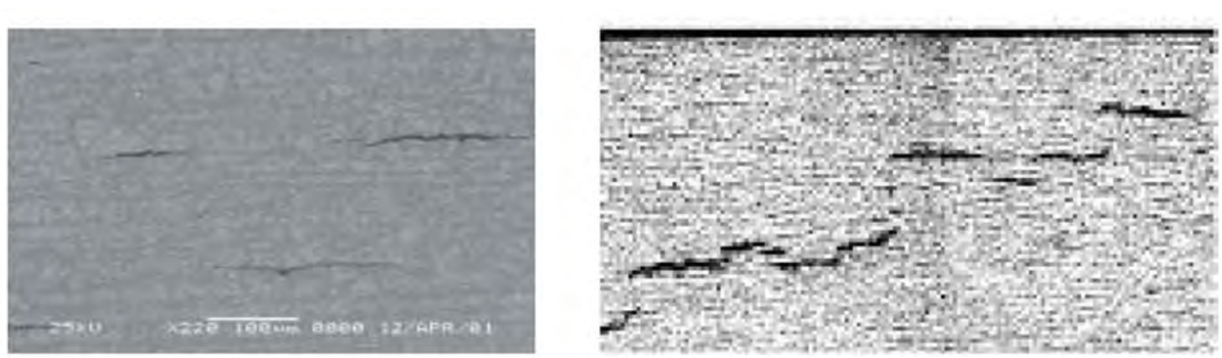

**Figure VI.5.** Typical hydrogen-induced cracks (An Atlas of Corrosion and Related Failures (1987)).

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

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

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

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

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

## **Aging risks and crack growth**

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

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

## Hydrogen Embrittlement in PWRs

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

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

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

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

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

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

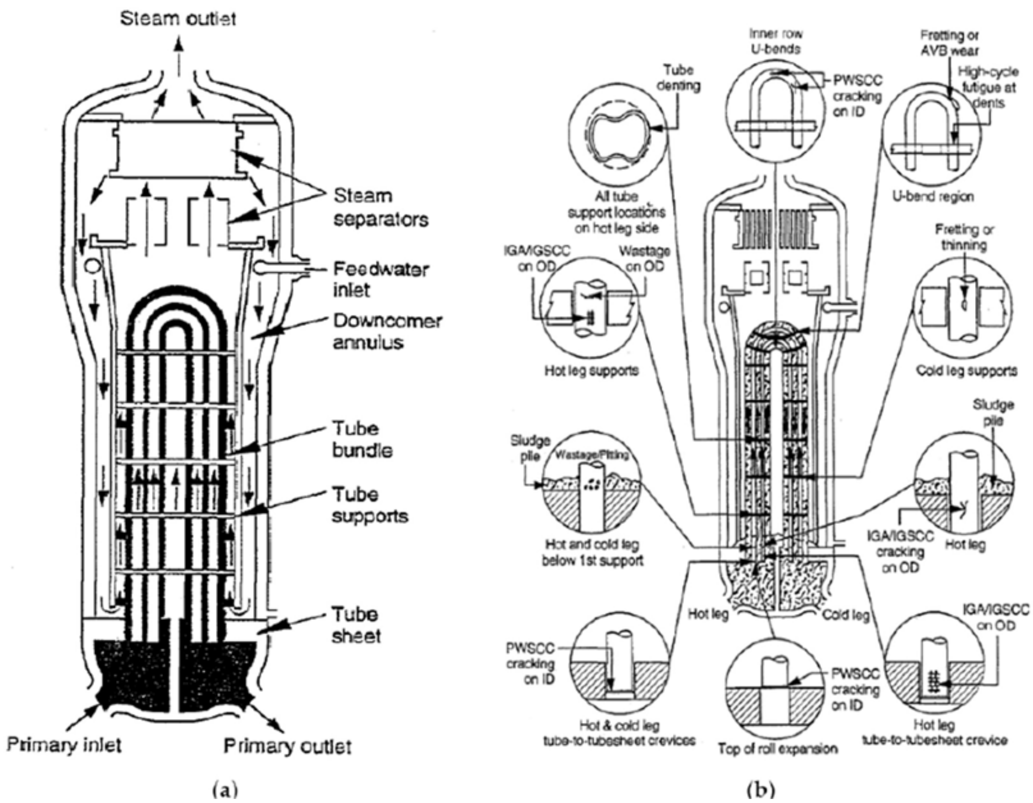


**Figure VI.6.** Corrosion problems in PWR steam generators. (a) General layout of a recirculating steam generator. (b) All PWSCC (Primary Water Stress Corrosion Cracking, including OD IGA/IGSCC) form a brittle fracture.

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

| **Property** | **Value** | **Comment** |
|---|---|---|
| Temperature | 295°C–330°C | Typical |
| Pressure | 150 bar (2250 psi) | Typical |
| Coolant composition | 4000–0 ppm B as boric acid, 4–1 ppm Li as lithium hydroxide, depending upon the burn-up of the fuel and the vendor | Li-B trajectory over a typical fuel cycle is shown in Figure 27(a). |
| Hydrogen concentration | 25–55 cc(STP [standard temperature and pressure])/kg($H_2O$) | Some noncommercial units operate with [$H_2$] as high as 70 cc(STP)/kg($H_2O$) |
| Core channel dose rate | | Typical |
| γ-Photon | $3\times10^5$ Rad/s | |
| Neutron | $6\times10^5$ Rad/s | |
| α Particles | $3\times10^5$ Rad/s | |
| Coolant Mass Flow Rate | 18,000 kg/s | Typical |

Compared with the work on modeling BWR primary coolant circuits, much less work has been reported on assessing electrochemical effects in PWR primary circuits. This situation reflects that cracking has not been as significant a problem in PWR primary coolant circuits as in BWR primary coolant circuits. However, the primary water stress corrosion cracking (PWSCC) of mill-annealed Alloy 600 steam generator tubes, pressurizer components, control rod drive tubes, and baffle bolts (highly cold-worked Type 316 SS) have been severe, recurring issues in PWR operation, for example. Because of the high hydrogen concentration [typically 25-50 cc(STP)/kg($H_2O$), corresponding to $1.12\times10^{-3}$ m to $2.24\times10^{-3}$ m or 2.23 to 4.46 ppm] employed in a PWR primary circuit to "suppress radiolysis," and given the lack of sustained boiling, it was generally believed that the hydrogen equilibrium potential dominates the ECP and hence that the coolant circuit acts as a "giant hydrogen electrode." If so, an approximate value of the ECP is readily calculated from the known pH, which, in turn, is easily estimated from the boron and lithium contents of the primary coolant and the known hydrogen concentration using the Nernst equation. Considering subsequent modeling, this picture is not entirely accurate; more importantly, PWRs are not free from cracking in their primary circuits, and the cracking observed is brittle and very potential-dependent. For example, Primary Water Stress Corrosion Cracking (PWSCC) of Alloy 600 steam generator tubes has plagued operators for many years, as noted above, and cracking of core barrel bolts (highly cold-worked Type 316 SS) has also been a recurring problem. A typical brittle fracture in highly cold-worked Alloy 600 in PWR primary coolant is shown in Figure VI.7, proving that brittle fracture occurs within many materials in PWR primary circuits. This issue is discussed further below.

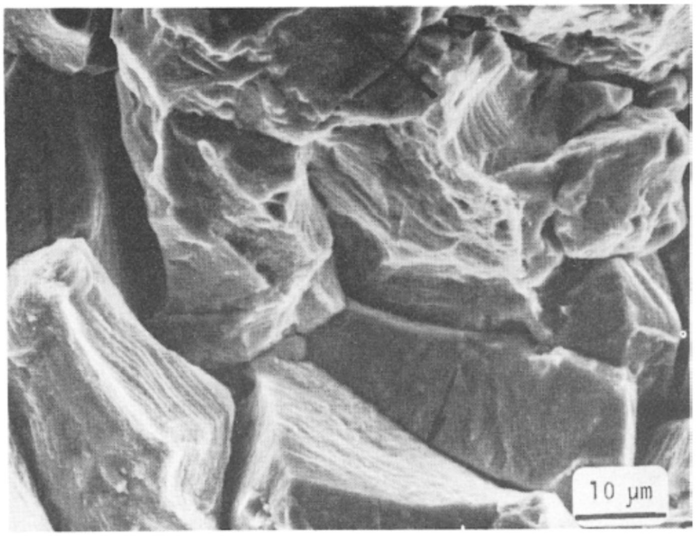


**Figure VI.7.** IGSCC in Alloy 600 after exposure to simulated PWR coolant at 350 °C and 0.1 MPa of $H_2$. Note that the solution was saturated with $H_2$ at ambient temperature and then pumped into the autoclave at 350 °C at sufficient pressure to suppress boiling. The specimen failed in a CERT.

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

To understand the chemistry of a PWR primary circuit coolant, we have employed a proprietary radiolysis model to calculate the evolution of various chemical species with time upon "switching on" the fission processes typical of a PWR operating at full power. The concentrations of all species are then plotted as a function of time in Figure VI.8. Of particular interest are the concentrations of $H_2$, H, and OH, all three potent reducing agents, one of which may enter a steel lattice through the surface without the need to be formed by the dissociation of hydrogen, $H_2 \rightleftarrows 2H$. From this equilibrium, we see that the hydrogen atom concentration is predicted to be about $2x10^{-8}$ M. From thermodynamic guesstimates, the concentration of hydrogen atoms in equilibrium with molecular hydrogen at a pressure of 1 atm at 300 °C is calculated to be about $1x10^{-20}$ m. The equilibrium constant for this reaction is written as:

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

where $K_p$ is estimated to be about $10^{-40}$ $M^2$/atm. We may turn this equation around and calculate the pressure of hydrogen needed to obtain an H concentration of $2x10^{-8}$ M. The result is an astounding $4x10^{24}$ atm. No such pressure has ever been generated on Earth.

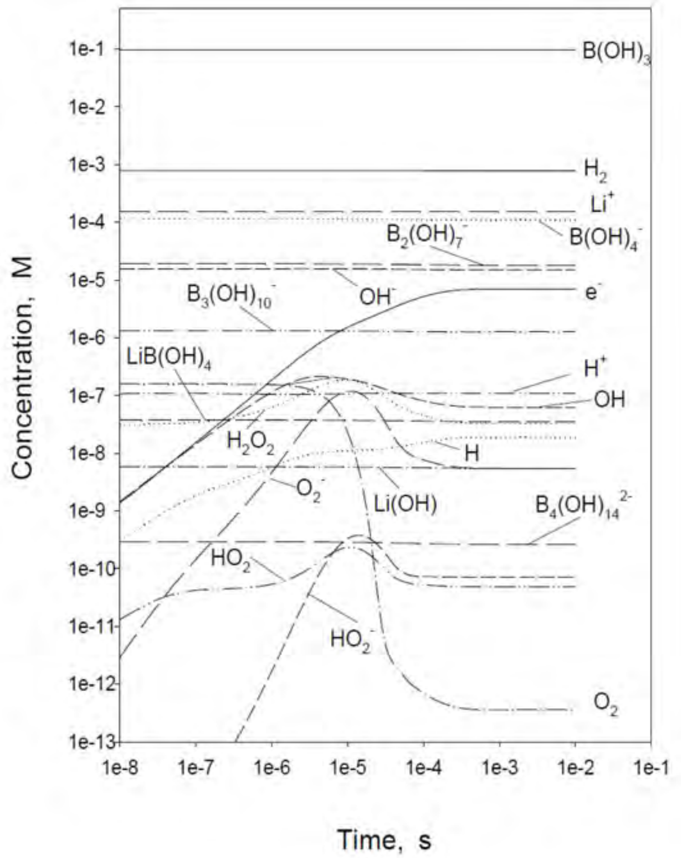


**Figure VI.8.** Predicted concentrations of radiolysis and pH control species in the primary coolant of a PWR as a function of time towards achieving a local steady state for $C_{B,T}$ = 1500 ppm, $C_{Li,T}$ = 1.5 ppm, T = 300 °C, $[H_2]$ = 25 $cm^3$/kg STP, $\Gamma_n = 1\times10^{20}$ $eV/cm^3s$, and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

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

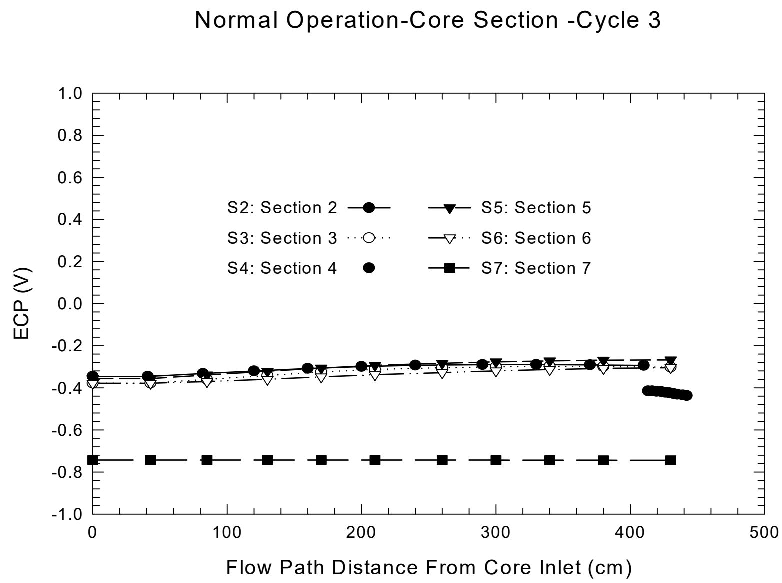


**Figure VI.9.** Predicted ECP vs distance in a PWR under normal, full power operation. S1 = Core return, S2 = Fuel channels, S3 = Hot zone, S4 = Top part of the core, S5 = By-pass Tube Guide, S6 = By-pass hot zone, S7 = By-pass grid.

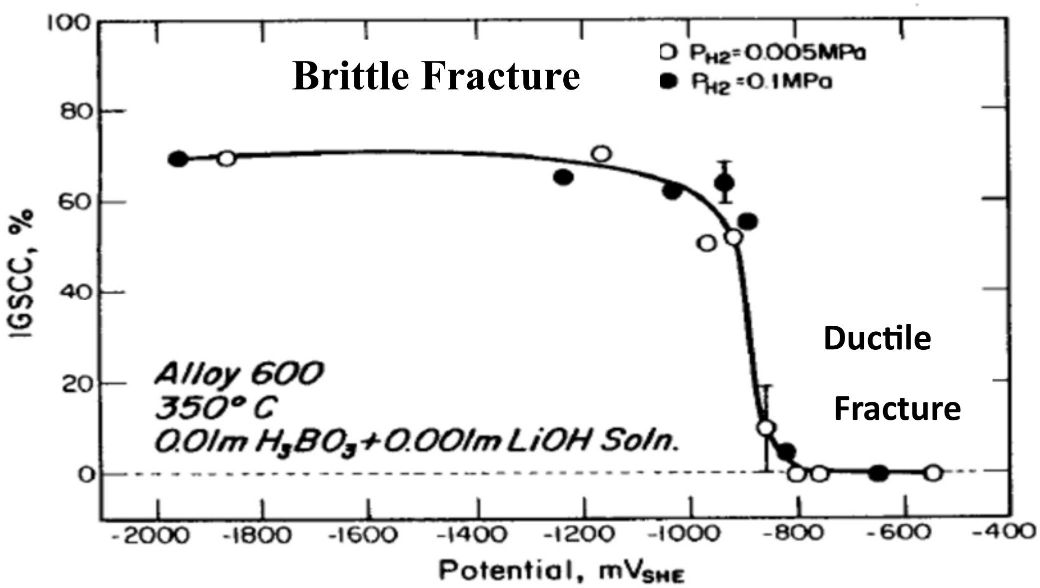


**Figure VI.10.** Illustration of electrochemically induced hydrogen embrittlement in Alloy 600 in hydrogenated PWR primary coolant. (Totsuka, N and Szklarska-Smialowska, Z (1987)).

Totsuka and Szklarska-Smialowska performed CERT experiments on Alloy 600 samples in PWR environments as a function of (applied) potential, and their data are plotted in Figure VI.10 (Totsuka, N and Szklarska-Smialowska, Z (1987)). They found a transition from ductile to brittle fracture as the potential is more negative. This is a direct confirmation of the theory that Macdonald and coworkers had put forward that the brittle failures experienced in a variety of materials in the primary coolant of PWRs could be traced to the corrosion potential being too negative, either because the pH was too high or, more importantly, because the hydrogen concentration was too high. Indeed, Bertuch, Pang, and Macdonald found (Bertuch, A; Pang, J; and Macdonald, D D (1995)) that only five cc(STP) $H_2$/kg$H_2O$ is needed to "suppress radiolysis," in contrast to the industry-recommended levels of hydrogen of 5 to 7 times as much. Had the industry accepted our lower figure, also proposed later by some other research groups, the enormous cost suffered by PWR operators might have been avoided.

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

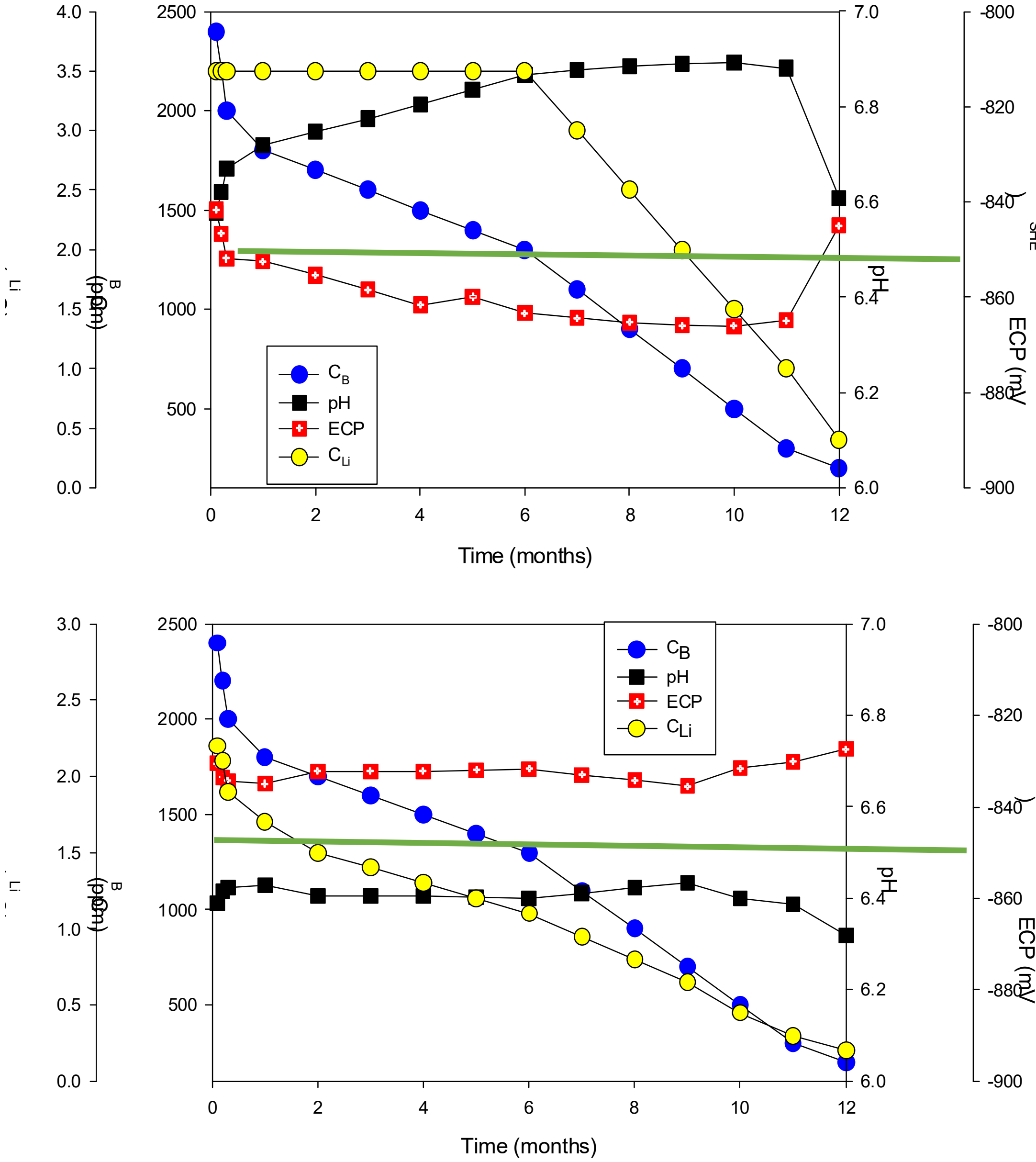


**Figure VI.11.** (a) Predicted ECP and pH over a fuel cycle for a PWR operating on coordinated water chemistry and (b) adjusted water chemistry. $[H_2]$ = 25 $cm^3$ (STP)/kg$H_2O$, T = 320 ºC (Kim, H S (2007)). The dark green lines are the critical ECP for brittle fracture.

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

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

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

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

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# ATTACHMENT D

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July 19, 2024

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

Dear Mr. Rathie:

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

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

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

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

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

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

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

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

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

Additionally, I note the following:

1. The uncertainty in fluence is estimated as ±14 to 20 % [WCAP 15958 (2003)].
2. I should reiterate my critique of Dr. Kirk for omitting error bars, as this practice appears prevalent in the field.
3. The three highest fluence points are from Palisades; the three lowest are from DCPP-1.
4. Intersection with PTS screening criterion occurs between $1.2x10^{19}$ and $3.0x10^{19}$ $n/cm^2$.
5. The plot predicts that the beltline welds will become unacceptably embrittled sometime within the first 20-year extension or early in the next.

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

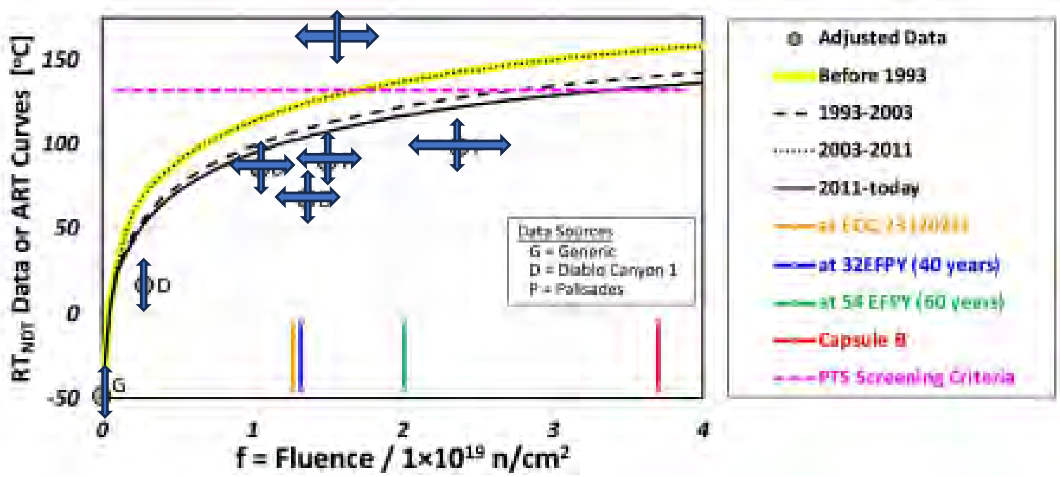


**Figure 1.** Corrected Figure 1 from June 14 Technical Report.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

**3. Concerning hydrogen induced cracking:**

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

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

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

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

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

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

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

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

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

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

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

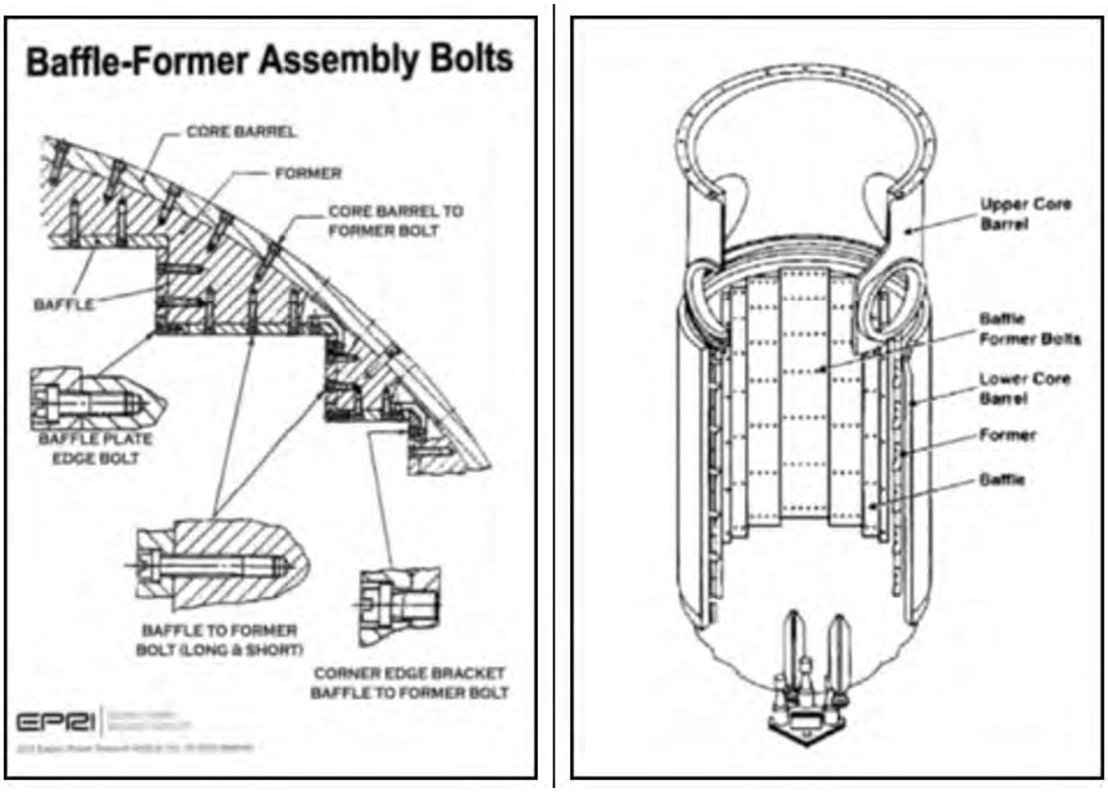


**Figure 2**. Configuration of baffle former bolts in a Westinghouse-designed PWR that frequently fail by HIC. [NRC (2012)]. Details of the bolts are given immediately following this caption, as follows:

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

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

The analysis of the equilibrium pressure from the CERT testing assumes that the material in the fracture region is in equilibrium with the surface hydrogen concentration produced from the corrosion reaction. There are two paths for hydrogen transport through the specimen. The first is by bulk diffusion, and the second is by dislocation transport. The test time for the X-750 specimen tested at 360°C was 40 h. The diffusivity of hydrogen at 360 °C is $1.4 \times 10^{-10}$ $m^2/s$. The test time required to achieve a near-uniform hydrogen distribution through the thickness for these conditions is about 15 h. Therefore, while dislocation transport may reduce this time, accounting for the high measured bulk hydrogen concentration, it is unnecessary.

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

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

Rubel, et.al (1989) has reviewed the German (KWU) experience with Alloy (Inconel) X-750 in PWR internals. The excellent mechanical properties (high strength, high creep resistance) are the result of the precipitation of a semi-coherent γ'-phase (type $Ni_3AI$) during age-hardening and its distribution, as well as of the distribution and morphology of $M_{23}C_6$ carbide formation [Rubel et.al (1989)]. These precipitates act as barriers to the movement of dislocations, which results in the loss of ductility (i.e., hardening). Most significant is the damage in Japanese, French, and American PWRs to the centering pins of the control-rod guide assemblies [Rubel et.al (1989)]. The cause of damage was IGSCC because of high local stresses (in particular, notch effects arising from crosssection transitions), to-helical control rod drive springs, fuel guide pins, core barrel bolts, and other fasteners, all with unfavorable heat treatment.

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

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

## 4. Concerning surveillance

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

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

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

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

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

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

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

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

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

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

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

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

**5. General questions**

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

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

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

See Figure 3 below.

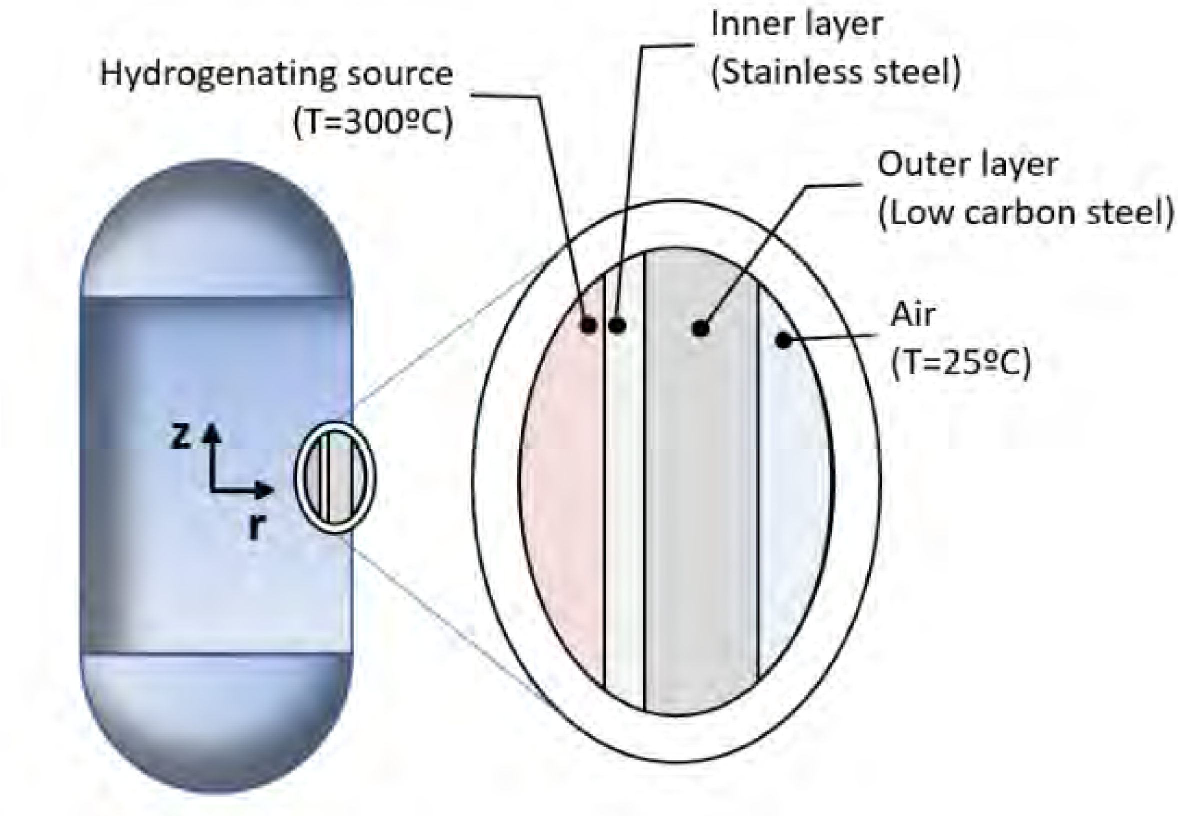


**Figure 3.** Schematic of the structure of the RPV of a WWER 440 Pressurized Water Reactor [Toribio et.al (2016)].

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

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

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

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

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

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

Thank you for your consideration.

Sincerely,

*Digby Macdonald*
Digby Macdonald, Ph.D

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

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Hallie Templeton, Esq., counsel to Friends of the Earth
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**Case No. 23-3884**

**IN THE UNITED STATES COURT OF APPEALS**
**FOR THE NINTH CIRCUIT**

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

v.

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

PACIFIC GAS & ELECTRIC COMPANY,
Intervenor

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

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# PETITIONERS' OPENING BRIEF

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March 20, 2024
**Corrected March 25, 2024**

## CORPORATE DISCLOSURE STATEMENT

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

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## GLOSSARY

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

## JURISDICTIONAL STATEMENT

### A. Hobbs Act Jurisdiction

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

### B. Standing of Petitioners

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

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

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

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

Petitioners also meet the judicial test for organizational standing because:

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

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

## STATEMENT OF ISSUES

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

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

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

## STATUTORY ADDENDUM

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

## STATUTORY AND REGULATORY BACKGROUND

### A. Atomic Energy Act

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

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

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

### B. Implementing Regulations and Guidance for Reactor Vessel Surveillance Programs

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

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

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

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

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

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

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

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

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

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

## STATEMENT OF THE CASE

### I. INTRODUCTION

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

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

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

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

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

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

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

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

## II. STATEMENT OF THE FACTS

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

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

### B. Initial Program for Monitoring Unit 1 Reactor Pressure Vessel

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

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

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

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

### C. NRC promulgation of license renewal rule

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

### D. PG&E’s Supplemental Surveillance Program

#### 1. PG&E application and NRC approval

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

**b. 2006 license amendment for Diablo Canyon Unit 1**

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

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

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

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

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

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

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

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

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

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

Based on these findings, the Safety Evaluation stated:

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

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

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

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

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

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

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

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

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

#### 1. 2008 Extension Decision: from 2007 to 2009

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

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

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

**2. 2010 Extension Decision: from 2010 to 2012**

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

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

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

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

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

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

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

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

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

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

### 3. 2012 Extension Decision: from 2012 to 2022

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Petitioners' hearing request included the following contention:

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

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

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

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

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

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

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

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

**I. Petition for Review**

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

**STANDARD OF REVIEW**

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

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

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

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

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

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

*Public Citizen*, 573 F.3d at 923.

## SUMMARY OF THE ARGUMENT

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

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

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

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

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

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

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

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

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

## ARGUMENT

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

### A. The Extension Decisions Were Unreasonable.

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

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

**B. The Extension Decisions Were Arbitrary and Capricious.**

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

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

## CONCLUSION AND REQUEST FOR RELIEF

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

Respectfully submitted,

*/s/Diane Curran*
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March 20, 2024
**Corrected March 25, 2024**

## PETITIONERS' CERTIFICATE OF COMPLIANCE

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

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

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

Respectfully submitted,

*/s/ Diane Curran*
Diane Curran

March 25, 2024

## PETITIONERS' CERTIFICATE REGARDING IDENTICAL COPIES

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

Respectfully submitted,

*/s/ Diane Curran*
Diane Curran

March 25, 2024

# ATTACHMENT E

# DCISC

## DIABLO CANYON INDEPENDENT SAFETY COMMITTEE

**COMMITTEE MEMBERS**

**WEBSITE** - WWW.DCISC.ORG

ROBERT J. BUDNITZ
PETER LAM
PER F. PETERSON

**VIA EMAIL ONLY**

February 29, 2024

Dear Dr. Macdonald:

We regret that you were unable to attend the February public meeting of the Diablo Canyon Independent Safety Committee where the topic of Diablo Canyon Unit 1's embrittlement was discussed. We are in receipt of your "declarations" on this topic and have reviewed them in our assessment of Unit 1. Our review was contained in a Report in two parts[1] which has been posted on the Diablo Canyon Independent Safety Committee website since February 1, 2024. Part 1 of the Report addressed the general topic of embrittlement and public concerns while Part 2 contained an assessment of Diablo Canyon Unit 1 embrittlement status. The Report was prepared by Dr. Mark Kirk, a recognized expert in the field of radiation embrittlement of reactor vessels.

Due to your inability to attend the public meeting on February 21-22, 2024 and not being able to provide comments on the Report, we are offering you another opportunity to officially comment in writing on the current status of Unit 1's embrittlement. Our concern is focused only on the current status and future condition for the continued operation of Unit 1 for the next 5 to possibly 20 years.

Thus, we are requesting your specific comments on Part 2 of Dr. Kirk's Report, specifically Chapters 2 and 3, which summarize his technical evaluation of Unit 1's current and future embrittlement, and how this compares with NRC screening criteria. You might want to refer to Part 2 of the Report (Chapter 2) in which Dr. Kirk describes the methodology used in the evaluation in more detail. We are specifically interested in the current and future embrittlement status. Making your comments specific and focused would be extremely helpful. For example, it would be good if you could draw our attention to specific statements, data or numbers in those sections of Dr. Kirk's Report that you feel are factually incorrect or incorrectly interpreted, explaining why you feel Dr. Kirk is in error.

In order for us to provide a timely review for the Committee, we request your written comments be received by email to info@dcisc.org by April 1, 2024 to enable a thorough evaluation of your comments in time for the next public meeting of the Committee on June 20-21, 2024.

As we were informed by Ms. Linda Seeley that you have been dealing with some health issues recently, on behalf of the Committee we send you our very best wishes for a quick and complete recovery.

Sincerely,

Robert W. Rathie
DCISC Assistant Legal Counsel

cc: Ms. Linda Seeley, San Luis Obispo Mothers for Peace (lindaseeley@gmail.com)
Hallie Templeton, Esq., Friends of the Earth (htempleton@foe.org)

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[1] An Evaluation of the Status of Diablo Canyon Unit 1 With Respect to Reactor Pressure Vessel Condition Monitoring and Prediction – Part 1 Addressing Public Concerns and Part 2 Evaluation of Diablo Canyon Unit 1 Embrittlement.

# DIABLO CANYON INDEPENDENT SAFETY COMMITTEE

## AGENDA TRANSMITTAL FORM

| | |
|---|---|
| MEETING DATE: | June 20, 2024 |
| AGENDA ITEM: | XV |
| AGENDA TITLE: | Report on Evaluations of Public Concerns on Studies by DCISC Consultant Dr. Mark T. Kirk Evaluating Unit 1 Reactor Pressure Vessel Integrity; Committee Consideration of Endorsing the Conclusions of Dr Kirk's Studies. |

CONSENT [ ] ACTION [X] INFORMATION [X]

Staff Summary: Unit 1 Reactor Vessel Integrity Study.

This work was performed by Dr. Mark T, Kirk as a Consultant to the Diablo Canyon Independent Safety Committee to review the status of Unit 1's reactor pressure vessel integrity which has come under question. Mr. Bruce Severance, a member of the public, and also the groups San Luis Obispo Mothers for Peace (SLOMFP) and Friends of the Earth (FOE) have raised concerns through their consultant Dr. Digby Macdonald alleging that the Unit 1's reactor pressure vessel has exceeded its lifetime based on review of certain licensing documents and information. In its recent Safety Evaluation Report issued in 2011 as part of Diablo Canyon's original license renewal application, the Nuclear Regulatory Commission accepted Diablo Canyon's assessment of Unit 1's reactor pressure vessel indicating that it was acceptable for the additional 20-year license extension after expiration of the original operating licenses for the power plant in 2024 (Unit 1) and 2025 (Unit 2) under aging management program.

At the February 2024 public meeting the Committee and the public received an extensive report from Dr. Kirk on this topic and on Dr. Kirk's written report to the DCISC of January 26, 2024, entitled "An Evaluation of the Status of Diablo Canyon Unit 1 With Respect to Reactor Pressure Vessel Condition Monitoring and Prediction" which was presented in two parts comprised of "Part 1 Addressing Public Concerns" and "Part 2 Evaluation of Diablo Canyon Unit 1 Embrittlement." The full Report has been available on the DCISC website since February 1, 2024. At the February 2024 public meeting the Committee took action to accept Dr Kirk's report and discussed and directed that consideration of endorsing the conclusions in Dr. Kirk's Report be deferred to a later public meeting.

Following the February public meeting, in response to the Committee's invitation, on April 15, 2024, Mr. Severance provided his Comments to the Diablo Canyon Independent Safety Committee raising a number of issues with Dr Kirk's report and the discussion at the February 2024 DCISC meeting. On May 28, 2024, Dr. Kirk's Replies to Mr. Severance's Comments were posted to the Committee website with Mr. Severance's Comments included as an appendix.

RECOMMENDED COMMITTEE ACTION:

Discussion and consider endorsing the conclusions in Dr Kirk's report of January 26, 2024, as stated on pages 3, 4 and 5 of Part 2 of that Report and take such other action and/or provide direction as may be appropriate through the adoption of Resolution 2024-07. Whatever action taken by the Committee on this matter will form a part of the Committee's 34th Annual Report.