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

Part 2: Evaluation of Diablo Canyon Unit 1 Embrittlement

Consultant's report to the Diablo Canyon Independent Safety Committee

Submitted by
Mark Kirk
Phoenix Engineering Associates Inc.
Unity, New Hampshire, USA

Date: 26 January 2024

*P*hoenix
*E*ngineering
*A*ssociates
*I*nc.

Disclaimer

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

Executive Summary

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

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

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

The information contained herein supports the following conclusions:

Concerning Surveillance Testing

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

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

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

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

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

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

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

Concerning Additional Testing

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

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

Table of Contents

List of Figures

List of Tables

Glossary & Acronyms

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

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

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

1. Background and Objective

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

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

---

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

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

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

The remainder of this report is structured as follows:

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

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

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

This Chapter provides the following information:

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

2.1. Diablo Canyon Unit 1 Reactor Vessel Surveillance Program

2.1.1. Original Program

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

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

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

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

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

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

Notes

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

2.1.2. Supplemental Program

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

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

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

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

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

---

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

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

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

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

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

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

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

Notes

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

2.2. Surveillance Data Summary

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

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

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

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

Identifier	Copper (wt%)	Nickel (wt%)	Unirradiated RT_{NDT} (°C)	RT_{NDT} Type	Unirradiated USE (J)	Maximum Inner Diameter Fluence ÷ 1×10^{19} n/cm^2		Maximum ¼T Fluence ÷ 1×10^{19} n/cm^2	
						32 EFPY	54 EFPY	32 EFPY	54 EFPY
PLATE MATERIALS									
Intermediate Shell B4106-1	0.125	0.53	-23	NB-2331	157	1.23	2.02	0.73	1.20
Intermediate Shell B4106-2	0.12	0.5	-19	NB-2331	155	1.23	2.02	0.73	1.20
Intermediate Shell B4106-3	0.086	0.476	-1	Generic	104	1.23	2.02	0.73	1.20
Lower Shell B4107-1	0.13	0.56	-9	NB-2331	149	1.22	2.01	0.73	1.20
Lower Shell B4107-2	0.12	0.56	-7	NB-2331	140	1.22	2.01	0.73	1.20
Lower Shell B4107-3	0.12	0.52	-30	NB-2331	157	1.22	2.01	0.73	1.20
Surveillance (B4106-3)	0.086	0.476	--	--	160	--	--	--	--
WELD MATERIALS									
Intermediate Shell Longitudinal 2-442 A, B, and C (Heat 27204)	0.203	1.018	-49	Generic	123	A,B: 0.905 C: 0.463	A, B: 1.49 C: 0.768	A, B: 0.539 C: 0.267	A, B: 0.888, C: 0.458
Lower Shell Longitudinal 3-442 A, B, and C (Heat 27204)	0.203	1.018	-49	Generic	123	A, B: 0.721 C: 1.22	A, B: 1.19 C: 2.01	A, B: 0.43 C: 0.727	A, B: 0.709 C: 1.198
Intermediate to Lower Shell Circumferential 9-442 (Heat 21935)	0.183	0.704	-49	Generic	148	1.22	2.01	0.73	1.20
Surveillance (Heat 27204)	0.198	0.999	--	--	123	--	--	--	--

Notes

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

[4] The "extended beltline" is the region of the RPV above and below the active core where the neutron fluence is projected to exceed 1×10^{17} n/cm^2 [NRC 2014]. Embrittlement measurable by the Charpy parameters ΔT_{41J} and USE can begin to be detected above this fluence.

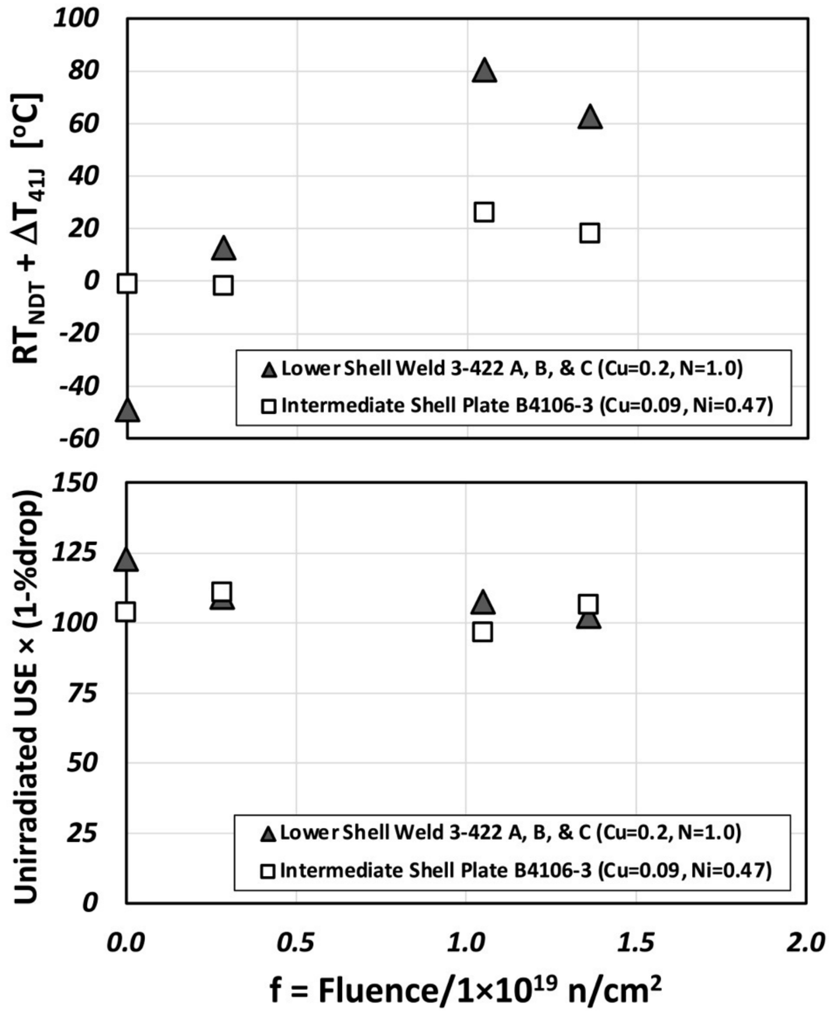

Figure 1. Comparison of the response to embrittlement of the plate and weld materials used to fabricate the Diablo Canyon Unit 1 RPV.

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

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

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

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

Plant	Capsule	Fluence ÷ 1×10^{19} n/cm^2	Time Averaged Cold Leg Temperature (°C)	T_{41J} (°C)	USE (J)	Cu (wt%)	Ni (wt%)
Diablo 1	--	0	--	-54.2	123.4	0.21	0.98
Diablo 1	S	0.283	284.4	7.3	109.8	0.21	0.98
Diablo 1	Y	1.05	283.3	75.0	81.3	0.21	0.98
Diablo 1	V	1.36	282.8	57.5	89.5	0.21	0.98
Palisades	--	0	--	-40.7	147.0	0.194	1.067
Palisades	SA-60	1.50	279.4	99.9	71.9	0.194	1.067
Palisades	SA-240	2.38	280	108.1	72.9	0.194	1.067

2.3. Capsule Withdrawal Schedule

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

2.3.1. Summary of Withdrawal Schedule Changes since 2003

The original schedule for Capsule B withdrawal was established in 2003 by Table 7-1 of the Capsule V surveillance report [WCAP-15958]; indicating Capsule B removal and testing after 20.7 EFPY at a fluence of approximately 2.91×10^{19} n/cm^2, this corresponding to the end of fuel cycle 15 which was projected to occur in 2009. However, in 2008 PG&E requested approval to

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

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

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

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

> *The withdrawal and testing of Capsule V … fulfilled the third and final recommendation of ASTM E 185-70 for the current Diablo Canton Unit 1 operating license. Capsule V, upon removal and testing at 32.3 EFPY, had an accumulated neutron fluence of 1.37×10^{19} n/cm^2, which is representative of the RPV fluence value at end of license (EOL). Therefore, the proposed delayed removal of Capsule B does not deviate from the licensee's current RPV materials surveillance program requirements. The change in withdrawal schedule for Capsule B to be withdrawn at a fluence approximately equivalent to 75.8 EFPY for the RPV will provide high fluence data of the RPV useful for license renewal application.*

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

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

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

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

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

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

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

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

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

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

2.3.2. Evaluation

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

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

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

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

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

---

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

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

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

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

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

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

3. RPV Safety Evaluations

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

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

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

3.1. Forecast of Adjusted Reference Temperature (ART)

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

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

where

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

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

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

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

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

3.1.1. Consultant's Evaluation

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

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

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

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

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

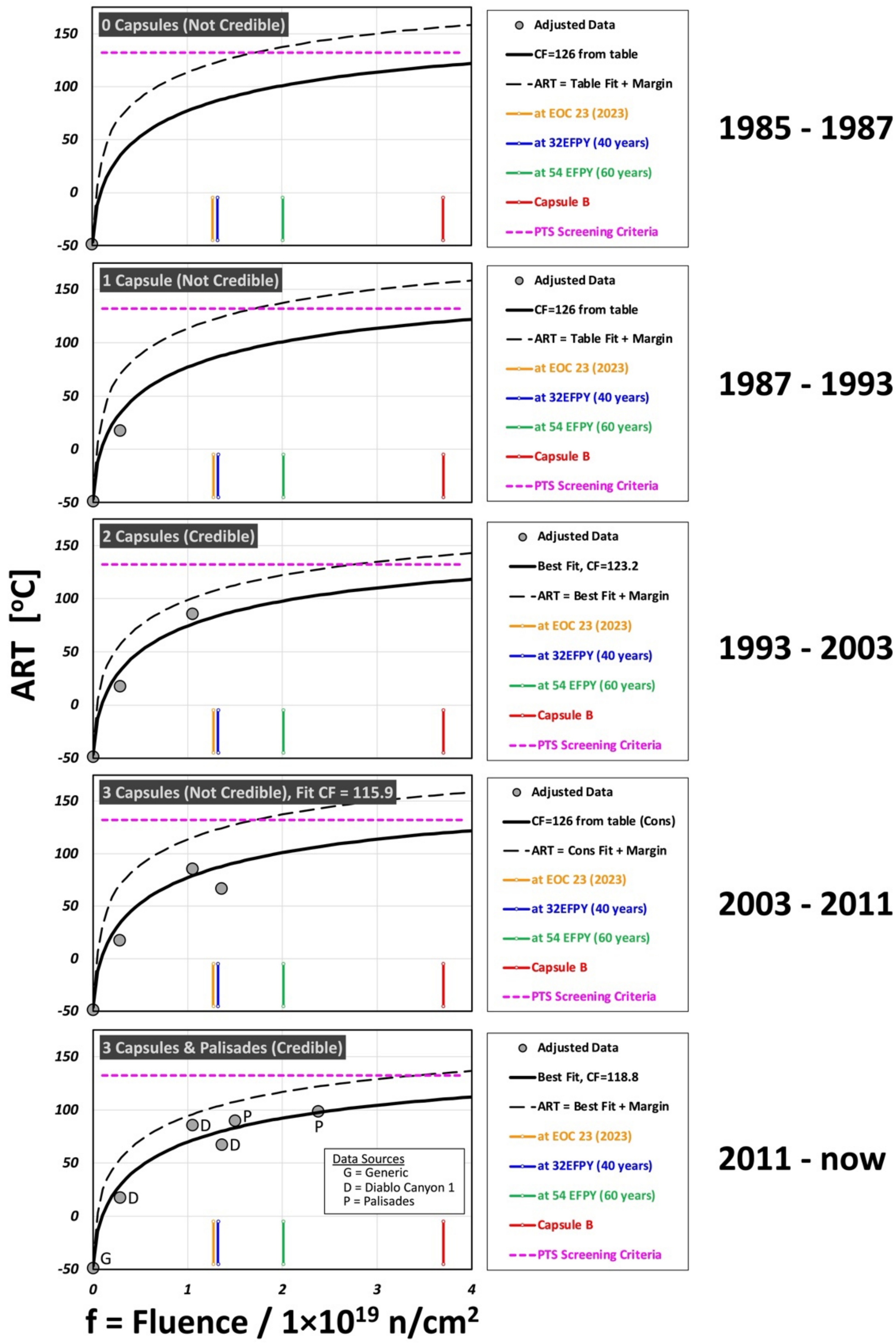

Figure 2. Evolution of ART predictions for the limiting weld in Diablo Canyon Unit 1 with time.

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

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

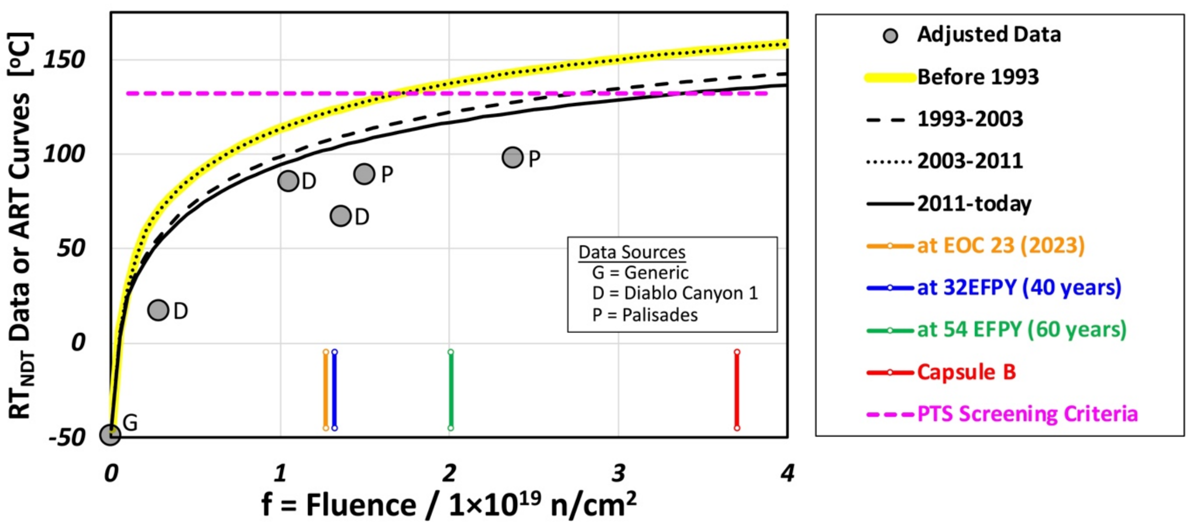

Figure 3. Summary of ART curves for different operating periods.

3.1.2. Comparison with PG&E Estimates Since 2003

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

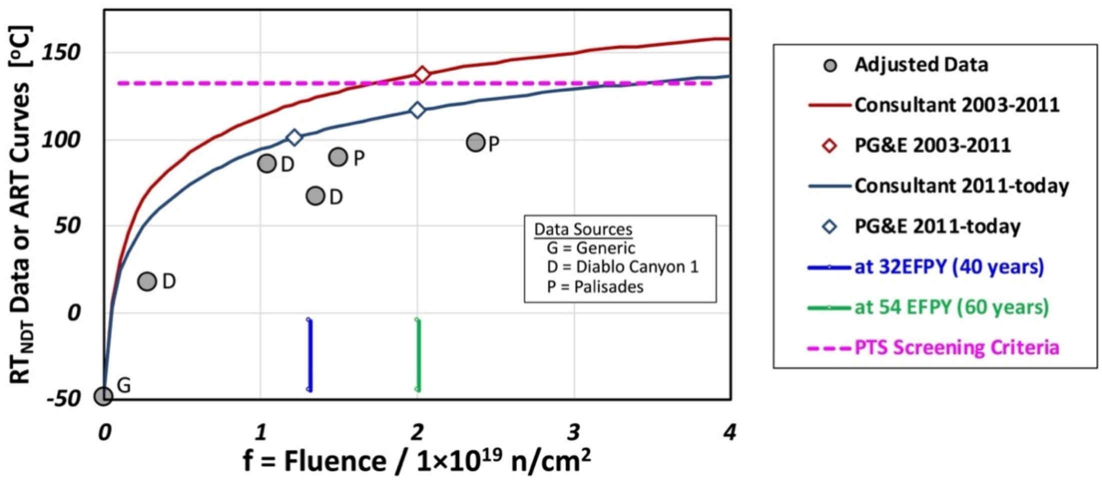

Figure 4. Comparison of consultant's ART estimates (curves) with those of PG&E (diamonds) for the timeframe 2003-2011 [Diablo LRA 2009] and 2011-today [WCAP-17315]. The NRC's required process for adjusting data is described in the Part 1 report, Section 2.4.2.2 [Kirk 2024].

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

3.2. Forecast of Upper Shelf Energy (USE)

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

3.2.1. Consultant's Evaluation

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

The history of the %drop prediction is as follows:

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

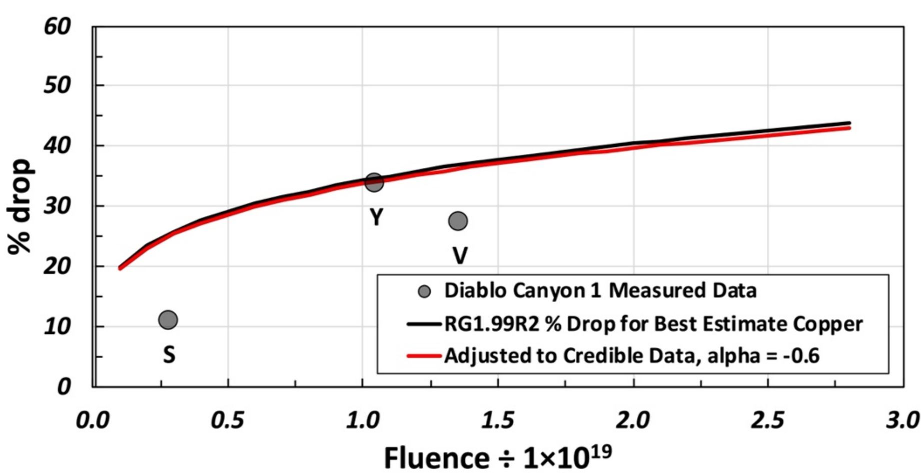

Figure 5. %drop values calculated from Diablo Canyon Unit 1 data for the surveillance weld.

3.2.2. Comparison with PG&E Estimates Since 2003

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

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

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

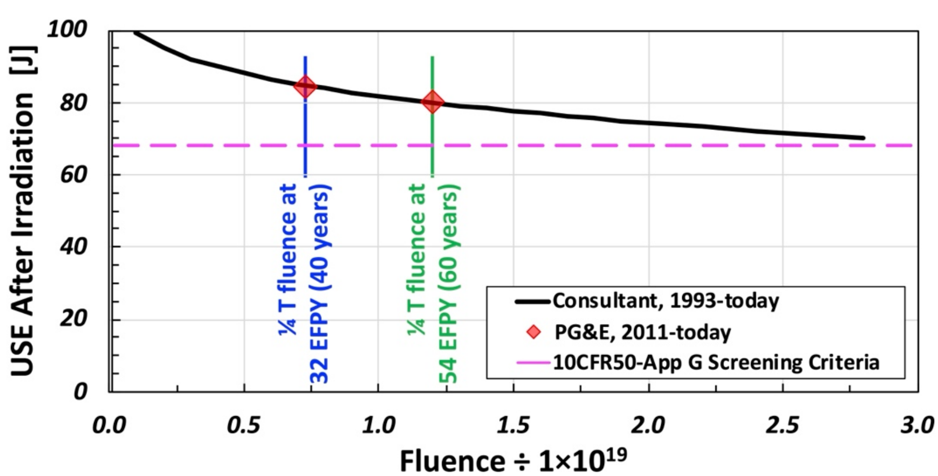

Figure 6. Comparison of consultant's and PG&E estimates of irradiated USE to the NRC's USE screening criteria.

3.3. Summary

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

4. Supplemental Analysis of Unit 1 Embrittlement Trends

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

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

4.1. Analysis of Transition Temperature (ΔT_{41J}) Data

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

4.1.1. Embrittlement Trends Exhibited by Similar Data

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

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

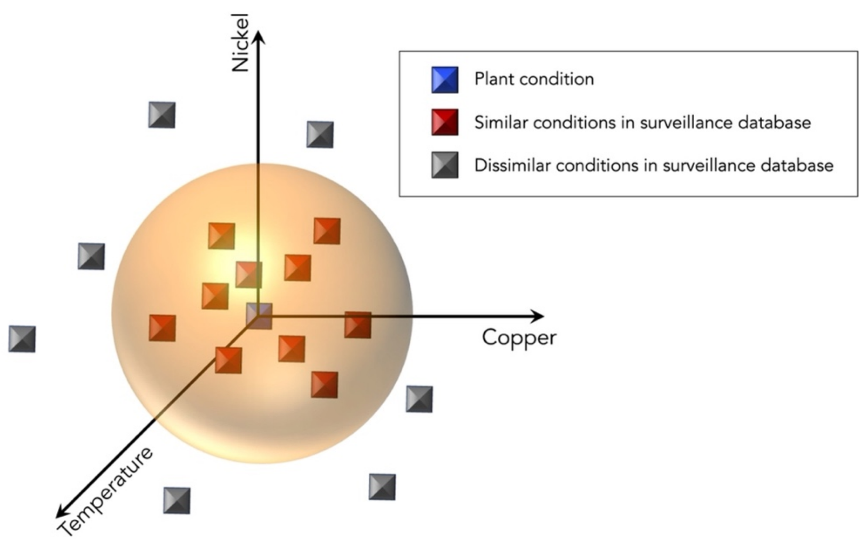

Figure 7. Illustration of kNN approach for defining similar data, based on information from [Kirk 2022].

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

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

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

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

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

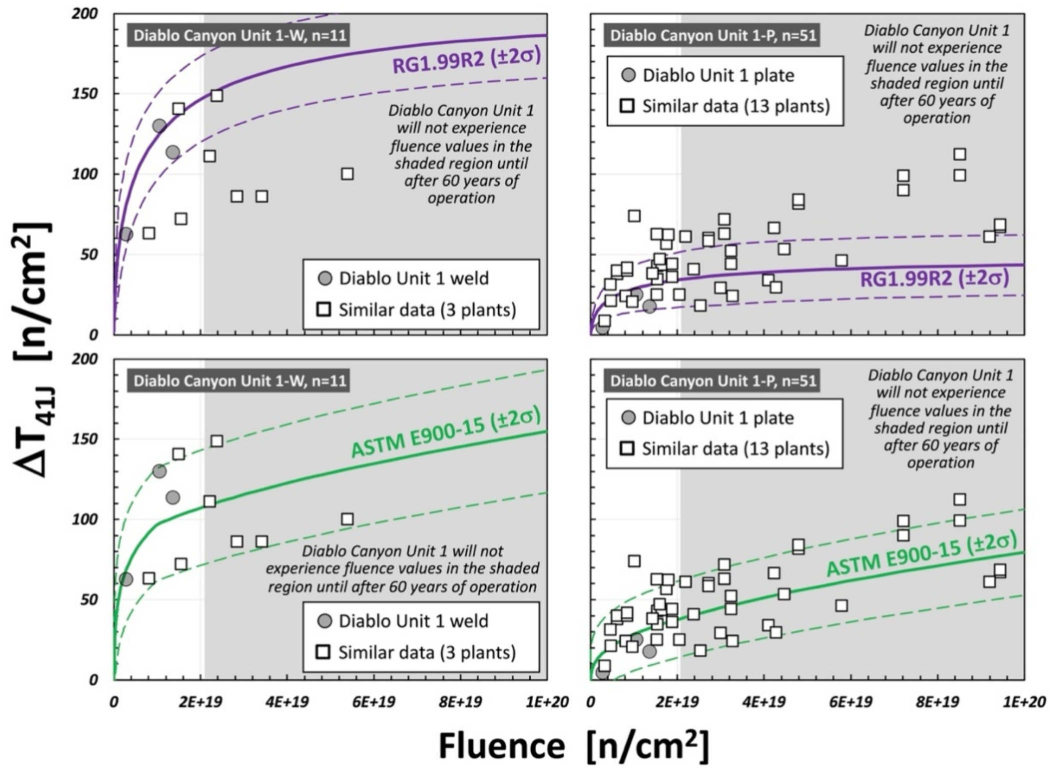

Figure 8. Data from the ASTM database of similar embrittlement sensitivity to the Diablo Canyon Unit 1 weld (left) and plate (right) compared to the predictions of the [RG1.99R2] ETC (top) and the [ASTM E900-15] ETC (bottom). For the weld Cu=0.196 and Ni=1.0 while for the plate Cu=0.081 and Ni=0.481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

4.1.2. Analysis following Draft ASME Code Case N-914

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

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

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

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

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

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

- RT_{INPUT} is the unirradiated value of RT_{To}. RT_{To} is estimated from the unirradiated value of T_0 from Table 7 plus a value of 19.4 °C. Thus, RT_{INPUT} = -73.1 °C.
- Shift is the value of the ASTM E900-15 ETC (see Appendix) increased by a value η = 14.7 °C. η is the value needed to make the predicted shift values pass through the mean of the measured ΔT_{41J} data from Table 5 and the ΔT_0 datum from Table 7. CC N-914 guidance says that for welds ΔT_{41J} and ΔT_0 values are equivalent.
- Margin is a value that is 0.684 times the standard deviation of the ASTM E900-15 ETC (see Appendix). The value of 0.684 represents an uncertainty reduction allowable by the Code Case if there are three or more ΔT_{41J} and/or ΔT_0 values. This is similar to the margin reduction allowed by [RG1.99R2] for credible data.
- Offset is a value of 10 °C that is added because the RT_{INPUT} value was based on T_0 testing of SE(B) specimens.

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

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

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

Fluence ÷ 1× 10^{19} n/cm^2	# specimens tested	# specimens meeting E1921 criteria	T_0 reported in [MRP-127] (°C)	T_0 from Consultant's re-calculation (°C)	Unirradiated RT_{T0} from Consultant's re-calculation [°C]	ΔT_0 from Consultant's re-calculation (°C)
0	9	9	-93.0	-92.5	-73.1	--
1.61	11	11	-43.0	-42.0	--	+134.5

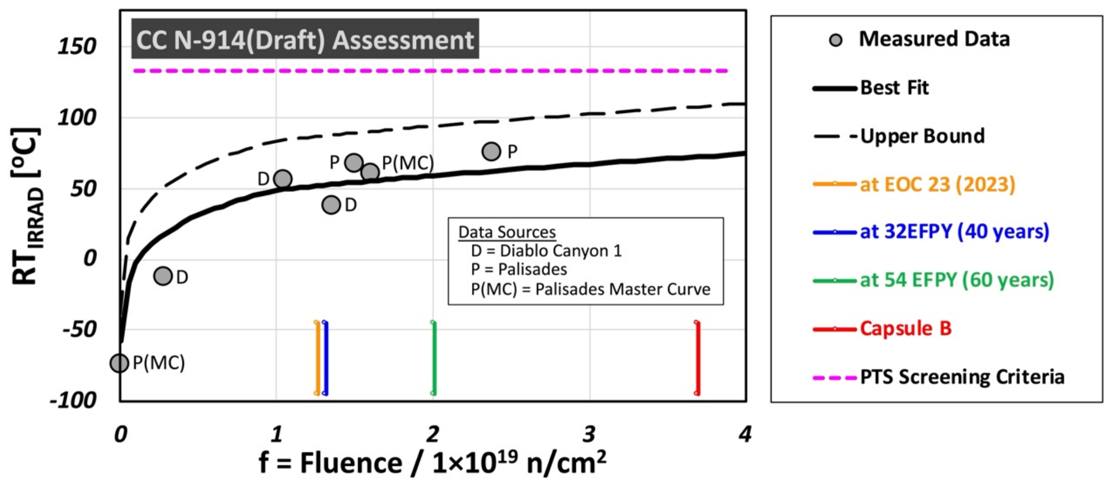

Figure 9. Estimate of the increase fracture toughness transition temperature with fluence for weld wire heat 27207 using ASME CC N-914 [MRP-462 Revision].

4.2. Analysis of Upper Shelf Energy (USE) Data

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

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

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

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

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

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

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

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

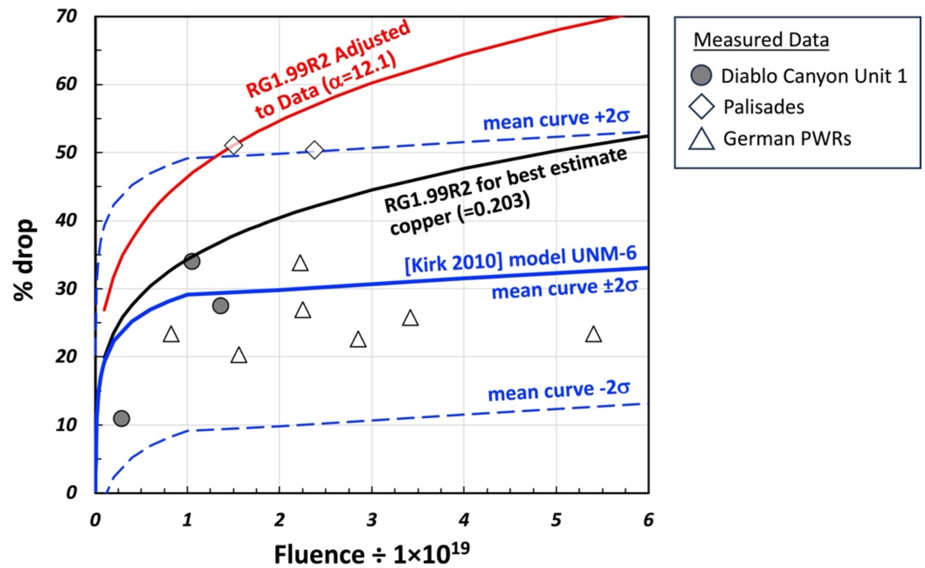

Figure 10. Estimates of percent USE drop determined from data having similar chemical composition and exposure temperature to the Diablo Canyon Unit 1 weld. The blue curves [Kirk 2010] was estimated for Cu=0.203, Ni=1.018, Mn=1.347, P=0.013, coolant temperature = 281.1 °C and illustrated the uncertainty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

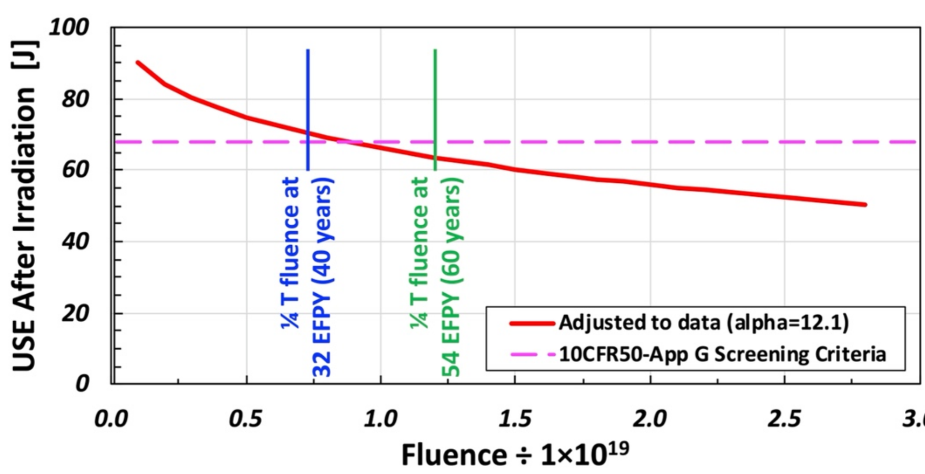

Figure 11. Supplemental USE analysis considering both the Diablo Canyon Unit 1 and Palisades data for weld wire heat 27204.

4.3. Summary and Evaluation of Need for Additional Data

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

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

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

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

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

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

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

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

5. Conclusions

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

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

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

The information contained herein supports the following conclusions:

Concerning Surveillance Testing

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

Concerning Safety Evaluations Performed to NRC Requirements

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

Concerning Safety Evaluations Performed Using Supplemental Techniques

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

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

<u>Concerning Additional Testing</u>

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

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

6. Citations

10CFR54	Requirements for Renewal of Operating Licenses for Nuclear Power Plants, https://www.nrc.gov/reading-rm/doc-collections/cfr/part054/index.html
10CFR50-AppG	Fracture Toughness Requirements, https://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-appg.html.
10CFR50.61	Fracture toughness requirements for protection against pressurized thermal shock events, https://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-0061.html.
ANP-3646	"Low Upper-Shelf Toughness Fracture Mechanics Analysis of Turkey Point Units 3 and 4 Reactor Vessels for Levels A & B Service Loads at 80-Years," AREVA Report ANP-3646NP Revision 0, NRC ADAMS ML18037A837.
ASTM E185-70	ASTM E185-82, Standard Practice for Design of Surveillance Programs for Light-Water Moderated Nuclear Power Reactor Vessels, ASTM, 1970.
ASTM E185-82	ASTM E185-82, Standard Practice for Design of Surveillance Programs for Light-Water Moderated Nuclear Power Reactor Vessels, ASTM, 1982.
ASTM E900-15	E900-15, Standard Guide for Predicting Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials, ASTM International, West Conshohocken, Pennsylvania, USA (revised 2015)
ASTM Adjunct	Adjunct for E900-15 Technical Basis for the Equation Used to Predict Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials, https://www.astm.org/adje090015-ea.html.
ASTM E1253-21	Standard Guide for Reconstitution of Charpy-Sized Specimens, ASTM E1253-21, https://www.astm.org/e1253-21.html.

ASTM E1921	Standard Test Method for Determination of Reference Temperature, T_0, for Ferritic Steels in the Transition Range, https://www.astm.org/e1921-22a.html.
BAW-2398	DeVan, M.J., "Test Results of Capsule SA-240-1 Consumers Energy Palisades Nuclear Plant," Babcock and Wilcox, Report BAW-2398, May 2001
BAW-2341R2	DeVan, M.J., "Test Results of Capsule SA-60-1 Consumers Energy Palisades Nuclear Plant," Babcock and Wilcox, Report BAW-2341 Revision 2, May 2001.
CEN-622	"Generic Upper Shelf Values for Linde 1092, 124, and 0091 Reactor Vessel Welds," prepared for the C-E Owners Group, June 1995, Legacy ADAMS 9508020111, ADAMS ML17228B226.
CPUC 2018	Decision Approving Retirement of Diablo Canyon Nuclear Power Plant, Application 16-8-006,https://docs.cpuc.ca.gov/PublishedDocs/Published/G000/M205/K423/205423920.PDF.
C-Senate 2022	California Senate Bill No. 846, Diablo Canyon powerplant: extension of operations (Sept. 2, 2022) https://leginfo.legislature.ca.gov/faces/billTextClient.xhtml?bill_id=202120220SB846, (SB 846).
DiabloLRA 2009	License Renewal Application, Diablo Canyon Power Plant Unit 1 and Unit 2, Facility Operating License Numbers DPR-80 and DPR-82, 23 September 2009, NRC ADAMS ML093340116.
DCL-08-021	PG&E Letter to NRC, "Revision to the Unit 1 Reactor Vessel Material Surveillance Program Withdrawal Schedule," 12 March 2008, NRC ADAMS No. ML080850564.
DCL-10-141	PG&E Letter to NRC, "Revision to the Unit 1 Reactor Vessel Material Surveillance Program Withdrawal Schedule," 25 October 2010, NRC ADAMS No. ML102990079.
DCL-11-122	PG&E Letter to NRC, "Revision to the Unit 1 Reactor Vessel Material Surveillance Program Withdrawal Schedule," 21 November 2011, NRC ADAMS No. ML113260072.
DCL-15-121	PG&E Letter to NRC, "Response to NRC Letter Dated September 24, 2015, Request for Additional Information for the Review of the Diablo Canyon Power Plant, Units 1 and 2, License Renewal Application – Set 38," 21 October 2015, NRC ADAMS No. ML15294A437.

DCL-18-015	PG&E Letter to NRC, "Request to Withdraw the Diablo Canyon Power Plant License Renewal Application," 7 March 2018, NRC ADAMS No. ML18066A937.
DCL-22-085	PG&E Letter to NRC, "Request to Resume Review of the Diablo Canyon Power Plant License Renewal Application or, Alternatively, for an Exemption from 10 CFR 2.109(b), Concerning a Timely Renewal Application," 31 October 2022, NRC ADAMS No. ML22304A691.
DCL-23-038	PG&E Letter to NRC, Diablo Canyon Units 1 and 2 License Renewal Application, NRC ADAMS No. ML23311A154.
EPRI 1993	Gamble, R.M., Kennedy, E.L., Server, W.L., Beaudoin, B.F., Conner, L., Venkatakrishnan, C.S., and Carter, R.G., "Reactor Vessel Embrittlement Management Handbook; A Handbook for Managing Reactor Vessel Embrittlement and Vessel Integrity," Electric Power Research Institute, TR-101975-T2, December 1993.
Hein 2010	H. Hein, A. Gundermann, E. Keim, H. Schnabel, and J. Ganswind, "Experimental Data Base for Assessment of Irradiation Induced Aging Effects in Pre-Irradiated RPV Materials of German PWR," Fontevraud 7 Conference, 26-30 September 2010, Avignon, France, Paper No. A015-T01.
Kirk 2010	M. Kirk, "A New Relationship for Upper Shelf Energy Drop," Presentation to the IAEA Specialists' Meeting on Irradiation Embrittlement, 20 October 2010, Znojmo, Czech Republic, https://www.nrc.gov/docs/ML1028/ML102850094.pdf.
Kirk 2014	M. Kirk, M. Erickson, W. Server, G. Stevens, and R. Cipolla, "Assessment of Fracture Toughness Models for Ferritic Steels used in Section XI of the ASME Code Relative to Current Data-Based Models," 2014 ASME Pressure Vessels and Piping Division Conference July 20-24, 2014, Anaheim, California, Paper PVP2014-28540.
Kirk 2022	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, Volume 568, 2022, https://doi.org/10.1016/j.jnucmat.2022.153886.
Kirk 2024	M.T. Kirk, "An Evaluation of the Status of Diablo Canyon Unit 1 With Respect to RPV Condition Monitoring and Prediction, Part 1: Addressing Public Concerns," Report to the Diablo Canyon Independent Safety Committee, 26 January 2024.
May 2010	J. May, H. Hein, J. Ganswind, and M. Widera, "Irradiation Behavior of PWR RPV Steels Under Operating Conditions," Fontevraud 7

Conference, 26-30 September 2010, Avignon, France, Paper No. A055-T01.

MRP-127 — *Materials Reliability Program: Fracture Toughness Evaluation of Unirradiated and Irradiated RPV Weld Metals (MRP-127),* EPRI, Palo Alto, CA: 2004. 1011112.

MRP-326 — *Materials Reliability Program: Coordinated PWR Reactor Vessel Surveillance Program (CRVSP) Guidelines (MRP-326).*EPRI, Palo Alto, CA: 2011. 1022871.

MRP-418 — *Technical Basis for ASME Code Case N-830-1, Revision 1 (MRP-418, Revision 1): Direct Use of Master Curve Fracture Toughness for Pressure-Retaining Materials of Class 1 Vessels, Section XI.* EPRI, Palo Alto, CA: 2019. 3002016008, https://www.epri.com/research/products/000000003002016008.

MRP-450 — *Materials Reliability Program: Technical Basis and Implied Margins of ASME Code, Section XI, Appendix G (MRP-450) Fracture Toughness Criteria for Protection Against Failure.* EPRI, Palo Alto, CA: 2020. 3002018241, https://www.epri.com/research/products/000000003002023871

MRP-462 — *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, https://www.epri.com/research/products/000000003002020911.

MRP-462 Revision — Draft Revision of MRP-462, 8 May 2023, ASME Section XI WGOPC Working Document, https://drive.google.com/file/d/16Wsna03G_1I-xJjW1MKEnCL5p286JGzA/view?usp=sharing.

NRC Renewals 80 — https://www.nrc.gov/reactors/operating/licensing/renewal/subsequent-license-renewal.html

NRC 1994 — Safety Assessment of Report WCAP-13587, Revision 1, "Reactor Vessel Upper Shelf Energy Bounding Evaluation for Westinghouse Pressurized Water Reactors," 21 April 1994, NRC ADAMS ML090580069.

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

NRC 2010 — NRC Letter to Diablo Canyon concerning Approval of Proposed Reactor Vessel Material Surveillance Capsule Withdrawal Schedule, 29 October 2010, ADAMS ML103010159.

NRC 2014	NRC Regulatory Information Summary 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components," 14 October 2014, NRC ADAMS ML14149A165.
NRC 2021	The meeting was held virtually on 21 January 2021. The following ADAMS ML numbers include the agenda, meeting summary, and presentations: ML21019A073, ML21048A119, ML21019A017, ML21019A019, ML21047A142, ML21015A336, ML21015A315, ML21019A355, ML21019A344, ML21015A289.
NRC 2023a	"License Renewal Application; Acceptance for Docketing and Opportunity to Request a Hearing and to Petition for Leave to Intervene," December 14, 2023, NRC ADAMS Accession Number ML23341A003.
NRC 2023b	NRC Letter to Diablo Canyon Concerning Revision of the Reactor Vessel Material Surveillance Capsule Withdrawal Schedule, 20 July 2023.
NUREG-1801	NUREG-1801 Rev. 2, "Generic Aging Lessons Learned (GALL) Report," United States Nuclear Regulatory Commission, December 2010, https://www.nrc.gov/docs/ML1034/ML103490041.pdf.
RG1.99R2	Regulatory Guide 1.99 Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988, https://www.nrc.gov/docs/ML0037/ML003740284.pdf.
RG1.147	Regulatory Guide 1.147, Revision 20, Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1, December 2021, https://www.nrc.gov/docs/ML2118/ML21181A222.pdf.
RG1.161	Regulatory Guide 1.161, "Evaluation of Reactor Pressure Vessels, with Charpy Upper-Shelf Energy Less Than 50 ft-lb," June 1995, https://www.nrc.gov/docs/ML0037/ML003740038.pdf.
Randall 1986	Randall, P.N., "Basis for Revision 2 of the U.S. Nuclear Regulatory Commission's Regulatory Guide 1.99," Radiation Embrittlement of Nuclear Reactor Pressure Vessel Steels: An International Review (Second Volume), ASTM STP 909, L.E. Steele, Ed., American Society for Testing and Materials, Philadelphia, 1986, pp. 149-162.
Server 2014	Server, W. L., Hardin, T. C., Brian Hall, J., and Nanstad, R. K., "U.S. High Fluence Power Reactor Surveillance Data—Past and Future," ASME Journal of Pressure Vessel Technology, April 2014; 136(2): 021603. https://doi.org/10.1115/1.4026149.
SLOMFP-NRC 2023	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

SLOMFP-CPUC 2023 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.

Vagins 1982 Memorandum to S. Hanauer from M. Vagins, "Findings of the *ad hoc* working group on the selection of RT_{NDT} values," 30 August 1982, United States Nuclear Regulatory Commission, Legacy ADAMS Number 8509200138.

WCAP-8465 J.A. Davidson, J.H. Phillips, and S.E. Yanichko, "Pacific Gas and Electric Co. Diablo Canyon Unit No.1 Reactor Vessel Radiation Surveillance Program," Westinghouse Electric Company, Report WCAP-8465, January 1975.

WCAP-11567 S.E. Yanichko, S.L. Anderson, J.C. Schmertz, and L. Albertin," Analysis of Capsule S From the Pacific Gas and Electric Company Diablo Canyon Unit 1 Reactor Vessel Radiation Surveillance Program," Westinghouse Electric Company, Report WCAP-11567, December 1987.

WCAP-13440 E. Terek, "Supplemental Reactor Vessel Radiation Surveillance Program for the Pacific Gas and Electric Company Diablo Canyon Unit No. 1," Westinghouse Electric Company, Report WCAP-13440, December 1992.

WCAP-13587 S. Tandon, M. J. Malone, T. R. Mager "Reactor Vessel Upper Shelf Energy Bounding Evaluation for Westinghouse Pressurized Water Reactors," Westinghouse Electric Corporation, September 1993, WCAP-13587 Revision 1.

WCAP-13750 E. Terek, S.L. Anderson, and A. Madeyski, "Analysis of Capsule Y From the Pacific Gas and Electric Company Diablo Canyon Unit 1 Reactor Vessel Radiation Surveillance Program," Westinghouse Electric Company, Report WCAP-13750, July 1993.

WCAP-15958 E. Terek, S.L. Anderson, and A. Madeyski, "Analysis of Capsule V From the Pacific Gas and Electric Company Diablo Canyon Unit 1 Reactor Vessel Radiation Surveillance Program," Westinghouse Electric Company, Report WCAP-15958, January 2003.

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

WCAP-17651 E.J. Long, "Palisades Nuclear Power Plant Reactor Vessel Equivalent Margins Analysis," Westinghouse Electric Company, WCAP-17651-NP Revision 0, February 2013, NRC ADAMS ML13295A451.

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

7. Mark Kirk – Resume

Education

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

Employment

Dates	Company Name & Location	Concluding Position Title
2018-date	Central Research Institute of Electric Power Industry (CRIEPI) *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
- Kirk, M.T., and Dickson, T.L., "Recommended Screening Limits for Pressurized Thermal Shock (PTS)," NRC, NUREG-1874, March 2010, https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1874/index.html
- Kirk, M., "Development of the Alternate Pressurized Thermal Shock Rule (10 CFR 50.61a) in the United States," *Nuclear Engineering and Technology*, Vol 45, No 3, June 2013.
- "Adjunct for ASTM E900-15: Technical Basis for the Equation used to Predict Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials," ASTM Subcommittee E10.02 Work Package WK34875, ASTM International, September 2015, https://www.astm.org/adje090015-ea.html.
- Kirk, M., Hardies, R., and Gordon, M., "Historical Review of the Development of ASTM Standard Practices E185 and E2215 from the 1960s to 2015," International Review of Nuclear Reactor Pressure Vessel Surveillance Programs, ASTM STP1603, W. L. Server and M. Brumovsky, Eds., ASTM International, West Conshohocken, PA, 2018, pp. 9–53, http://dx.doi.org/10.1520/STP160320170006
- Kirk, M., "Summary of Work to Develop the Transition Temperature Shift Equation Used in ASTM Standard Guide E900-15," International Review of Nuclear Reactor Pressure Vessel Surveillance Programs, ASTM STP1603, W. L. Server and M. Brumovsky, Eds., ASTM International, West Conshohocken, PA, 2018, pp. 432–456, http://dx.doi.org/10.1520/STP160320170009

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

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

Appendix

Embrittlement Trend Curves

A Trend Curve for ΔT_{41J}

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

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

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

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

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

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

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

A Trend Curve for ΔT_{41J}

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

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

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

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

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