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

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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 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

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[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.

Citations to SLOMFP and FOE Concerns

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 1×10^{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,

[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].

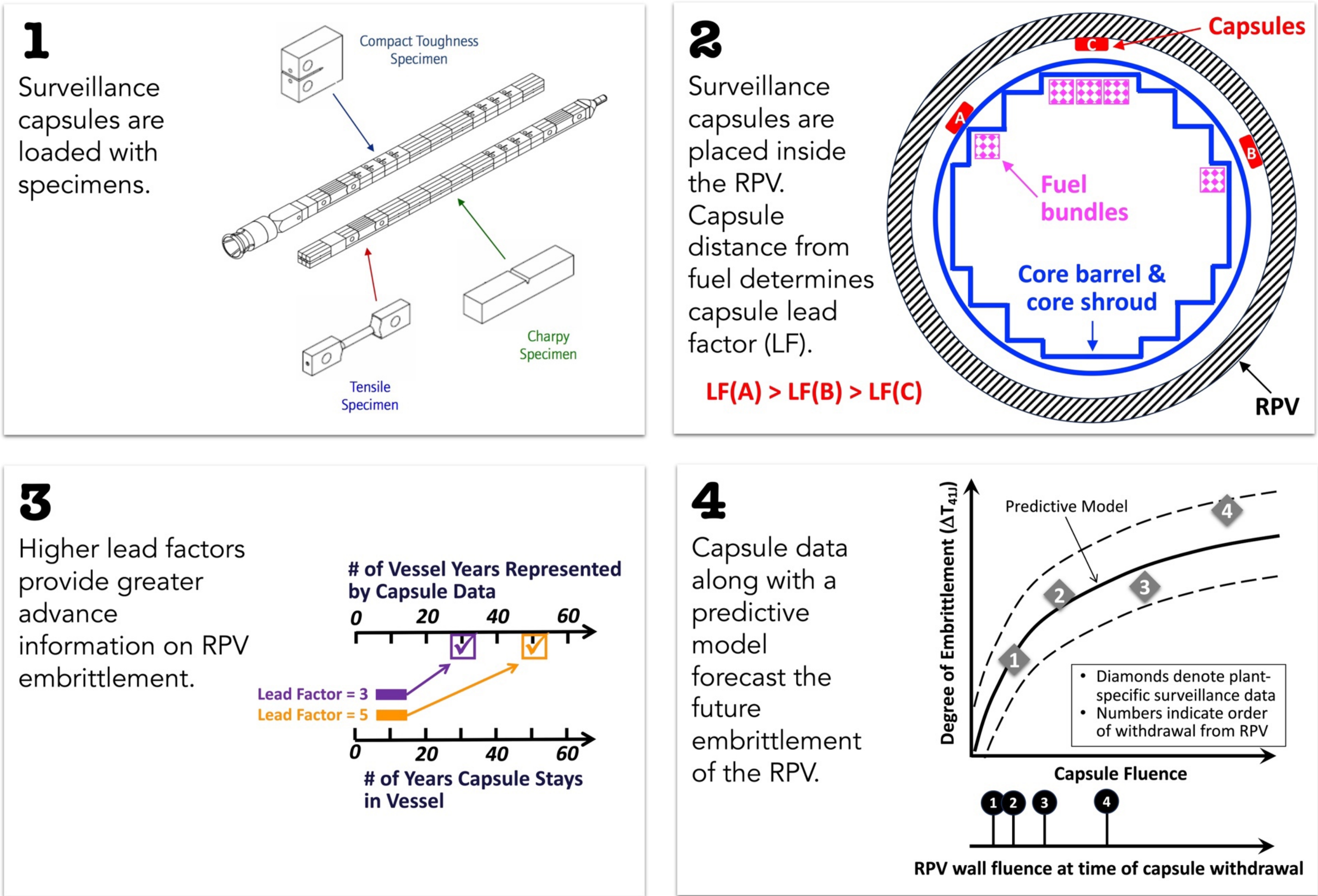

Figure 1. Illustration of the design and conduct of a nuclear RPV surveillance program.

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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 5×10^{22} n/m^2 (5×10^{18} n/cm^2), or at the time when the highest predicted ΔRT_{NDT} of all encapsulated materials is approximately 28°C (50°F), whichever comes first.

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

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

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

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

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.

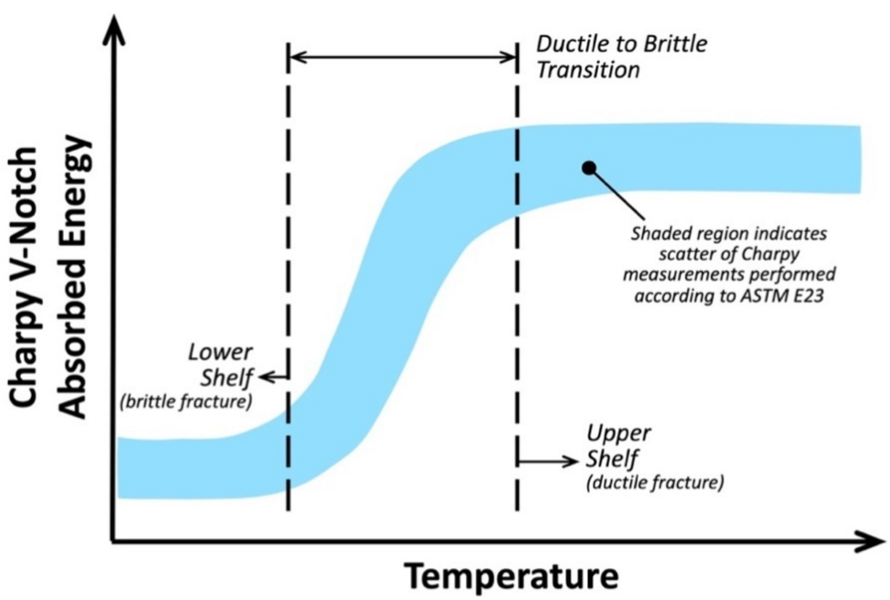

Figure 3. Schematic illustration of the effect of the ductile to brittle transition of a ferritic steel on toughness properties.

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

[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.

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[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.

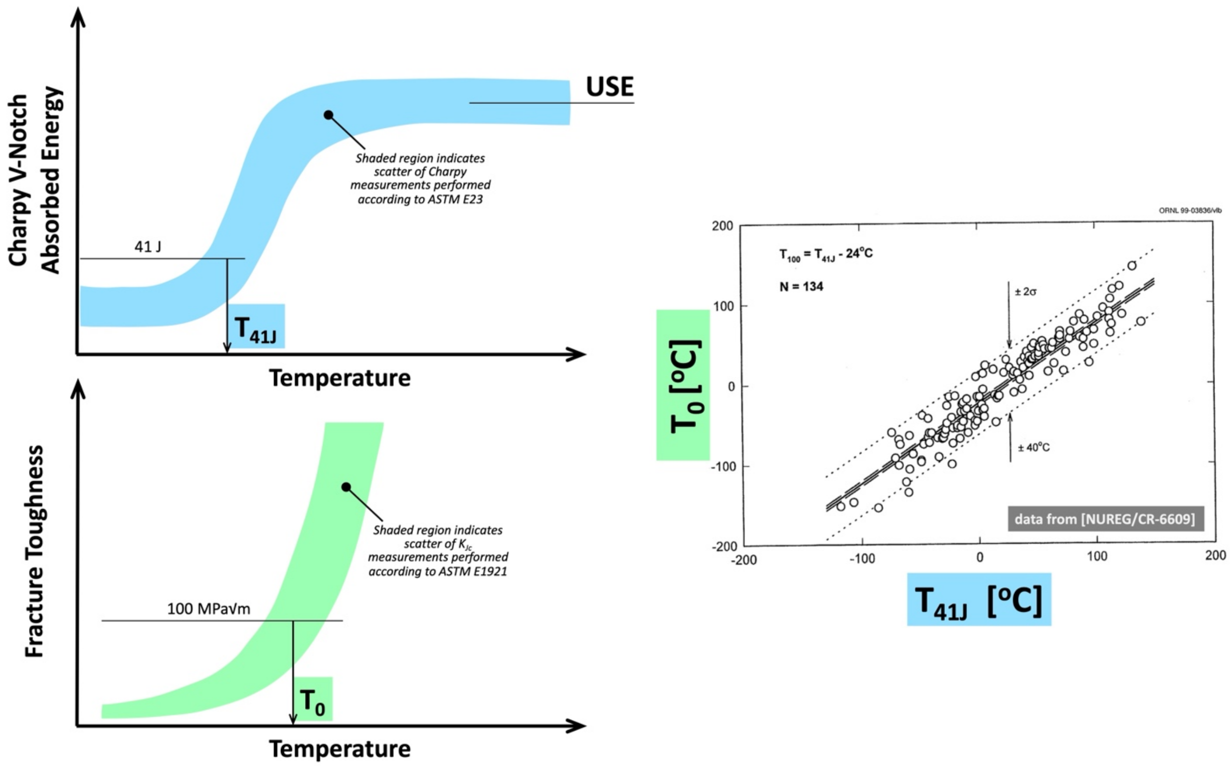

Figure 4. Comparison of Charpy impact toughness and fracture toughness data.

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 (ΔT_0 and ΔT_{41J}) change with increasing neutron exposure as plants age.

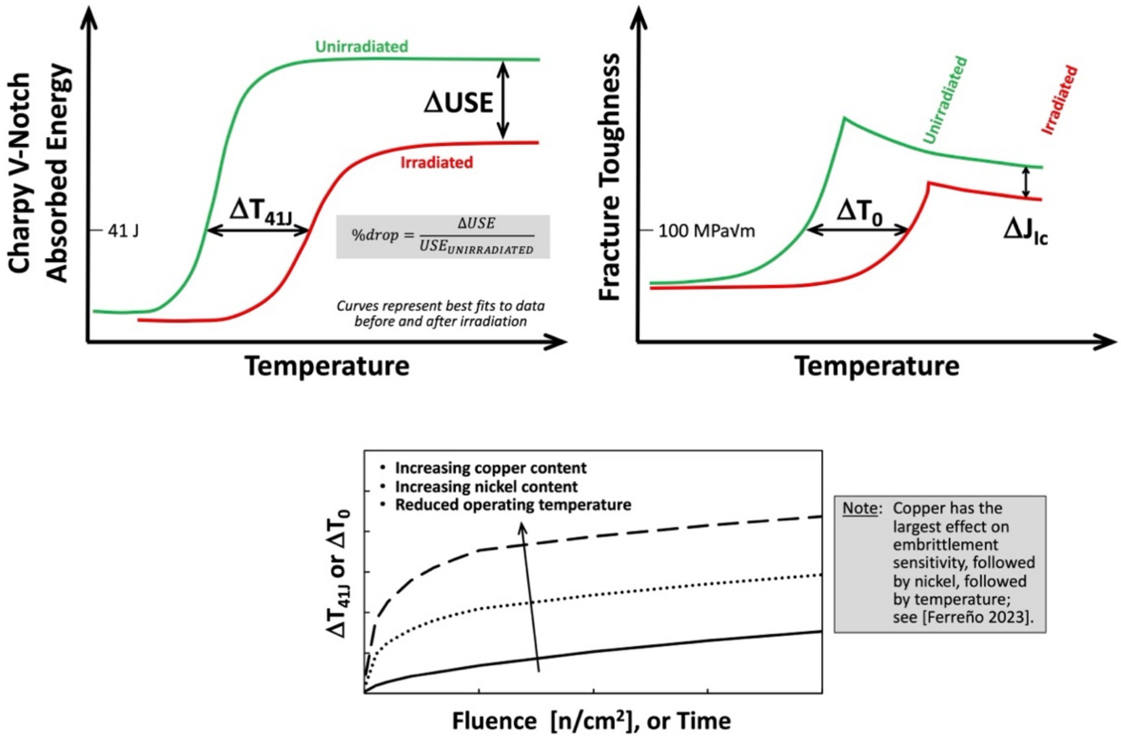

Figure 5. Effect of embrittlement on the fracture toughness and Charpy impact toughness properties of a RPV steel.

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 (ΔT_{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 [ΔT_{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.

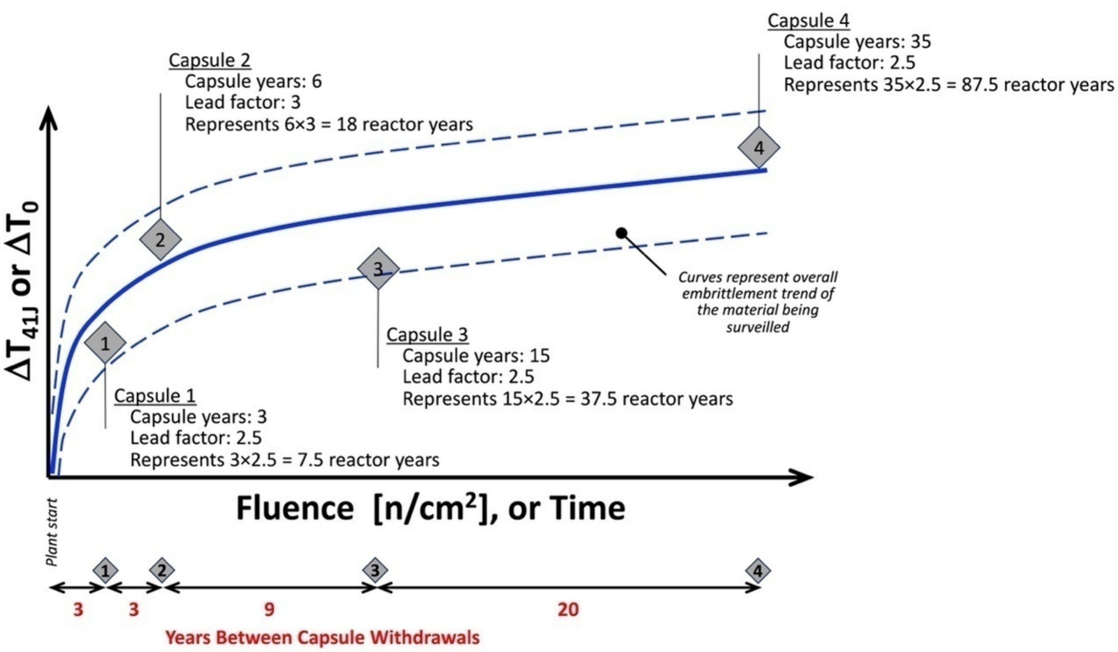

Figure 6. Illustration of how surveillance data accumulates over time.

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.

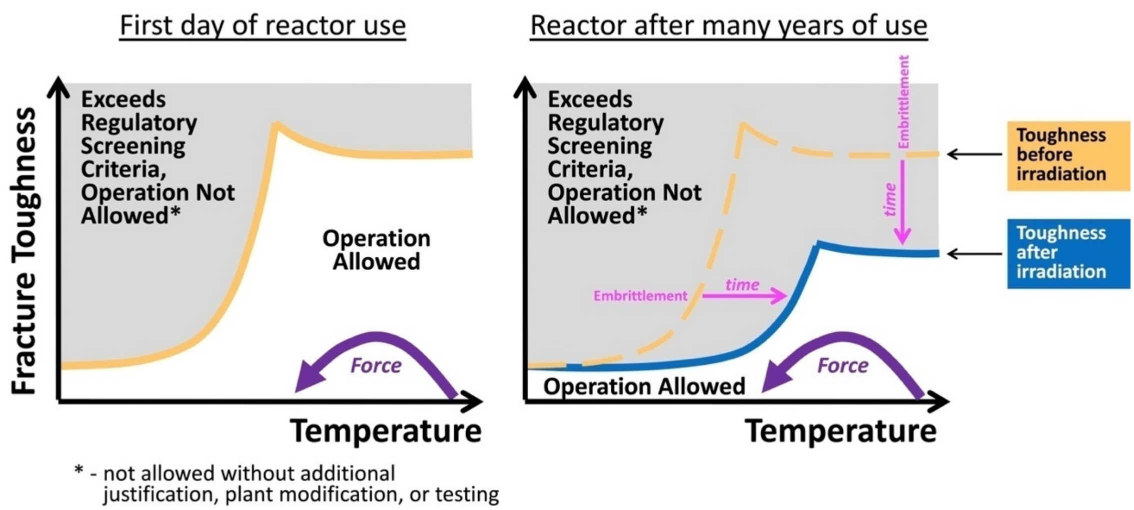

Figure 7. Illustration of how embrittlement forecasts are used to predict fracture toughness, and how fracture toughness compares to the force applied on the RPV structure by the operating temperature and pressure conditions.

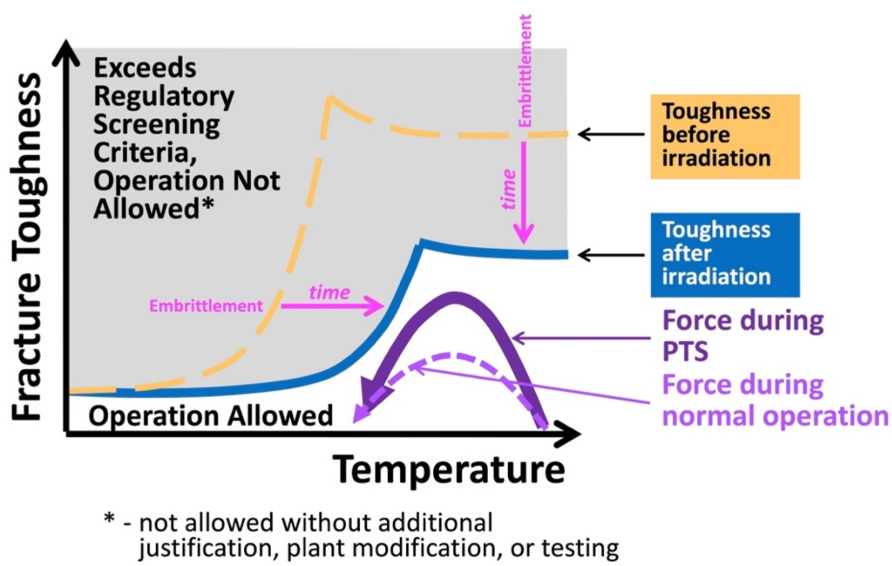

Figure 8. Illustration of how the amount of force applied on the RPV structure differs between normal operations and accident (PTS) conditions, and how these force levels compare to the fracture toughness.

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 ΔT_{41J} and on ΔUSE for future conditions. ΔT_{41J} procedures are identical in both documents while ΔUSE procedures appear only in [RG1.99R2]. An estimate of ΔT_{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). ΔUSE is needed to assess compliance with the USE requirements of [10CFR50-AppG] (see Section 2.5.3.1). The ΔT_{41J} and ΔUSE 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. ΔT_{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 ΔT_{41J} is described in [Randall 1986]. That ETC estimates $\Delta T_{41J}$from exposure, composition, and categorical variables as follows:

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

The $\frac{5}{9}$ factor does not appear in the NRC equations, but is used here to express ΔT_{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 1×10^{19} n/cm^2).

The uncertainty in the estimate of ΔT_{41J} calculated using equation (2-1) is called σ_Δ, which is the standard deviation of residuals reported for equation (2-1) in [Randall 1986]. σ_Δ 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 <u>must</u> 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 [ΔT_{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 ΔT_{41J} data.
- If data from other plants are used (see Section 2.4.4) the measured ΔT_{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 ΔT_{41J}.

The statement of Criteria (C) in [10CFR50.61] and [RG1.99R2] is not explicit regarding the number of ΔT_{41J} data that may lie outside the scatter limits expressed by the criteria. For example, if only one ΔT_{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 ΔT_{41J} data outside the permissible scatter range are considered not credible. [Wichman 1998] also gives an example where 17% of ΔT_{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).

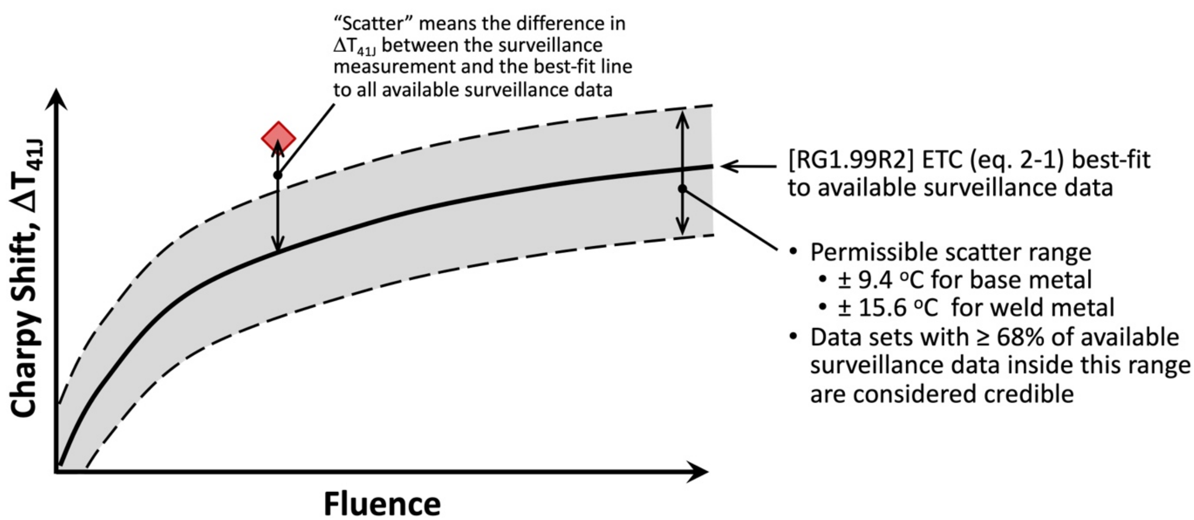

Figure 9. Illustration of credibility Criteria (C).

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 ΔT_{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 ΔT_{41J} values from other plants. The chemistry adjustment is performed using equation (2-1) to represent the effect of both copper and nickel on ΔT_{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 ΔT_{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 ΔT_{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.

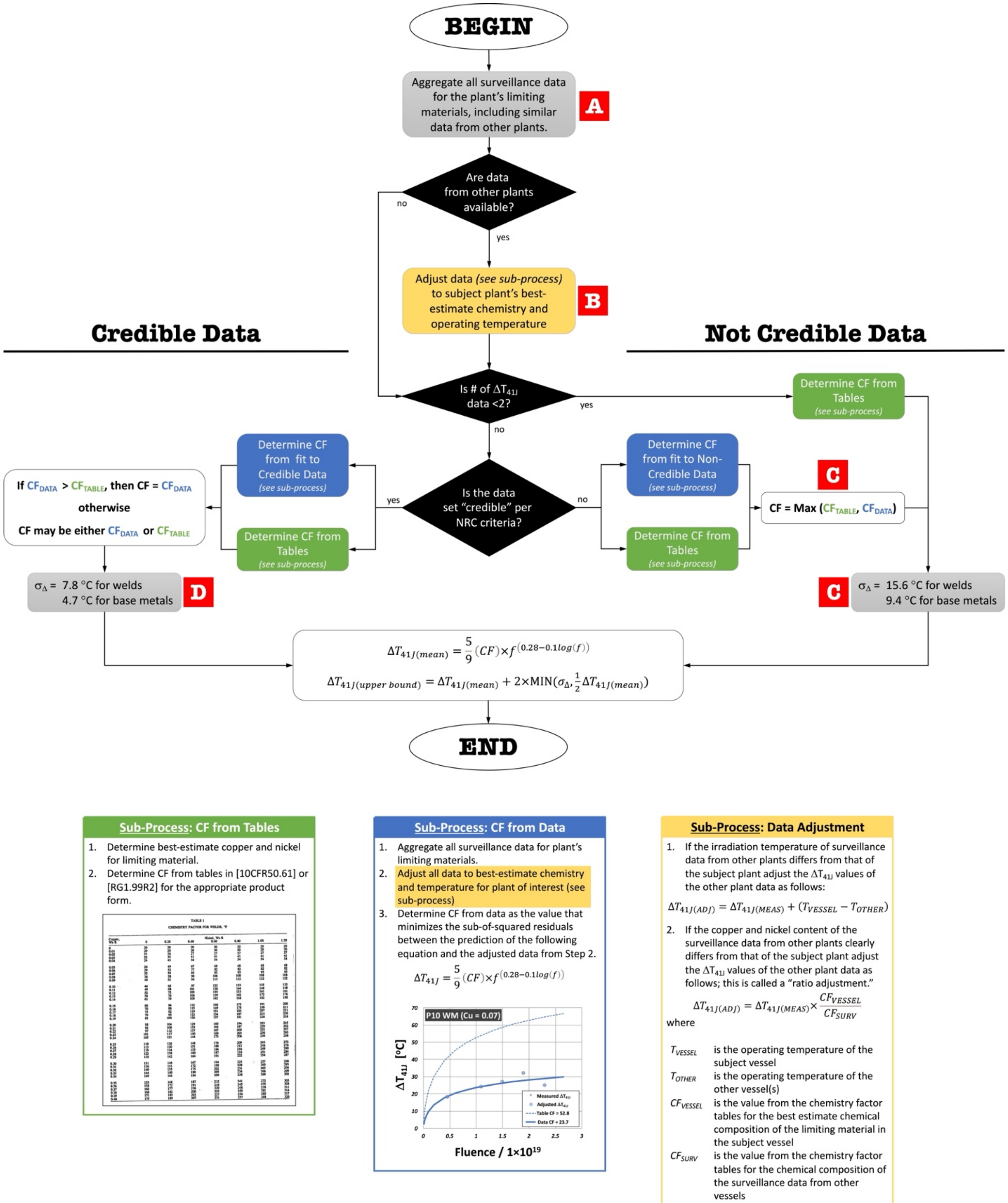

Figure 10. Diagram showing ΔT_{41J} forecasting procedures for both credible and not-credible data as required by [10CFR50.61] and as described by [RG1.99R2] and [Wichman 1998].

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]:

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

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

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

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

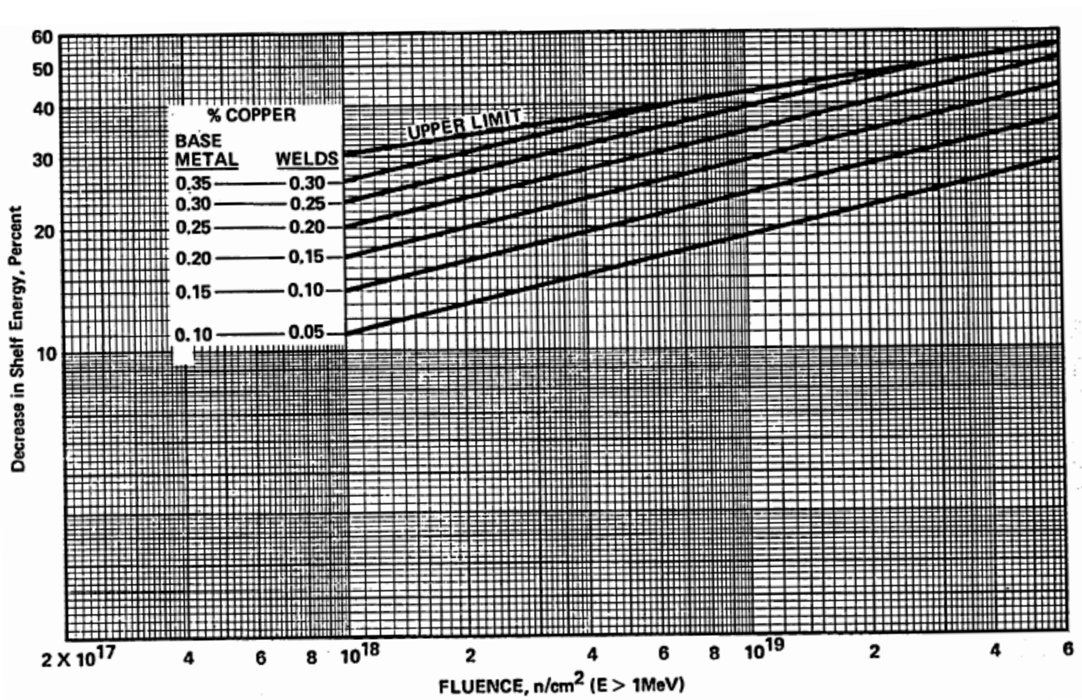

Figure 11. Graphical relation for upper-shelf energy drop from RG1.99R2.

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 ΔT_{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.

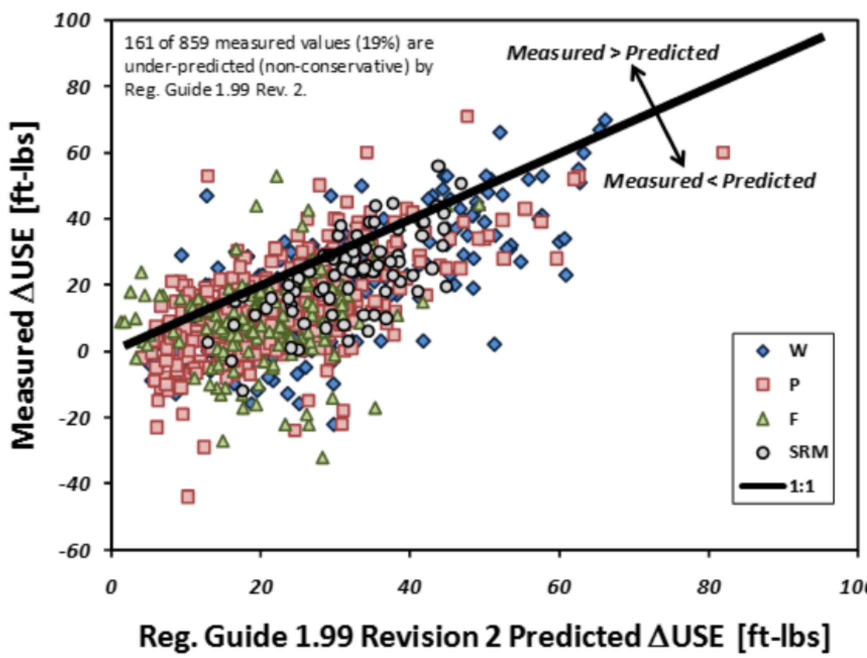

Figure 12. Comparison of the [RG1.99R2] prediction of ΔUSE to measured data, graph from [Widrevitz 2019]. In the legend W=weld, P=plate, F=forging, and SRM=-standard reference material.

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 ΔT_{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 ΔT_{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 α in equation (2-2) be adjusted until the prediction of %drop over-estimates all available data (α 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.

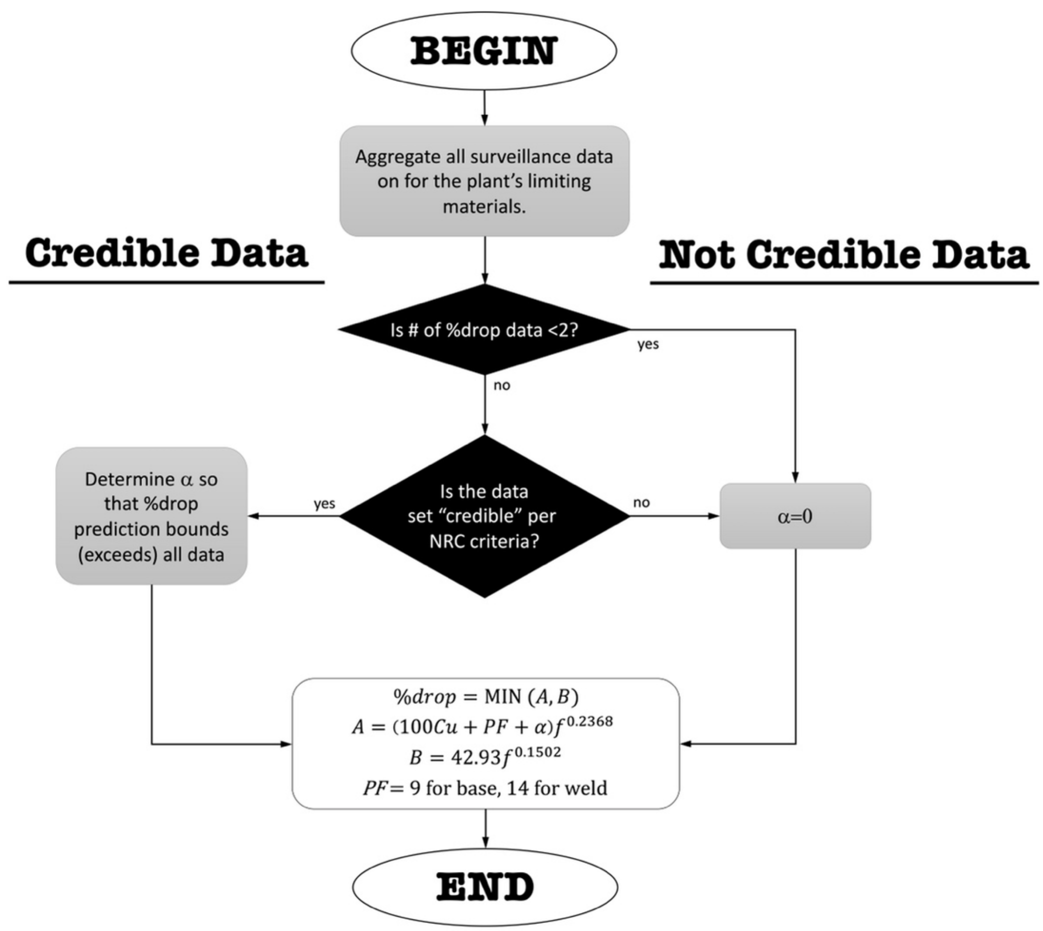

Figure 13. Diagram showing %drop forecasting procedures for both credible and not-credible data as described by [RG1.99R2].

2.4.4. Data to Use

2.4.4.1. To estimate ΔT_{41J}

For ΔT_{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 <u>underlines</u> added for clarity):

> *"To verify that [the embrittlement forecast] for each vessel beltline material is a bounding value for the specific reactor vessel, <u>licensees shall consider</u> 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. <u>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</u> 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* ***<u>and</u>*** *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* ***<u>only</u>***," 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 ΔT_{41J} embrittlement forecasting; the NRC has never objected to this usage.

The other aspect of the process for determining which ΔT_{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 ΔT_{41J} surveillance data from a plant of interest can be stated as follows:

> **Plants must consider all ΔT_{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 ΔT_{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 5×10^{-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 (5×10^{-6} for [10CFR50.61], to 1×10^{-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 $T_{WALL} \le$ 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})		
	$T_{WALL} \le$ 9.5 in.	9.5 in. < $T_{WALL} \le$ 10.5 in.	10.5 in. < $T_{WALL} \le$ 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 1×10^{-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), <u>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</u>.*

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 <u>the underlined text</u> 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.

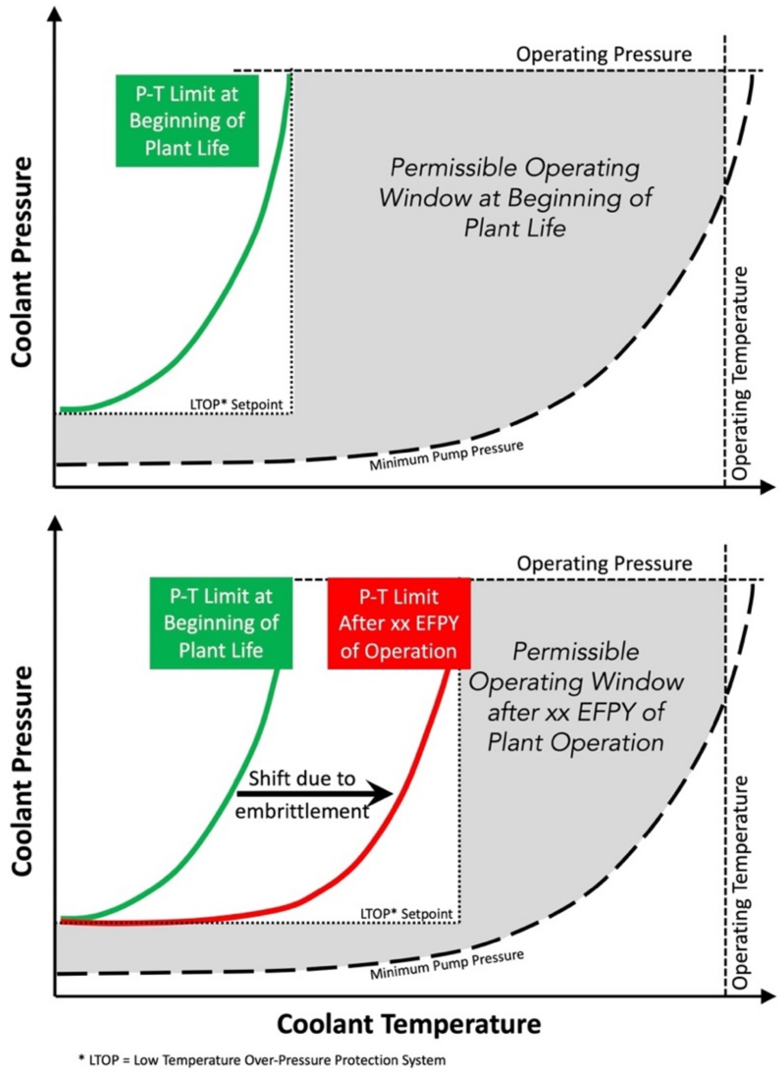

Figure 15. Schematic illustration of the P-T limit curve before (top) and after (bottom) irradiation embrittlement. The curve labeled "minimum pump pressure" depends on plant design and is unchanged by embrittlement; this curve represents the pressure needed to prevent cavitation in the main coolant 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 ΔT_{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 ΔT_{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 ΔT_{41J} due to the more conservative estimation of the *CF* and σ_{Δ} 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 ΔT_{41J} forecasting procedure to obtain a conservative upper-bound estimate of ΔT_{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 ΔT_{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 1.34×10^{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 ΔT_{41J} forecast for the Diablo Canyon Unit 1 RPV was augmented by considering two additional ΔT_{41J} surveillance data collected by the Palisades plant located in Michigan. The data from Palisades extends to a higher fluence (2.36×10^{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 2.36×10^{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 ΔT_{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 ΔRT_{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 ΔRT_{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-boundΔT_{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 ΔT_{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 ΔT_{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.

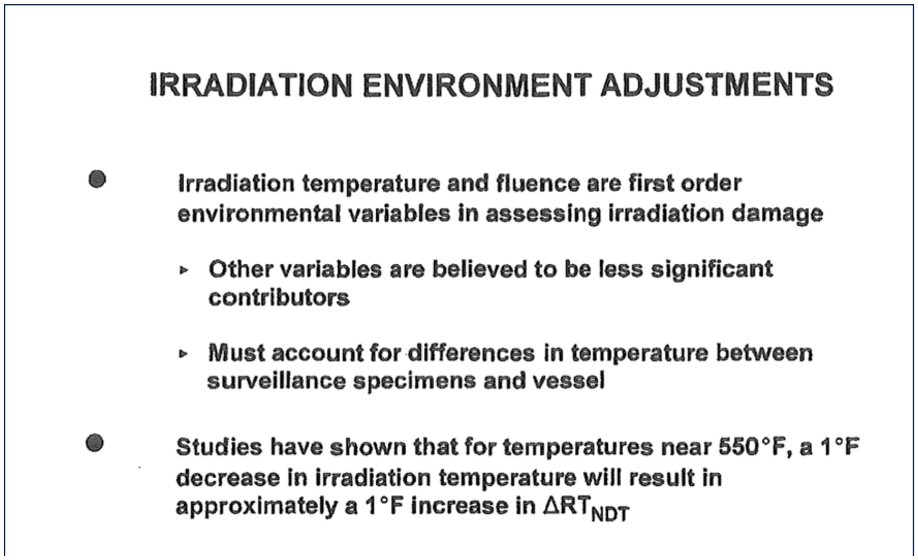

Figure 16. NRC's "degree-per-degree" ΔT_{41J} adjustment, after [Wichman 1998].

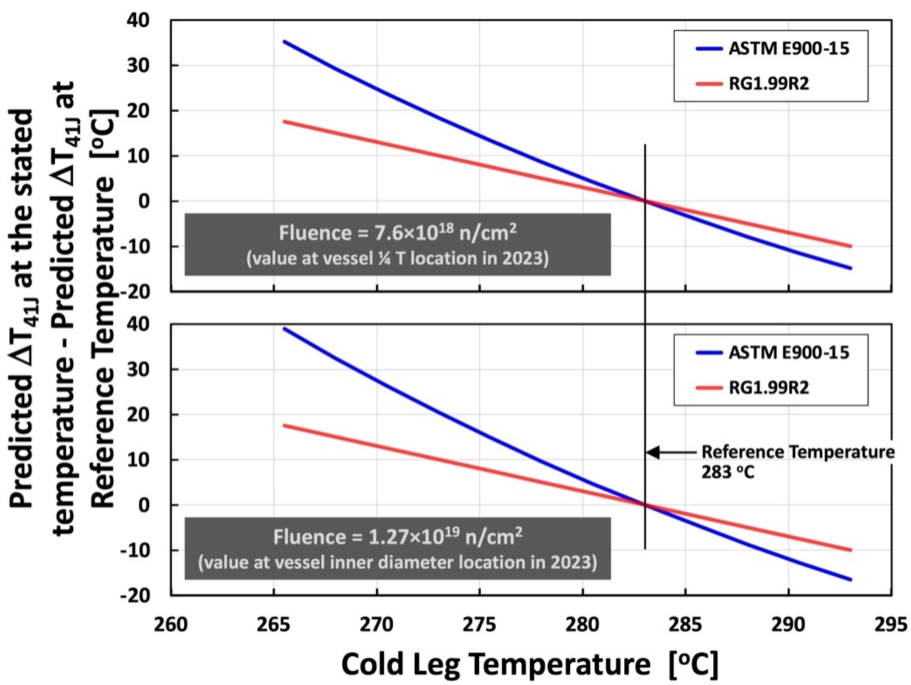

Figure 17. Comparison of the predictions of the NRC's "degree-per-degree" rule that is used with equation (2-1) with the temperature function of ASTM E900-15 ETC for the Diablo Canyon Unit 1 limiting weld material (Cu=0.196, Ni=1, Mn=1.347, P=0.013). An Appendix to this report provides the formula for 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 ΔT_{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 theΔT_{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 ΔT_{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 ΔT_{41Jc} combined with other conservatisms inherent to the assessment process provide protection against making erroneous decisions.

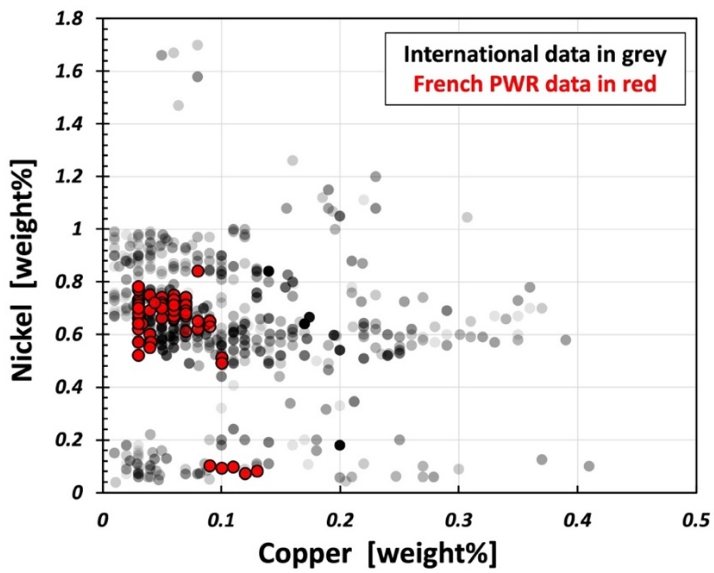

Figure 18. Composition of French RPV steels compared with international data.

Table 1. Comparison of ΔT_{41J} prediction uncertainty between two ETCs.

Product Form	Standard deviation of ΔT_{41J} prediction uncertainty for ASTM E900-15 relative to all international data	Standard deviation of ΔT_{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

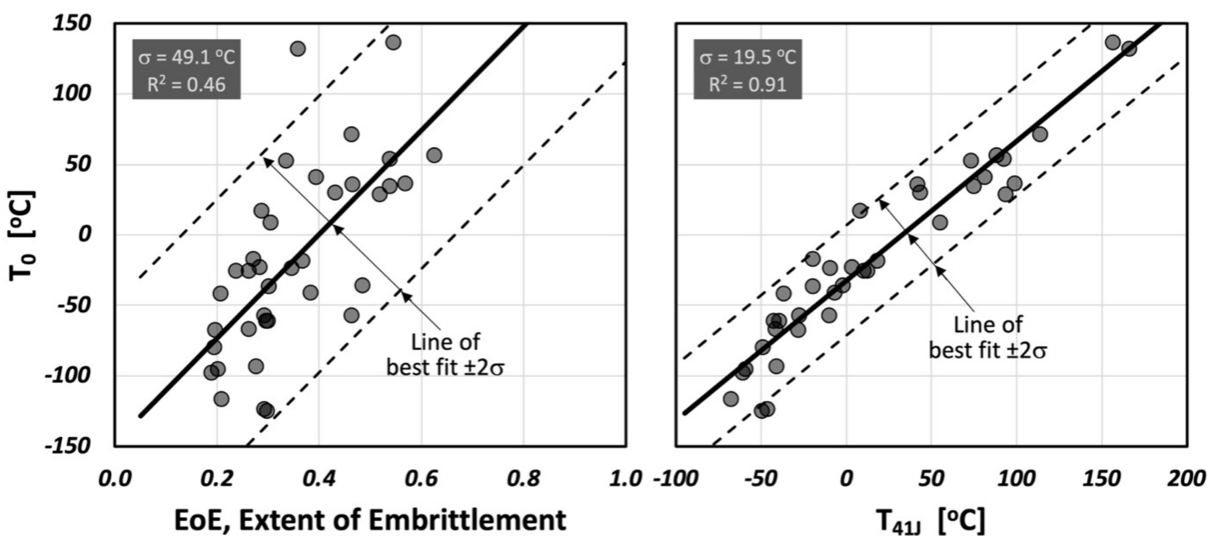

Figure 19. Comparison of the EoE and T_{41J} transition temperature metrics estimated from Charpy data (left and right graph, respectively) to the fracture toughness transition temperature T_0, estimated by ASTM Standard E1921 (vertical axis on both graphs). Data from [EricksonKirk 2009].

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 (σ) value of 49.1 °C for EoE with a σ value of 19.5 °C for ΔT_{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).

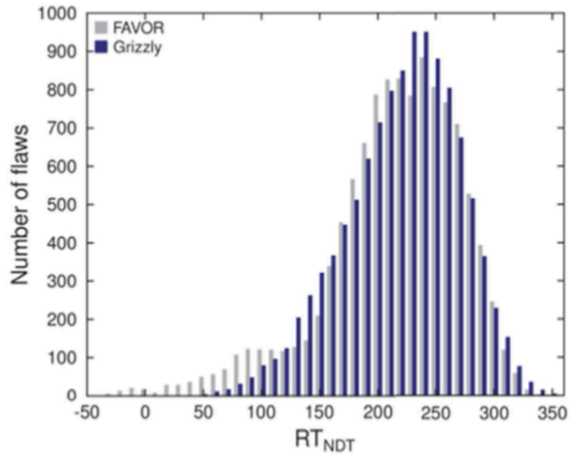

Figure 7: Comparison of RT_{NDT} distributions in the same plate analysis in Grizzly and FAVOR. After Spencer et.al. (2016).

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* $\approx 10^{-5}$ *mm/s (*$\approx 4 \times 10^{-7}$ *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).

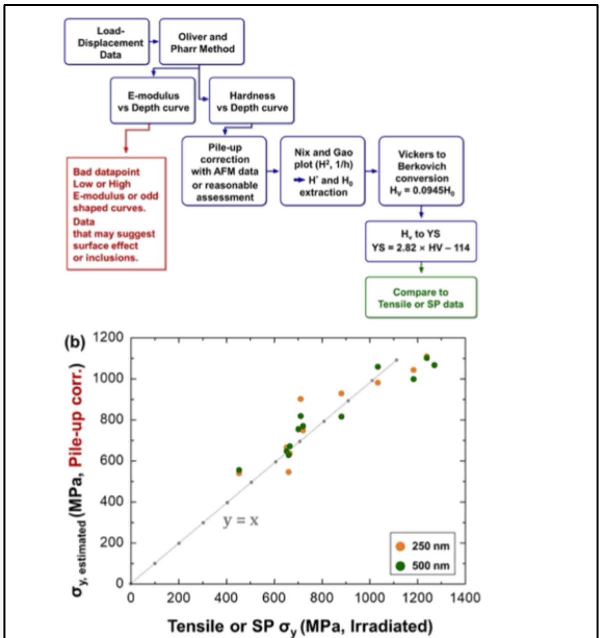

Figure 10: (a) Roadmap of nano indentation techniques. (b) Correlation between tensile test and shear punch test generated data to data obtained from nano hardness.

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 3×10^{19} to 1×10^{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 3×10^{19} to 1×10^{17} n/cm^2

3.5.2.1. SLOMFP and FOE Concerns

What is the justification for reducing the fluence from $3x10^{19}$ 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 ΔT_{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 ΔT_{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 ΔT_{41J} and ΔUSE 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 5.2×10^{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 1×10^{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)	ΔRT_{NDT} (°F)	σ_U[d] (°F)	σ_Δ[f] (°F)	Margin (°F)	RT_{PTS} (°F)
Beltline Materials										
IS Plate B4106-1	1.1	85.3	2.02	1.1917	-10	101.7	0	17	34	126
IS Plate B4106-2	1.1	81.0	2.02	1.1917	-3	96.5	0	17	34	128
IS Plate B4106-3	1.1	55.2	2.02	1.1917	30	65.8	17[e]	17	48.1	144
→ Using non-credible surveillance data	2.1	37.4	2.02	1.1917	30	44.6	17[e]	17	48.1	123
LS Plate B4107-1	1.1	89.8	2.01	1.1904	15	106.9	0	17	34	156
LS Plate B4107-2	1.1	82.2	2.01	1.1904	20	97.9	0	17	34	152
LS Plate B4107-3	1.1	81.4	2.01	1.1904	-22	96.9	0	17	34	109
IS Longitudinal Weld 2-442A & B	1.1	226.8	1.49	1.1104	-56[e]	251.8	17[e]	28	65.5	261
→ Using credible surveillance data	2.1	214.1	1.49	1.1104	-56[e]	237.7	17[e]	14	44.0	226
IS Longitudinal Weld 2-442C	1.1	226.8	0.768	0.9259	-56[e]	210.0	17[e]	28	65.5	220
→ Using credible surveillance data	2.1	214.1	0.768	0.9259	-56[e]	198.2	17[e]	14	44.0	186
LS Longitudinal Weld 3-442A & B	1.1	226.8	1.19	1.0485	-56[e]	237.8	17[e]	28	65.5	247
→ Using credible surveillance data	2.1	214.1	1.19	1.0485	-56[e]	224.5	17[e]	14	44.0	213
LS Longitudinal Weld 3-442C	1.1	226.8	2.01	1.1904	-56[e]	270.0	17[e]	28	65.5	280
→ Using credible surveillance data	2.1	214.1	2.01	1.1904	-56[e]	254.9	17[e]	14	44.0	**243**
IS to LS Circ. Weld 9-442	1.1	172.2	2.01	1.1904	-56[e]	205.0	17[e]	28	65.5	215

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)	ΔRT_{NDT} (°F)	σ_U[d] (°F)	σ_Δ[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, σ_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 σ_Δ = 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 σ_Δ = 28°F for Position 1.1 and with credible surveillance data σ_Δ = 14°F for Position 2.1. However, σ_Δ 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.

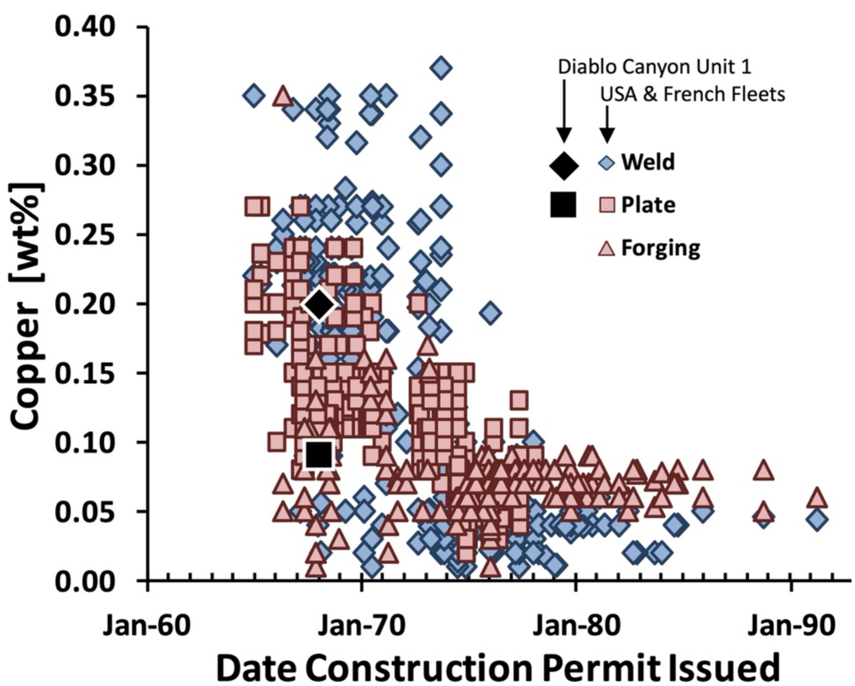

Figure 23. Variation of copper in RPV beltline materials versus plant construction date, after [Kirk 2018b].

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 1.36×10^{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 2.38×10^{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 1.27×10^{19} n/cm^2. The fluence of Capsule B, when withdrawn, will be at least 3.7×10^{19} n/cm^2 while at the end of 60 operating years the vessel fluence at the inner diameter is projected at 2.07×10^{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 3×10^{19} to 1×10^{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 ΔT_{41J} and ΔUSE 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 1×10^{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 1×10^{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) *Yokosuka, Kanagawa, Japan*	Guest Research Fellow
2018-date	Phoenix Engineering Associates, Inc. (PEAI) *Unity, New Hampshire, USA*	Principal Engineer
1999-2018	Nuclear Regulatory Commission *Rockville, Maryland, USA*	Senior Materials Engineer
1997-1999	Westinghouse Electric Corporation *Pittsburgh, Pennsylvania, USA*	Senior Engineer
1992-1997	Edison Welding Institute *Columbus, Ohio, USA*	Energy & Chemical Industry Team Leader
1981-1992	David Taylor Research Center (U.S. Navy) *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.
- 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
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- 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.
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- *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.
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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 ΔT_{41J}

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

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

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

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

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

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].