

Surveillance Capsule B from Diablo Canyon Unit 1

A Review of Data reported in WCAP-19173-NP and its Implications

Consultant's report to the Diablo Canyon Independent Safety Committee

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Disclaimer

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

Executive Summary

In 2022 the State of California decided to not terminate the operation of the Diablo Canyon Nuclear Power Plant at the conclusion of its original 40-year operating license in 2024 and 2025. Consequently, in October 2022 the Pacific Gas and Electric Company (PG&E) informed the Nuclear Regulatory Commission (NRC) that it wished to re-initiate their license renewal application [DCL-22-085]. In November 2023 PG&E submitted a license renewal application to the NRC to extend by 20-years the operating licenses of both Unit 1 and Unit 2. A renewed license was granted by the NRC in 2026. Following this decision to pursue 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.

To assist the DCISC in their deliberations on these matters they retained the services of this consultant in 2023. Since that time a number of reports and memorandums have been prepared for the DCISC. For example:

- [Kirk Part 1] is a report that describes embrittlement prediction and condition monitoring practices used in the USA. Further, the report responds directly to a number of public concerns raised before January of 2025.
- [Kirk Part 2] is a report that evaluates Unit 1 with respect to the NRC's PTS and USE screening criteria.
- [Kirk 2024] performs a further evaluation of the status of Unit 1 with respect to the USE screening criteria.

These and several other memoranda can be found on [DCISC Website]. The additional memoranda address questions posed by Mr. Severance and Professor Macdonald concerning Unit 1 and also provide an assessment of possible future changes to embrittlement regulations discussed by the NRC in [SECY-22-0019] and in [NRC 2026b] on Unit 1.

The current report provides an evaluation of data obtained by testing Capsule B from the Unit 1 surveillance program. In April 2025, surveillance Capsule B was removed from Unit 1 and sent to the Westinghouse hot cell laboratory in Pittsburgh Pennsylvania for testing. The results of this testing, which were reported in June 2026, included two new measurements of embrittlement made using Charpy V-notch specimens from weld wire heat 27204. Welds made from heat 27204 are the most sensitive to neutron irradiation embrittlement of the various steels from which the beltline of the reactor pressure vessel in Unit 1 was fabricated and, as such, most limit the operation of that Unit.

This report independently confirms the “credibility” analysis of the Charpy V-notch data set for weld wire heat 27204 reported by Westinghouse in Appendix D of [WCAP-19173]. This data set includes the two measurements from Capsule B not available before June 2026. The available Charpy data (ΔT_{41J} , ΔUSE) for heat 27204 were obtained over a range of neutron exposure spanning conditions characterizing the beginning of Unit 1 operation in the late 1980s to conditions that will not be reached until long after expiration of its current operating license in 2045. More specifically, the data from Capsule B was obtained at a fluence slight less than twice that expected in the Unit 1 RPV at the conclusion of the current 60-year license term.

This report used the new measurements on weld wire heat 27204 along with existing data on this weld wire heat to assess the status of Unit 1 with respect to the NRC’s screening criteria on pressurized thermal shock (PTS) as expressed in [10CFR50.61] and with respect to the NRC’s screening criteria on upper shelf energy (USE) as expressed in [10CFR50-AppG]. The results were as follows:

- **Pressurized Thermal Shock**

- NRC Method: An analysis performed using the NRC’s required analysis procedures as expressed in [10CFR50.61] determined that Unit 1 is forecast to remain 18.4 °C (33.2 °F) below the PTS screening criteria after 60-years of operation and, therefore, remains in compliance with those requirements through that time. The analysis also forecast that the embrittlement of weld wire heat 27204 will not reach the screening criteria until a neutron exposure nearly twice that expected in Unit 1 after 60-years of operation. These results are very similar to prior PTS assessments of weld wire heat 27204 that were based on five measurements, which have been available since 2011 and serve as the current licensing basis
- Supplemental Analysis: Additionally, a supplemental analysis was performed using a more up-to-date analysis method as described in draft ASME Code Case N-914; this method is currently under review by the American Society of Mechanical Engineers. Use of CC N-914 provides an independent check on the conclusions based on the NRC’s analysis method. The CC N-914 analysis used fracture toughness data on weld wire heat 27204 that is not used by the NRC. The CC N-914 analysis assessed Unit 1 to be 29.8 °C (53.7 °F) below the NRC’s PTS screening criteria, which confirms the conservatism of the NRC analysis.

- **Upper Shelf Energy**

- NRC Method: An analysis was performed using the new measurements of Charpy samples from weld wire heat 27204 irradiated in Capsule B and reported in [WCAP-19173]. Using the NRC’s required methodology, this analysis predicts that Unit 1 will have a USE after 60-years of 80J, which is 12J (8.9 ft-lb) above the NRC’s USE screening criteria and, thus, complies with this NRC requirement. The analysis also forecasts that the USE of weld wire heat 27204 will not reach the screening criteria until it reaches a neutron exposure over three times that expected in Unit 1 after 60-years of operation.

- Supplemental Analysis: The predictions of the NRC method were checked using a data set consisting of all measurements from weld wire heat 27204, including both data from the sister weld and surrogate weld. Together these data make a set that is 75% larger than that required by the NRC. This supplemental analysis predicts that Unit 1 will have a USE after 60-years of 74J, which is 6J (4.4 ft-lb) above the NRC's USE screening criteria and, thus, complies with NRC requirements. The analysis also forecasts that the USE of weld wire heat 27204 will not reach the screening criteria until it reaches a neutron exposure of approximately 1.8-times that expected in Unit 1 after 60-years of operation.

Finally, this report reviewed possible future changes to embrittlement and surveillance guidelines that have been recently discussed by the NRC to assess their potential impact on Unit 1. These include a possible change to the NRC's method of forecasting embrittlement, a possible change in surveillance requirements after the original 40-year license term, and the possible allowance by the NRC for plants to use fracture toughness data in their assessments. The impact of these changes on Unit 1, were they to occur, is as follows:

- There is no impact of the possible change in embrittlement formula because the NRC does not anticipate changing this formula except for fluence values much higher than Unit 1 will experience in its first 60-years of operation.
- The impact of changes to surveillance requirements on Unit 1 is more difficult to assess because the NRC has provided limited information about its future intentions. If future requirements consider the relationship of capsule fluence to reactor fluence then no more capsules should be required of Unit 1 because Capsule B was withdrawn and tested at a fluence over 2-times the predicted 60-year fluence. However, if the NRC's new capsule withdrawal schedule considers only the interval in years between surveillance capsule withdrawals then another capsule may be required before the end of the 60-year license. If this is the case Unit 1 has three capsules all containing the limiting weld wire heat that could be used to satisfy such a testing requirement.
- In what the NRC has written the use of fracture toughness data has been expressed as an option, not a requirement. The supplemental PTS analysis reported here used fracture toughness data for weld wire heat 27204 along with draft ASME CC N-914. This use of fracture toughness data demonstrated a greater distance between the assessed RT_{PTS} value for Unit 1 and the NRC's screening criteria.

Table of Contents

1.	BACKGROUND AND OBJECTIVE	7
2.	UNIT 1 SURVEILLANCE PROGRAM AND DATA.....	10
2.1.	SURVEILLANCE PROGRAM.....	10
2.2.	DATA SUMMARY	14
3.	DATA ANALYSIS METHODS.....	19
3.1.	CREDIBILITY EVALUATION	19
3.2.	PTS EVALUATION [10 CFR 50.61]	21
3.3.	USE EVALUATION.....	24
4.	CREDIBILITY ANALYSIS.....	26
4.1.	EVALUATION OF THE LIMITING WELD WIRE HEAT	26
4.2.	EVALUATION OF CORRELATION MONITOR MATERIALS	27
5.	PTS ANALYSIS.....	29
5.1.	EVALUATION OF CURRENTLY AVAILABLE DATA USING THE REQUIRED NRC METHOD.....	29
5.2.	EVALUATION OF PRIOR CAPSULE B DATA SETS USING THE REQUIRED NRC METHOD	30
5.3.	SUPPLEMENTAL EVALUATION USING DRAFT ASME CODE CASE N-914.....	32
5.4.	SUMMARY	34
6.	USE ANALYSIS	36
6.1.	EVALUATION USING THE REQUIRED NRC METHOD	36
6.2.	SUPPLEMENTAL EVALUATION USING AN EXPANDED DATA SET.....	37
6.3.	SUMMARY	37
7.	ASSESSMENT OF THESE RESULTS RELATIVE TO POSSIBLE REGULATORY CHANGES, AND OPTIONS FOR UNIT 1 40	
7.1.	IMPACT OF POSSIBLE REGULATORY CHANGES ON UNIT 1	40
7.2.	OPTIONS FOR UNIT 1	42
8.	SUMMARY	43
9.	CITATIONS.....	46

1. Background and Objective

In 2022 the State of California decided to not terminate the operation of the Diablo Canyon Nuclear Power Plant at the conclusion of its original 40-year operating license in 2024 and 2025 [C-Senate 2022], directing the California Public Utilities Commission *“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 the Pacific Gas and Electric Company (PG&E) informed the Nuclear Regulatory Commission (NRC) that it wished to re-initiate their license renewal application [DCL-22-085]. In November 2023 PG&E submitted a license renewal application to the NRC to extend by 20-years the operating licenses of both Unit 1 and Unit 2 [DCL-23-118]. In 2025 the NRC issued their safety evaluation report [NRC 2025a], and in 2026, the renewed licenses [NRC 2026a].

Following this decision to pursue 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. To assist the DCISC in their deliberations on these matters they retained the services of this consultant in 2023. Since that time a number of reports and memorandums were included in DCISC Annual Reports and posted on the [DCISC website] to document this effort.

The objective of the current report is to review new data obtained from the testing of surveillance Capsule B, and to assess its implications on the operation of Unit 1. To satisfy the requirements for surveillance monitoring expressed in the license for Unit 1¹, Capsule B was

¹ Chapter 2 of [Kirk Part 2] provides a detailed history of the Unit 1 surveillance program, the requirements on which it is based, and how the withdrawal date for Capsule B evolved. This information demonstrates that Capsule B was not part of the surveillance requirements for the first 40-

removed at the end of Fuel Cycle 25 in April 2025 and shipped to the Westinghouse hot cell laboratory in Pittsburgh Pennsylvania for testing. The results of those tests were reported to the NRC in June 2026 [WCAP-19173]. Capsule B provided two new data on limiting² weld wire heat 27204. These data help to inform assessment of the embrittlement status of Unit 1 with respect to the NRC’s requirements concerning pressurized thermal shock (PTS) [10CFR50.61] and upper shelf energy (USE) [10CFR50-AppH]. Chapter 2 of [Kirk Part 1] provided a detailed description of these requirements and the role of data obtained by surveillance capsule testing in assessing compliance with them. A synopsis of this information also appears later in this report, the remainder of which is structured as follows:

- Chapter 2 summarizes the surveillance program for Unit 1, including tables and graphs showing the new information obtained by testing materials contained in Capsule B.
- Chapter 3 provides an overview of how data from the testing of materials contained in Capsule B are used to assess compliance with NRC rules to demonstrate the unit’s continued operating safety with respect to RPV embrittlement.
- Chapter 4 assesses the “credibility” of the Unit 1 dataset expressed in [RG1.99R2] and in [10 CFR50.61].
- Chapter 5 assesses Diablo Canyon Unit 1 with respect to the NRC’s pressurized thermal shock (PTS) screening criteria as expressed in [10CFR50.61].
- Chapter 6 assesses Diablo Canyon Unit 1 with respect to the NRC’s upper shelf energy (USE) screening criteria as expressed in [10CFR50-AppG].
- Chapter 7 assesses these results with respect to possible changes to RPV regulations being considered by the NRC [SECY-22-0019, NRC 2026b] and discusses options for Unit 1
- Chapter 8 summarizes the information provided by this report.

In Chapters 5 and 6 the status of Unit 1 will be assessed with respect to the NRC’s PTS and USE screening criteria using two different methods, as follows:

- First the methods required by the NRC is used; these having been previously described in [Kirk Part 1, Kirk Part 2] and summarized in Chapter 3 of this report. These calculations are the same as those required of licensees to demonstrate the compliance of with these regulations.
- Second, supplemental analyses are performed as a check on the outcome of the NRC analysis methods.
 - **For PTS**, draft³ ASME Code Case (CC) N-914 defines the supplemental analysis method, which will be described in greater detail in Section 5.3 [Kirk 2026, MRP-

years of operation. Rather, testing of Capsule B is a requirement during the period of license renewal from 40 to 60 years, which was applied for in 2024 [SCL-23-118] and approved by the NRC in 2026 [NRC 2026a].

² In this context the word “limiting” means the constituent of the RPV beltline that most limits the safe operability of the unit.

³ As detailed in [MRP-462], CC N-914 has been under development since 2019. It is not yet a part of the ASME Code and is, as of the date of this publication, under ballot consideration within ASME

462]. This up-to-date method benefits from the nearly 40-years of operational experience, embrittlement data, and understanding of embrittlement damage mechanisms that have occurred since the NRC's required method was developed [Randall 1986, RG1.99-R2]. While the author of this report has played a key role in the development of CC N-914 and of the embrittlement trend curve used by that CC [ASTM E900], the consensus processes of both ASME and ASTM ensures that ASME and ASTM procedures cannot be the work of any one individual but, rather, reflect the consensus opinion of a diverse technical community concerning best practices.

- **For USE** the supplemental analysis adopts the NRC's method and also uses the "sister" and "surrogate" data from Unit 1's limiting weld wire heat that is not considered by the NRC. This approach was taken to evaluate all available data for the limiting weld wire heat in Unit 1.

While similar in form to the NRC's required procedures, these supplemental analysis methods stand apart from current NRC requirements and exclusive NRC influence, providing an independent check on the NRC's methods. In particular, the supplemental PTS analysis using CC N-914 consolidates the considerable experience gained on this topic in the last four decades. That method is likely to be adopted by the ASME Code in the near future.

2. Unit 1 Surveillance Program and Data

2.1. Surveillance Program

Chapter 2 of the “Part 2” report, which was prepared for the DCISC in 2024, described the Unit 1 surveillance program in detail [Kirk Part 2]. This section summarizes that information for context and for ease of reference; [Kirk Part 2] should be consulted for full details.

Unit 1 has both an *original* surveillance program [WCAP-8465] and a *supplemental* surveillance program [WCAP-13440]. Original program capsules were installed in the reactor before it began operation in 1985 while supplemental program capsules were installed in the early 1990s after Fuel Cycle 5. Table 1 and Table 2 summarize the capsule designations and the capsule loadings for the original and supplemental programs, respectively.

As described in [WCAP-13440], the supplemental program provided additional capsules to accommodate operation beyond 40 years, provided additional standby capsules for unforeseen future contingencies, and provided additional capsules containing samples from weld wire heat 27204. The latter were needed because by fuel cycle 5 it was recognized that welds made from heat 27204 were the most embrittlement sensitive of the different steels in the Unit 1 beltline and, therefore, the steel most likely to limit future operations (lower shell longitudinal welds designated 3-442 A, B, and C were all fabricated using weld wire heat 27204). However, as detailed in Table 1 (see *italic red text*), only three capsules⁴ from the

⁴ Construction of Diablo Canyon Unit 1 began in 1968; the original surveillance program being designed to the requirements of [ASTM E185-70]. At that time copper content had not yet been identified a key factor in determining the embrittlement sensitivity of RPV steels. It is for this reason that only three of eight capsules from the original surveillance program contained samples of what would be later recognized as the heat of material, weld wire 27204, that would most limit the future operability of Unit 1.

original program (S, Y, and V) contained samples of heat 27204 (see citations [WCAP-11567], [WCAP-13750], and [WCAP-15958], respectively). The supplemental program increased this supply by another four capsules (A, B, C, and D), see *italic red text* in Table 2. Two of these capsules, B and D, contained samples of the surveillance weld from the original program that had already received irradiation in Capsule S⁵. All four of the supplemental capsules contained samples from a so-called “surrogate weld”⁶ also fabricated using weld wire heat 27204. The similarity of the copper and nickel content between the surveillance weld in the original program and the surrogate weld in the supplemental program make the surrogate weld suitable to provide information on the embrittlement trends of the limiting weld in the Unit 1 beltline.

Table 3 is based on information taken from Table 7-1 of [WCAP-19173]; it provides the current status of all capsules from both the original and supplemental programs. As indicated in Table 3:

- Four capsules have been tested. Per NRC requirements the remnants from these tested specimens have been stored at the Westinghouse hot cell facility for potential future use.
- Four have been removed from the reactor and stored at Unit 1 but have not been tested.
- Four remain in the reactor in a “standby” status, meaning that there are no current plans for removal or testing.

Of the stored and standby capsules, three (Capsules A, C, and D) contain samples of weld wire heat 27204 (see *red text* in Table 3). Under current NRC rules [10CFR50-AppH] no further capsule testing is required for Unit 1 during its 60-year renewed license term.

⁵ The broken halves of Charpy specimens can be fabricated into new specimens using a process called “reconstitution” as described by [ASTM E1253] which, in principle, can generate two new test specimens from every previously tested sample. The supplemental surveillance program used this technique, distributing the broken Charpy specimens from Capsule S heat 27204 between Capsule B and Capsule D.

⁶ The term “surrogate weld” is not commonly used. Here it designates a weld made from weld wire heat 27204 at a different time than the surveillance weld for Unit 1 was fabricated.

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

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	---
Intermediate Shell Plate B4106-2	8 CVN, 1 Tensile, 2 WOL	---
Intermediate Shell Plate B4106-3	8 CVN, 1 Tensile, 2 WOL	8 CVN, 2 Tensile, 2 WOL
<i>Weld Heat 27204, Linde 1092 flux</i>	---	<i>8 CVN, 2 Tensile, 2 WOL</i>
Heat Affected Zone	---	8 CVN
Correlation Monitor Material	8 CVN	8 CVN

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 (CMM), 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 (see Section 4.2 for more details).

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

Material	Capsule A	Capsule B	Capsule C	Capsule D
<i>Surrogate Weld from Nozzle Drop-Out. Weld heat (27204) and flux type (Linde 1092) same and flux lot (3714) as surveillance weld.</i>	<i>15 CVN, 3 Tensile</i>	<i>15 CVN, 3 Tensile</i>	<i>30 CVN, 3 Tensile</i>	<i>15 CVN, 3 Tensile</i>
Lower Shell Plate B4107-1	15 CVN, 3 Tensile	15 CVN, 3 Tensile	15 CVN, 3 Tensile	15 CVN, 3 Tensile
Correlation Monitor Material	12 CVN	8 CVN	---	---
<i>Surveillance Weld (from Capsule S), Heat 27204, Longitudinal Beltline Weld (Note 1)</i>	---	<i>2 WOL, 9 Charpy Inserts</i>	---	<i>8 Charpy Inserts</i>
Weld 8B (Linde 80) (Note 2)	---	8 CVN	9 CVN	12 CVN
Weld 9B (Linde 0091) (Note 2)	---	7 CVN	9 CVN	11 CVN
Weld W7 (Linde 80) (Note 2)	---	7 CVN	8 CVN	11 CVN
Weld 72W (Linde 0124) (Note 2)	---	8 CVN	9 CVN	12 CVN

- Notes:** (1) These samples were previously irradiated in Capsule S, so at insertion in Capsules B and D they already had a fluence of 2.93×10^{18} n/cm². After withdrawal from Capsules B and D these samples will be reconstituted following the procedures of [ASTM E1253]. Reconstitution is a practice by which the broken halves of tested Charpy specimens have end-tabs welded to them so that they can be tested again to produce a new measurement of absorbed energy. Reconstitution techniques were researched extensively in the 1990s [van Walle 1996, Onizawa 1998, NUREG/CR-6777] and standardized as [ASTM E1253]. When the [ASTM E1253] procedures are followed results from reconstituted samples are statistically indistinguishable from normal Charpy samples.
- (2) Weld 8B, 9B, W7, and 72W are not used in the Unit 1 RPV. These welds were encapsulated to support a program conducted by the Electric Power Research Institute (EPRI).

Specimen Types

CVN = Charpy V-notch specimen [ASTM E23]

Charpy Insert = half of broken Charpy (for reconstitution) [ASTM E1253]

Tensile = Tensile specimen [ASTM E08]

WOL = wedge-opening load specimen [Wessel 1968]

Table 3. Current status of Diablo Canyon Unit 1 surveillance capsules based on information taken from Table 7-1 of [WCAP-19173].

Status	Capsule	Insert Fuel Cycle	Removal Fuel Cycle	EFPY at Removal	Lead Factor	Fluence x 1E19 at Removal [n/cm ²]
Tested	<i>S</i>	<i>0</i>	<i>1</i>	<i>1.26</i>	<i>3.44</i>	<i>0.293</i>
	<i>Y</i>	<i>0</i>	<i>5</i>	<i>5.86</i>	<i>3.43</i>	<i>1.07</i>
	<i>V</i>	<i>0</i>	<i>11</i>	<i>14.23</i>	<i>2.20</i>	<i>1.35</i>
	<i>B</i>	<i>5</i>	<i>25</i>	<i>34.87</i>	<i>2.65</i>	<i>3.62</i>
Removed & Stored	T	0	5	5.86	3.43	1.07
	Z	0	5	5.86	3.43	1.07
	<i>C</i>	<i>5</i>	<i>12</i>	<i>15.84</i>	<i>1.84</i>	<i>1.24</i>
	<i>D</i>	<i>5</i>	<i>12</i>	<i>15.84</i>	<i>1.84</i>	<i>1.24</i>
Standby	<i>A</i>	<i>5</i>	<i>in reactor</i>	<i>tbd</i>	<i>1.04</i>	<i>tbd</i>
	U	0	in reactor	tbd	1.19	tbd
	X	0	in reactor	tbd	1.19	tbd
	W	0	in reactor	tbd	1.19	tbd

Notes: *Italic red text* indicates capsules containing samples of weld wire heat 27204.

Standby capsules remain in the reactor without a planned withdrawal date.

EFPY refers to the effective full power years of reactor operation

Capsule B contained samples of weld heat 27204 irradiated previously in Capsule S. The fluence on these samples is therefore $3.62+0.293 = 3.93 \times 10^{19}$ n/cm².

2.2. Data Summary

The Westinghouse report on Capsule B [WCAP-19173] provides the data collected by testing samples removed from Capsule B, including Charpy V-notch (CVN) testing, tensile testing, melt wire analysis, and dosimetry evaluations to estimate fluence. Of these, the CVN and fluence data have the greatest impact on PTS and USE assessments (see Chapters 5 and 6 respectively) and, so, are summarized here. Appendix D of [WCAP-19173] also contains an assessment of data credibility, which will be reviewed in Chapter 4.

By way of example, hyperbolic tangent (*tanh*) curves are fit through the Charpy data sets from each capsule to determine the transition temperature (T_{30} standing for temperature at 30 ft-lbs absorbed energy or, equivalently, T_{41J} standing for temperature at 41 Joules (J) absorbed energy) and upper shelf energy (USE) corresponding to the fluence value for each capsule. Figure 1 shows a schematic illustrating how T_{41J} and USE are determined from the *tanh* curves fit to the test data. [ASTM E2215] defines both the procedure for *tanh* fitting (see its Appendix X1) as well as the procedure for calculation of USE (see its Sections 3.1.5 and 3.1.6). [WCAP-19173] states that a program called “CVGRAPH” was used to perform the *tanh* fits. CVGRAPH has been used industry-wide for decades and is the basis for the procedure in Appendix X1 of [ASTM E2215].

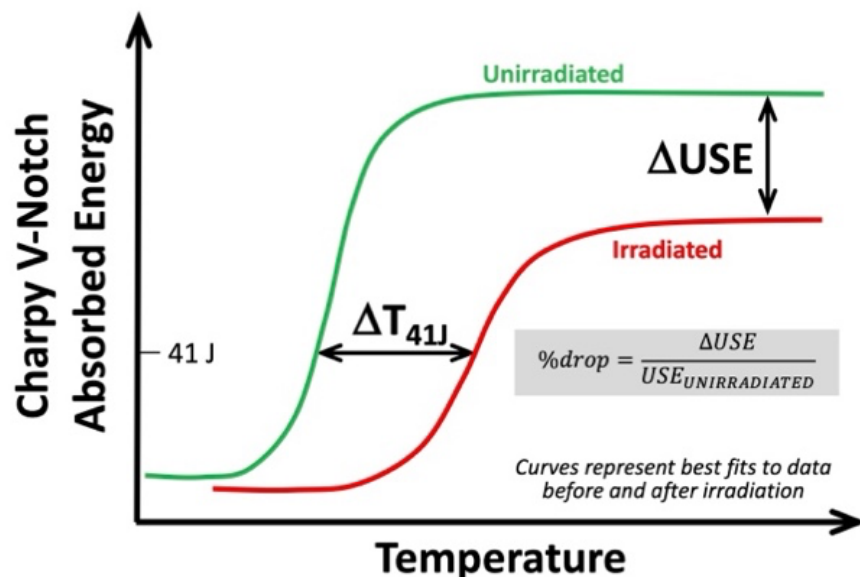


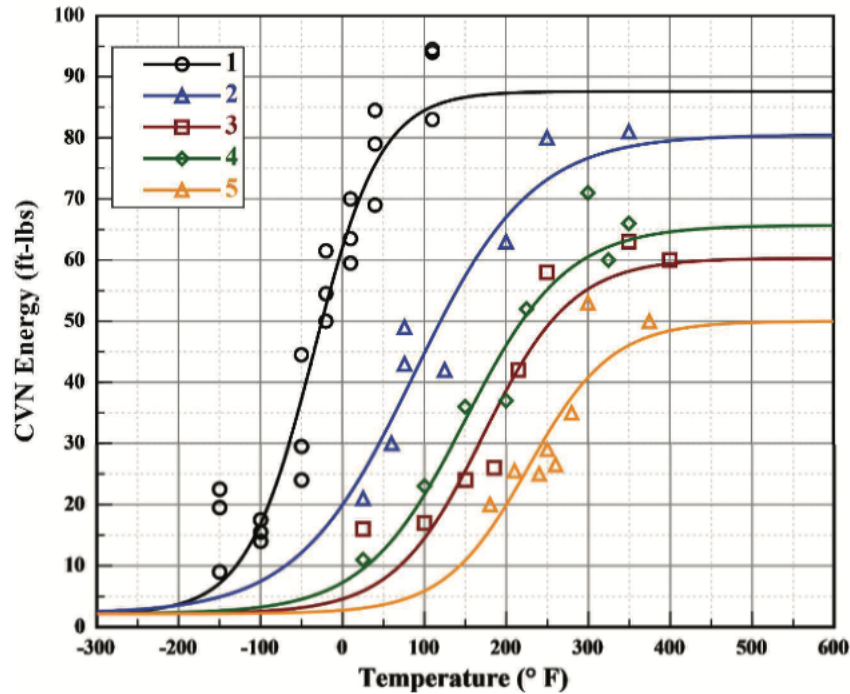
Figure 1. Hyperbolic tangent (*tanh*) curves, illustrating transition temperature (T_{41J}) and upper shelf energy (USE) metrics and how these change with neutron irradiation embrittlement.

The Charpy transition curves for the Diablo Canyon Unit 1 surveillance weld and the surrogate weld of weld wire heat 27204 appear in Figure 2 and Figure 3, respectively. These figures are direct copies from [WCAP-19173]. Each symbol on these plots represents the result of an individual CVN test. Much of the data on these figures was available prior to Capsule B testing; only the curve marked "5" on Figure 2 and the curve marked "2" on Figure 3 were determined from Capsule B testing

Weld Metal

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 6/9/2026 4:06 AM

Curve	Plant	Capsule	Material	Ori.	Heat #
1	Diablo Canyon 1	Unirrad	Weld	NA	27204
2	Diablo Canyon 1	S	Weld	NA	27204
3	Diablo Canyon 1	Y	Weld	NA	27204
4	Diablo Canyon 1	V	Weld	NA	27204
5	Diablo Canyon 1	B	Weld	NA	27204



Curve	Fluence	LSE	USE	d-USE	T @30	d-T @30	T @50	d-T @50
1	0.00E+000	2.2	87.6	0	-65	0	-25	0
2	2.93E+018	2.2	80.5	-7.10	45.1	110.1	120	145
3	1.07E+019	2.2	60.3	-27.3	167	232	255.6	280.6
4	1.35E+019	2.2	65.7	-21.9	135.2	200.2	219.3	244.3
5	3.91E+019	2.2	50	-37.6	243.7	308.7	0 ⁽¹⁾	25

Note:

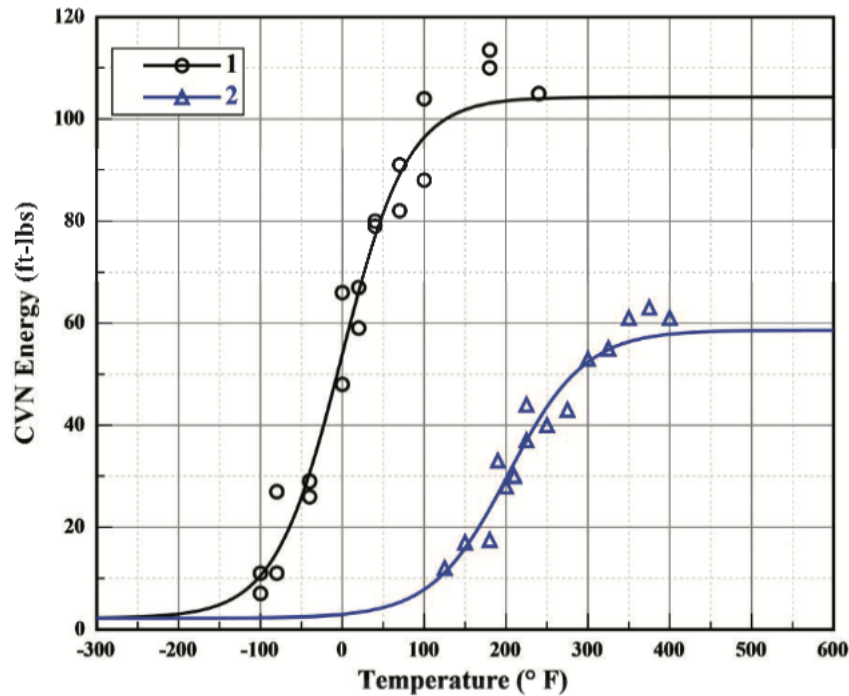
1. The fitted Charpy energy curve did not attain the 50 ft lb energy level within the evaluated temperature range; therefore, the irradiated 50 ft lb transition temperature was set to the highest temperature which achieved a measurement of 50 ft-lb.

Figure 2. CVN transition temperature curves for the Diablo Canyon Unit 1 surveillance weld (this is Figure 5-7 from [WCAP-19173]). Only data from the curve marked "5" (in orange) is new from Capsule B; all other data were previously reported.

Surrogate Weld (Heat # 27204)

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 6/7/2026 1:23 PM

Curve	Plant	Capsule	Material	Ori.	Heat #
1	Diablo Canyon 1	Unirrad	Weld	NA	27204
2	Diablo Canyon 1	B	Weld	NA	27204



Curve	Fluence	LSE	USE	d-USE	T @30	d-T @30	T @50	d-T @50
1	0.00E+000	2.2	104.3	0	-41.1	0	-6.2	0
2	3.62E+019	2.2	58.6	-45.7	201.5	242.6	282.7	288.9

Figure 3. CVN transition temperature curves for the Diablo Canyon Unit 1 surrogate weld (this is Figure 5-4 from [WCAP-19173]). Only data from the curve marked "2" (in blue) is new from Capsule B; all other data were previously reported.

Table 4 summarizes all of the values needed for the credibility analysis, the PTS analysis, and the USE analyses described later in this report. The fluence values reported in this table sometimes have small differences from values reported in prior Westinghouse reports as well as in [Kirk Part 2]. These differences reflect updates to available dosimetry information obtained by testing Capsule B and are used here to provide results that can be compared to those in [WCAP-19173].

Table 4. Summary of surveillance data for the Diablo Canyon Unit 1 limiting weld material, weld wire heat 27204 [WCAP-19173].

Plant	Capsule	Fluence $\div$ 1×10^{19} n/cm ²	Time Averaged Cold Leg Temperature (°C)	T _{41J} (°C)	ΔT_{41J} (°C)	USE (J)	% USE drop	Cu (wt%)	Ni (wt%)
Diablo 1 (surveillance weld)	--	unirrad	--	-53.9	--	118.8	--	0.198	0.999
	S	0.293	284.4	+7.3	61.2	109.1	8.1		
	Y	1.07	283.1	+75.0	128.9	81.8	31.2		
	V	1.35	282.8	+57.5	111.2	89.5	24.6		
	B+S	3.91 ⁽¹⁾	282.0	+117.6	171.5	67.8 ⁽²⁾	42.9		
Diablo 1 (surrogate weld)	--	unirrad	--	-40.6	--	141.4	--	0.22	1.0
	B	3.62	282.0	+94.2	134.8	79.5	43.8		
Palisades (sister ⁷)	--	unirrad	--	-40.7	--	147.0	--	0.194	1.067
	SA-60	1.50	279.4	+99.9	140.6	71.9	51.1		
	SA-240	2.38	281.9	+108.1	148.8	72.9	50.4		

General Notes: Mean cold leg temperature for Diablo Canyon Unit 1 is 283.1 °C.
Best estimate copper for weld 27204 is 0.2 weight percent
Best estimate nickel for weld 27204 is 1.019 weight percent.
Unirradiated RT_{NDT} used for Weld 27204 = -48.9 °C (generic value)

Numbered Note (1): Specimens placed in Capsule B received a fluence of 3.62×10^{19} n/cm². However, the surveillance weld samples in Capsule B, see the orange highlight above, were irradiated in Capsule S prior to their insertion in Capsule B. Thus, the fluence associated with these specimens is the sum of the Capsule S and Capsule B fluence values, which is 3.91×10^{19} n/cm².

Numbered Note (2): The measured USE value from the B+S surveillance sample is 68J, which is the same as the NRC's USE screening criteria as expressed in [10CFR50-AppG]. This value will be used as part of the USE analysis in Chapter 6. However, the measured value being equal to the screening criteria does not suggest that Unit 1 is expected to reach this value within its 60-year licensed lifetime, during which the fluence at the RPV inner diameter will only reach 2.14×10^{19} n/cm², or slightly more than half of the fluence to which the Caps B+S surveillance sample was subjected.

⁷ The surveillance program of the Palisades reactor in Michigan also contained samples of weld wire heat 27204. Such samples are commonly called "sister plant welds." By NRC requirement in [10 CFR 50.61], transition temperature (T_{41J}) data from sister plants must be included in the plant-specific evaluations of other plants. Extensive discussion of the applicability and appropriateness of using sister plant data appear in [Kirk Part 1].

3. Data Analysis Methods

Prior reports by this consultant that appear on the [DCISC Website] describe in detail the analysis procedures for (1) assessment of surveillance data credibility, (2) estimation of PTS reference temperature values (RT_{PTS}) for comparison to the NRC's PTS screening criteria of 132.2 °C (270 °F) for axial welds [10CFR50.61], and (3) estimation of the USE value after irradiation for comparison to the NRC's USE screening criteria of 68 J (50 ft-lbs) [10CFR50-AppG]. The following sub-sections provide short summaries of these procedures for context and for ease of reference; [DCISC Website] should be consulted for further details.

3.1. Credibility Evaluation

Conceptually the purpose of the credibility analysis is to determine how well plant-specific surveillance data are represented by the embrittlement trend curve (ETC) model used by the NRC, which appears in [RG1.99R2]. If enough of the plant-specific data lie sufficiently close to the [RG1.99R2] ETC model then the data set is called "credible." Conversely, if there are not enough data, or if the available data show trends sufficiently different to the [RG1.99R2] ETC model then the data set is called "not credible." These procedures are fully described in [Kirk Part 1] with reference to the source documents. Not credible data sets are treated more conservatively than credible data sets. This process ensures the ΔT_{41J} and % USE drop values are bounded (i.e., over-estimated) even when the embrittlement trends exhibited by the data diverge from those of the [RG1.99R2] models. These procedures are fully described in [Kirk Part 1] with reference to the source documents.

Sections 2.4.2.2 and 2.4.3.2 in [Kirk Part 1] describe in full detail the NRC's procedures for assessing the credibility of a surveillance data set⁸ for the analysis of transition temperature data and upper shelf data, respectively. Figure 4 and Figure 5, both of which appeared in [Kirk Part 1], show the details of these procedures in the form of a flowchart.

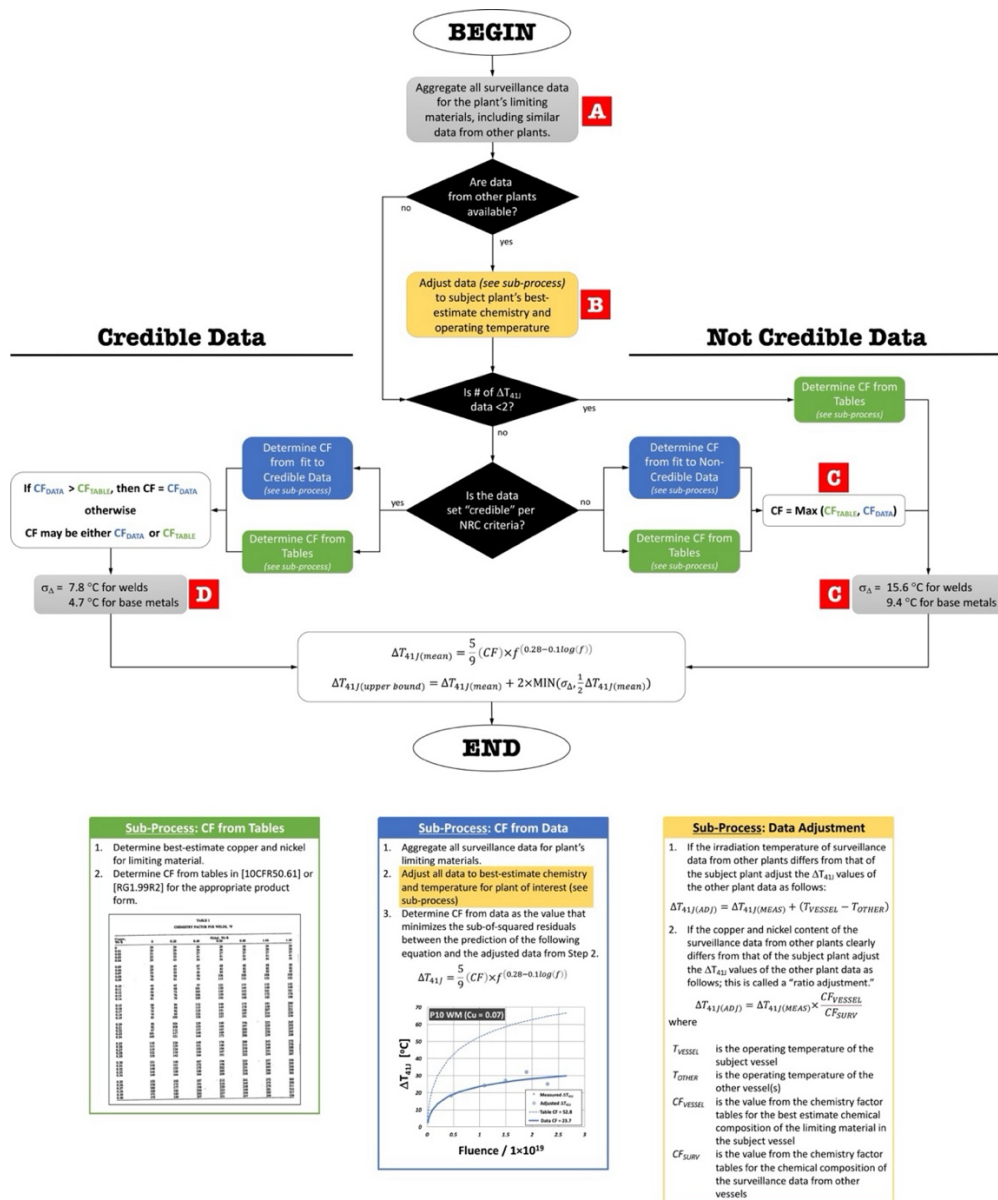


Figure 4. Diagram showing ΔT_{41J} forecasting procedures for both credible and not-credible data as required by [10CFR50.61] and as described by [RG1.99R2] and [Wichman 1998].

This figure appeared as Figure 10 in [Kirk Part 1].

⁸ A credibility evaluation is performed for a data set, not for individual datum. The evaluation determines if the data set as a whole is described well by the ETC model in [RG1.99R2].

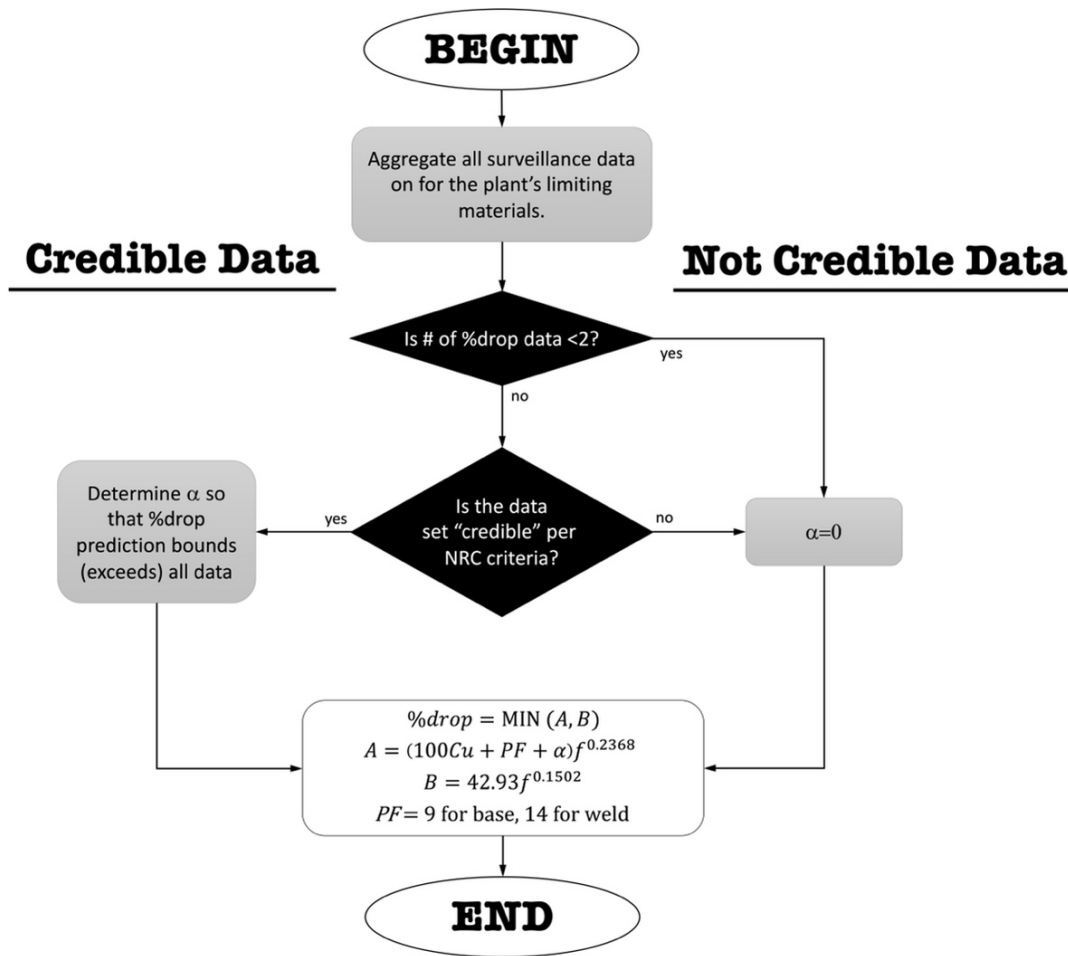


Figure 5. Diagram showing %drop forecasting procedures for both credible and not-credible data as described by [RG1.99R2]. *This figure appeared as Figure 12 in [Kirk Part 1].*

3.2. PTS Evaluation [10 CFR 50.61]

Section 2.5.1 in [Kirk Part 1] describes the NRC's PTS screening criteria from [10CFR50.61]. The [10CFR50.61] PTS screening criteria are 270 °F (132 °C) for axial welds, plates, and forgings and 300 °F (149 °C) for circumferential welds. The proximity of a plant to the screening criteria is determined as follows (see also Figure 6):

- Step (1) Obtain an unirradiated value of RT_{NDT} determined following ASME Code Section III Article NB-2331 for the limiting material. If a plant-specific RT_{NDT} value is not available, a generic value may be used along with an appropriate Margin term.
- Step (2) Assess surveillance data and credibility as described in Section 3.1 to establish the variation of an upper-bound estimate of ΔT_{41J} with fluence.
- Step (3) Add the results from Step (1) and Step (2) to establish the variation of an upper-bound estimate of RT_{NDT} with fluence. In this context the phrase "upper-

bound" is understood to mean a "mean plus 2-times standard deviation" value of RT_{NDT} .

- Step (4) At the fluence of interest (in the case of Unit 1 this is the 60-year fluence on the vessel ID, which is 2.14×10^{19} n/cm²) determine the value of the upper bound curve from Step (3), and compare this value to the NRC's PTS screening criterion. Using the upper-bound curve takes account of the various uncertainties in the analysis to ensure that the PTS evaluation is conservative (i.e., over-estimates the plant's RT_{PTS} value and under-estimates the operational duration it will for a plant to reach the NRC's PTS screening criterion).

[10CFR50.61] states that if the PTS screening criteria cannot be satisfied, then *"the licensee shall implement those flux reduction programs that are reasonably practicable to avoid exceeding the PTS screening criterion."* In practice most plants having embrittlement-sensitive steels in their beltline, including Unit 1, implemented flux reduction early in their operating lifetime to limit embrittlement and thereby preserve their options for future operability.

[10CFR50.61] also states that if flux reduction alone is insufficient, then other alternatives may be pursued to demonstrate the continued operating safety of a plant. These alternatives include operational modifications (specifics are not mentioned, however, as one example, plants in other countries have heated the emergency core cooling water), demonstration of compliance with the alternative PTS rule [10 CFR 50.61a], collection of additional data and use of alternative analytical techniques (for example, draft ASME CC N-914 could be used, as discussed in Section 5.3 of this report), use of a probabilistic fracture mechanics analysis, or thermal annealing, to provide increased confidence in the continued operating integrity of the vessel. In such a situation, [10CFR50.61] requires the plant to submit its plans to the NRC at least three years in advance of the date on which the plant is forecast to exceed the PTS screening criteria.

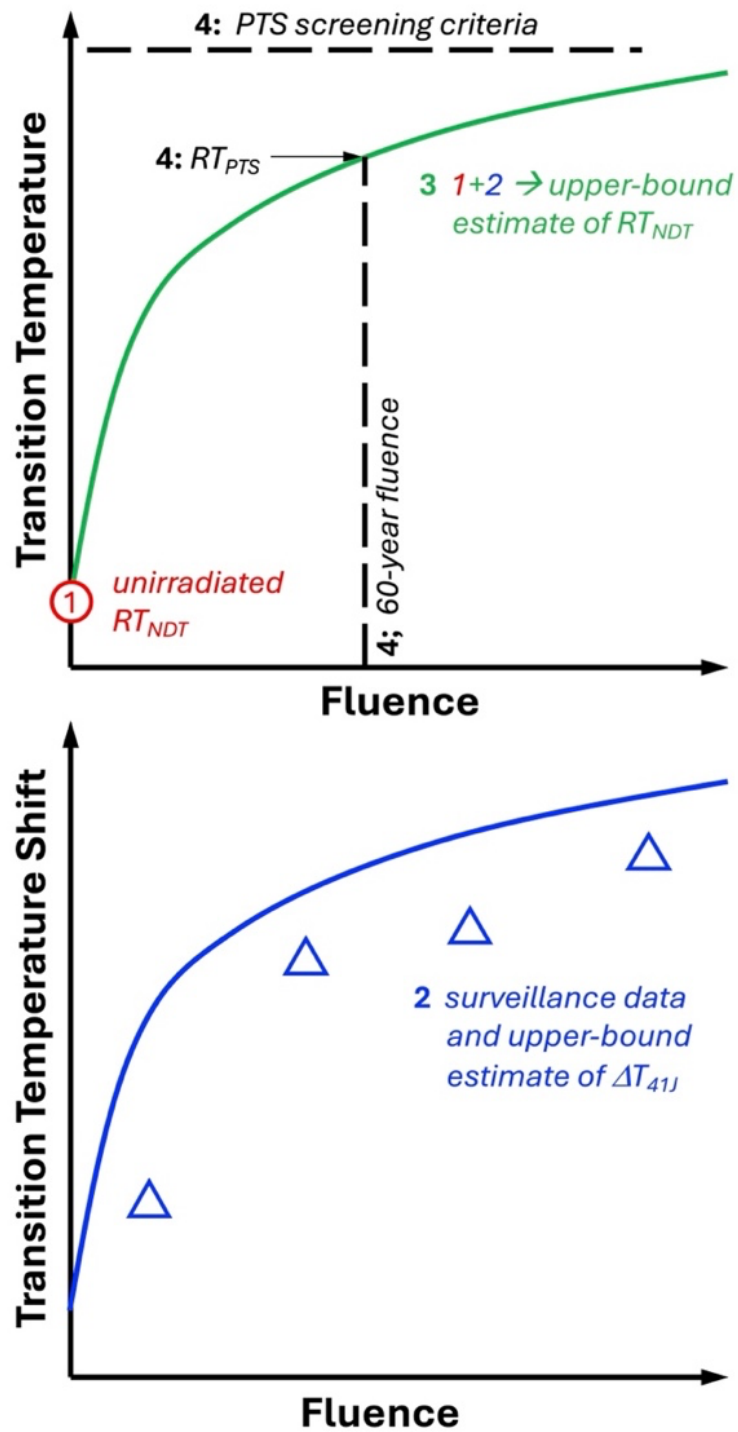


Figure 6. PTS evaluation schematic.

3.3. USE Evaluation

Section 2.5.3 in [Kirk Part 1] describes the NRC's USE screening criteria⁹ from [10CFR50-AppG], which is that the energy absorbed in breaking a Charpy V-notch specimen must be at least 68J (50 ft-lbs) throughout the licensed life of the reactor.

The proximity of a plant to the screening criteria is determined as follows (see also Figure 7):

- Step (1) Obtain an unirradiated estimate of USE by CVN testing according to ASTM Practice E185, which specifies that the USE is the numeric average of all specimens tested that exhibit 95% or more shear fracture area [ASTM E185].
- Step (2) Assess surveillance data and credibility as described in Section 3.1 to estimate of how % drop changes with fluence.

Note: As explained in Section 2.4.3 of [Kirk Part 1], unlike the analysis for ΔT_{41J} , there is no statistical treatment of uncertainties associated with the %drop or ΔUSE estimates obtained according to the approach described by [RG1.99R2], which continues to be accepted by the NRC. Contemporary NRC evaluation of the circa-1980s [RG1.99R2] procedure for USE demonstrates that these procedures bound roughly 80% of available data [Widrevitz 2019]. The NRC continues to accept the screening procedure described in [RG1.99R2] at least in part because evaluations of plants that fail the 68J screening have, without exception, been determined to be acceptable on the basis of more detailed and accurate assessments performed using elastic-plastic fracture mechanics as described in [RG1.161].

- Step (3) Use the results from Step (1) and Step (2) to establish the variation of an assessment curve that gives the variation of USE with fluence.
- Step (4) At the fluence of interest (in the case of Unit 1 this is the 60-year fluence at the quarter-thickness ($\frac{1}{4}T$) location, which is 1.27×10^{19} n/cm²) determine the value of the assessment curve from Step (3), and compare this value to the NRC's USE screening criteria.

⁹ The NRC does not refer to 50 ft-lbs (68J) as a "screening criteria." Instead [10 CFR 50-App G] states that *"Reactor vessel beltline materials must have Charpy upper-shelf energy in the transverse direction for base material and along the weld for weld material according to the ASME Code, of no less than 75 ft-lb (102 J) initially and must maintain Charpy upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb (68 J), unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of Charpy upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code."* Nevertheless, the label "screening criteria" is appropriate because 50 ft-lbs (68J) performs the same role in a USE analysis as does the PTS screening criteria value of 270 °F (132 °F). Specifically, the status of a particular plant material can be assessed relative to these screening criteria by performing a relatively simple algebraic calculation using plant data to develop a bounding estimate of the RPV's embrittlement at some fluence of interest. In the event either criterion is transgressed a more detailed analysis, or compensatory actions, or both are required to demonstrate plant safety.

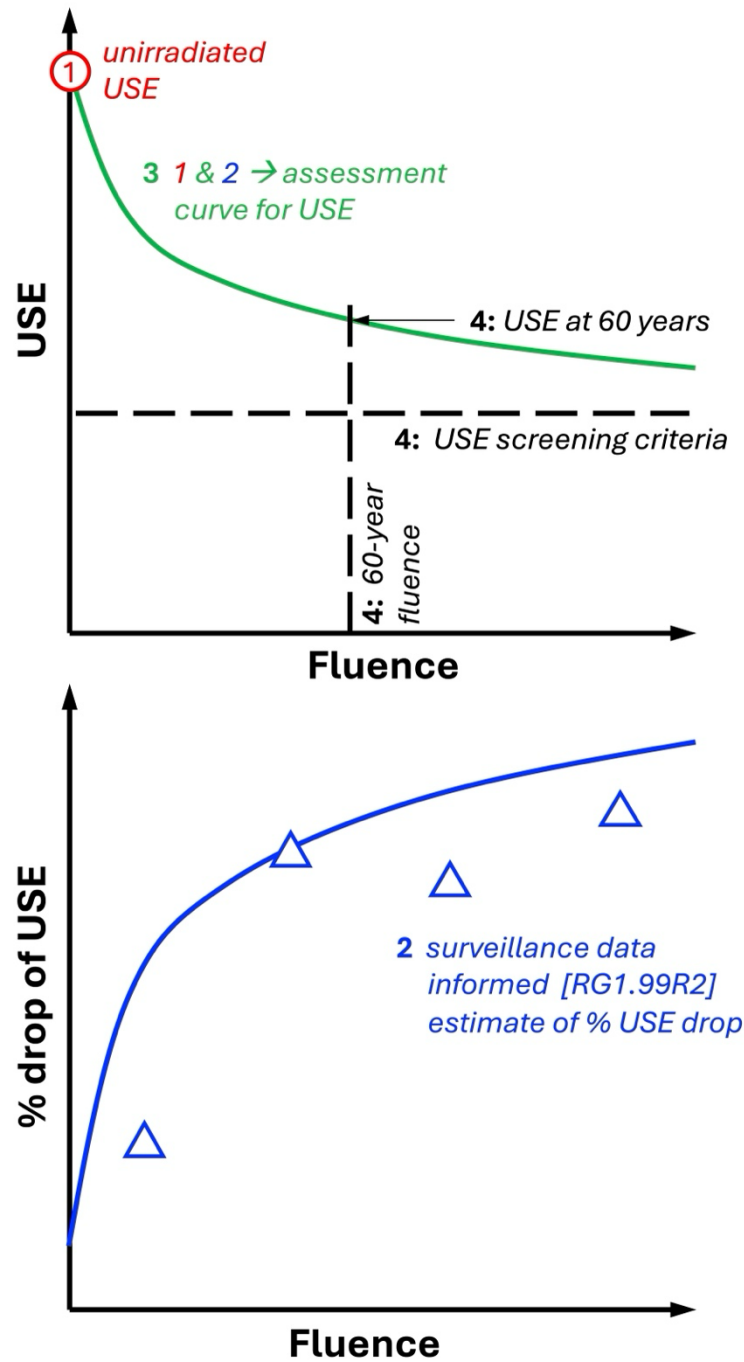


Figure 7. USE evaluation schematic.

4. Credibility Analysis

Sections 2.4.2.2 and 2.4.3.2 in [Kirk Part 1] describe the NRC's requirements for data "credibility" when analyzing surveillance data; that report should be consulted for full details. In this Chapter the recently obtained data from Capsule B is added to the data previously available from capsule testing and assessed for credibility. Section 4.1 evaluates data from the limiting weld wire heat while Section 4.2 evaluates data from the correlation monitor material.

4.1. Evaluation of the Limiting Weld Wire Heat

Using the data reported in Table 4 along with the NRC procedures for credibility evaluation outlined in [RG1.99R2] and in [Wichman 1998] produced a best-fit chemistry factor (CF) for the [RG1.99R2] ETC of 114.7 °C (206.5 °F). Using this value data credibility was assessed as summarized in Table 5. Out of seven available measurements, five are within the $\pm 1\sigma$ range of ± 15.6 °C for welds (yellow highlighted cells in the table indicate the two measurements outside of the $\pm 1\sigma$ range). This dataset meets the NRC's long-accepted¹⁰ requirement that at least 68% of available data fall within the $\pm 1\sigma$ range, so this data set is assessed to be credible.

No particular concern should be attached to the observation from Table 5 that the two data falling outside of the NRC's $\pm 1\sigma$ range are both from Capsule B, one lying above and one below the [RG1.99R2] curve. Such an occurrence is an expected result for a set of only seven observations; random chance alone produces such a result.

These results compare well to those reported in Appendix D of [WCAP-19173]. In Table D-6 of that document, a best fit CF value of 207.3 °F (115.2 °C) is reported, which differs from the

¹⁰ The 68% value is nowhere explicitly stated but has been long accepted. Section 2.4.2.2 in [Kirk Part 1] contains a more detailed explanation of the basis for this practice, see especially the discussion concerning Criteria (C) and Figure 9 in that report.

calculation performed here by 0.5 °C, which is less than a 0.5% difference. The reason for this small difference could be a slightly different calculation practice, or differences in rounding off numbers. These small differences do not change the credibility assessment. [WCAP-19173] also finds that a data set including capsule B data is credible, with the same five out of seven measurements falling within the $\pm 1\sigma$ range for welds.

In September 2026 the NRC issued its assessment [NRC 2026c] of the Capsule B report [WCAP-19173]. Consistent with the results reported here the NRC “noted that the available surveillance data for weld Heat # 27204 is credible in accordance with Regulatory Guide 1.99.”

Table 5. Credibility analysis of the Diablo Canyon Unit 1 incorporating the Capsule B data as reported in [WCAP-19173]

Plant	Capsule	Measured ΔT_{41J} [°C]	Consultant's Result (CF = 114.7 °C)			Westinghouse Result (CF = 115.2 °C)	Absolute Scatter Difference [°C]
			Adjusted ΔT_{41J} [°C]	[RG1.99R2] Predicted ΔT_{41J} [°C]	Absolute Scatter (Adj-Pred) [°C]	[WCAP-19173] Absolute Scatter [°C] (Table D7)	
Diablo 1	S	61.2	63.3	76.2	12.8	12.2	0.7
	Y	128.9	130.5	116.8	13.7	14.0	-0.3
	V	111.2	112.4	124.2	11.8	11.6	0.3
	B (surv weld)	171.5	172.6	154.9	17.6	17.6	0.0
	B (surrogate)	134.8	129.7	153.0	23.3	22.9	0.4
Palisades	SA-60	140.6	135.7	127.5	8.2	8.6	-0.4
	SA-240	148.8	145.9	141.5	4.4	2.8	1.6

Notes: Adjusted ΔT_{41J} values were modified from the measured ΔT_{41J} values following the NRC's procedures to account for small differences in irradiation temperature and in chemical composition.

Cells shaded in yellow have scatter exceeding the 1σ value for welds (15.6 °C).

4.2. Evaluation of Correlation Monitor Materials

As described in Section 2.4.2.2 in [Kirk Part 1], an assessment of the ΔT_{41J} values measured by testing correlation monitor materials (CMM) is part of the NRC's credibility evaluation. Both [10CFR50.61] and [RG1.99R2] express the same requirement, that:

“the surveillance data for the correlation monitor material in the capsule, if present, must fall within the scatter band of the data base for the material.”

No more explicit definition of this requirement has ever been provided by the NRC.

A CMM is a set of Charpy specimens taken from a reference block of steel rather than from the reactor vessel material itself. The response to radiation of CMMs (e.g., their ΔT_{41J} values and how ΔT_{41J} evolves with fluence) is well characterized [NUREG/CR-4947, IAEA-1290] by testing conducted after irradiation in many different commercial reactors. By comparing the value of ΔT_{41J} measured from CMM testing in a particular RPV with the larger database of ΔT_{41J} values it should be possible, in principle, to use these results as a check on the accuracy of the surveillance capsule dosimetry. Here the operative phrase is “in principle.” As reported in

[NUREG/CR-4947] and in [Wallin 1999], the variability in the mechanical properties of CMMs has made them less than ideal for their intended purpose. Nevertheless, the NRC's credibility requirement for CMMs remains in place in both [10CFR50.61] and [RG1.99R2]

Today the scatter shown by available CMM data relative to [RG199-R2] predictions is best quantified using the database of surveillance measurements compiled in [ASTM Adjunct] and updated in 2022. Figure 8 shows these data as colorful dots and compares them to the results of the four CMM measurements for Diablo Canyon Unit 1, which are shown as black triangles. Figure 8 shows that the CMM data from Unit 1 fall within the scatter band of CMM test results from many plants as compiled by ASTM. Thus, these data fulfill this NRC requirement. This conclusion is consistent with that reported by Westinghouse in the Capsule B report [WCAP-19173].

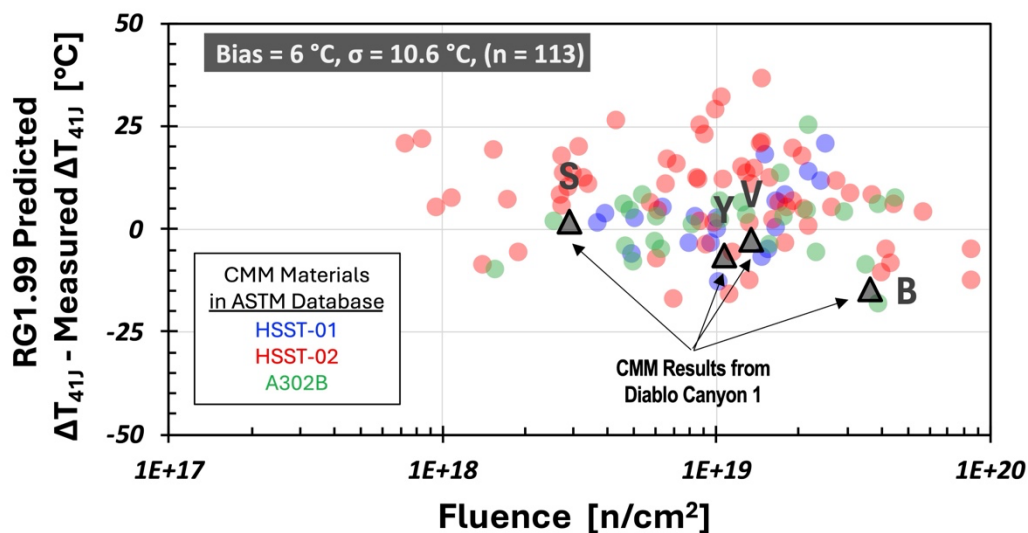


Figure 8. Comparison of CMM data assembled in [ASTM Adjunct] (colorful dots) with CMM data from Diablo Canyon Unit 1 capsules S, Y, V, and B. The vertical axis shows the difference between the [RG1.99R2] prediction of ΔT_{41J} and the measured value while the horizontal axis gives the fluence of each value.

5. PTS Analysis

5.1. Evaluation of Currently Available Data using the Required NRC Method

Figure 9 illustrates the results of a PTS assessment following the NRC's required procedure using all Charpy and RT_{NDT} data for weld wire heat 27204. This analysis includes the two new data obtained from Capsule B as reported in [WCAP-19173]. The available data cover a range of fluence to almost twice the value expected at 60-years, which meets the guidance of [ASTM E185-82], incorporated by reference into [10CFR50-AppH], that the final capsule test be tested as a fluence between 1- and 2-times the expected fluence at end of license.

Key results from this analysis are as follows:

- Best fit chemistry factor: 114.7 °C (206.5 °F)
- RT_{PTS} at 60 calendar years: 113.8 °C (236.8 °F)
- Comparison to PTS screening criteria of 132.2 °C: 18.4 °C below (33.2 °F below)
- Fluence when PTS screening criteria is reached: 4.2×10^{19} n/cm²

The estimated peak RPV fluence at the end of the 60-year license is 2.14×10^{19} n/cm² as reported in [WCAP-19173]. Thus, this analysis demonstrates that Unit 1 is not expected to exceed the PTS screening criteria until a time well beyond its current licensed term. Using a fluence to years conversion factor of

$$\frac{2.14 \times 10^{19} \text{ n/cm}^2}{60 \text{ years}} = 0.036 \times 10^{19} \frac{\text{n/cm}^2}{\text{year}}$$

indicates that it would take Unit 1 approximately¹¹ 118 calendar years of operation to reach this fluence. This expression of “years to reach the screening criteria” is intended to express the outcome of the analysis using units that are more accessible to the general public (years of operation) than the values of fluence that would be used in a report written for subject-matter experts.

These predictions are very similar to the prior PTS assessment, which was based on the five measurements available before Capsule B was tested [WCAP-17315]. That assessment was performed and reported in 2011 and was later confirmed by this consultant for the DCISC [Kirk Part 2].

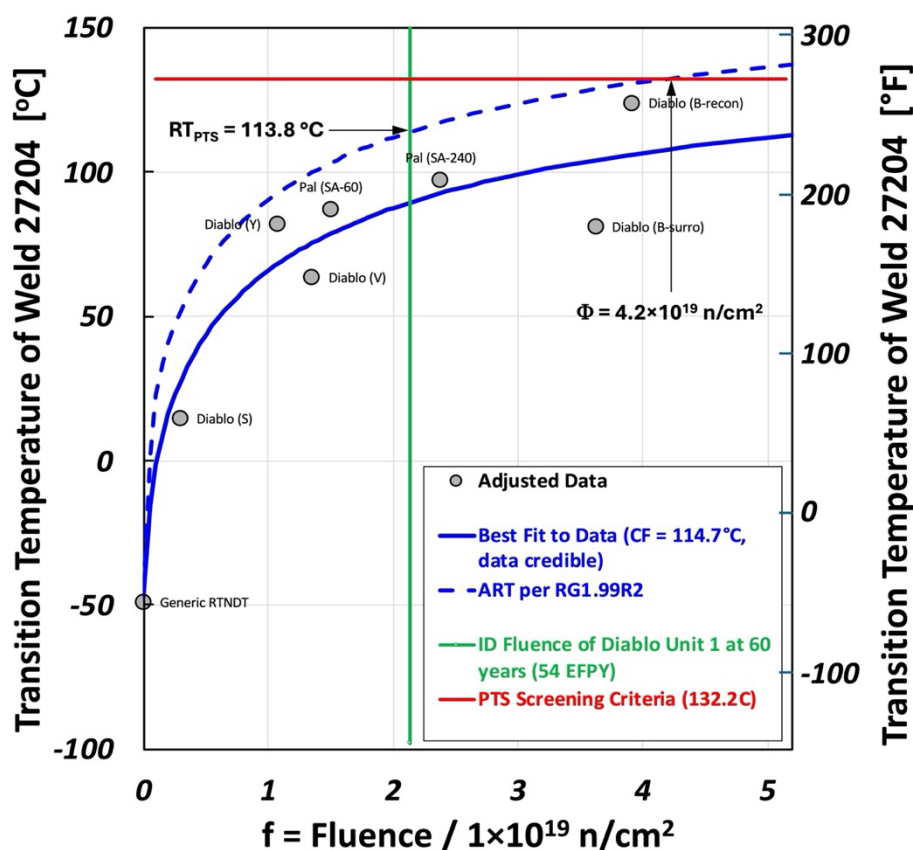


Figure 9. PTS assessment for Diablo Canyon Unit 1 including data from Capsule B.

5.2. Evaluation of Prior Capsule B Data Sets using the Required NRC Method

To provide additional insight concerning the effects of the data considered as part of the RT_{PTS} estimate for Unit 1, calculations like those performed in the preceding section were performed

¹¹ A linear conversion between fluence and operating years is approximate because it does not reflect changes in operating efficiency over time.

for all prior data states that have existed since the beginning of plant operation. These calculations were reported in [Kirk Part 2] and are summarized here in Figure 10. The horizontal axis on Figure 10 is calendar year. The numbers **in red** along the top of the figure give the number of irradiated T_{41J} and USE measurements that were available for weld wire heat 27204 and how this number has increased since Unit 1 was placed into operation in 1985. As stated previously these data have become available through surveillance capsule testing for Unit 1 and also through testing at the Palisades plant, which provided the sister-plant data. The **orange** and **blue** lines show the values of RT_{PTS} estimated from these data sets. **Orange lines denote RT_{PTS} estimates from not-credible data sets** while **blue lines denote RT_{PTS} estimates from credible data sets**. The **black dashed line** at 132 °C indicates the NRC's PTS screening criteria applicable to Unit 1.

- When the data set is evaluated as being "**credible**," (as it is today and has been since 2011), the estimated RT_{PTS} at 60 years is, on average, ≈ 17 °C below the PTS screening criteria. Consequently, the estimated number of years it would take to reach the NRC's PTS screening criteria exceeds considerably (by decades) the current license term of 60 years. The stability of this forecast occurs because, as illustrated in Figure 9, the measured ΔT_{41J} data follow a consistent trend that agrees reasonably well with the functional form of the ETC in [RG1.99R2].
- When the data set is evaluated as being "**not-credible**," which was the case before 1993 when there were not yet sufficient data to perform a credibility evaluation and again between 2003-2011 when three ΔT_{41J} measurements were available, the estimated RT_{PTS} at 60 years is 6 °C above the PTS screening criteria. In this situation it would take approximately 50-years of operation to reach the PTS screening criteria.

As illustrated in Figure 10, the NRC's credibility criteria influences significantly the outcome of the PTS assessment for Unit 1. As stated earlier, the NRC's criteria is based on the number of data falling outside of a $\pm 1\sigma$ range from the mean prediction of [RG1.99R2]; if more than 32% of available data fall outside of this range the data set is deemed to be non-credible and the estimate of RT_{PTS} is intentionally biased in a conservative direction (i.e., increased) in that case.

For a normal distribution of residuals about a mean prediction curve, the NRC's criteria will, by definition, be violated by 32% of randomly drawn data sets. This cautiously high alert rate for adding conservatism to the PTS assessment was established by the NRC between 1986 and 1998. It reflects the state of knowledge at that time, which was informed by a much smaller quantity of available surveillance data¹² and a less well-developed understanding of the mechanisms responsible for neutron irradiation embrittlement. Therefore, to provide a further check on the conservatism of these RT_{PTS} predictions obtained using the NRC's method, the next section analyzes the Unit 1 data using a more up-to-date analytical method for embrittlement prediction that is now under development by Section XI of the ASME Code.

¹² The [RG1.99R2] ETC was established in [Randall 1986] using a data set of fewer than 200 surveillance data measurements. Today over 2,000 measurements are available [Kirk 2023, ASTM Adjunct].

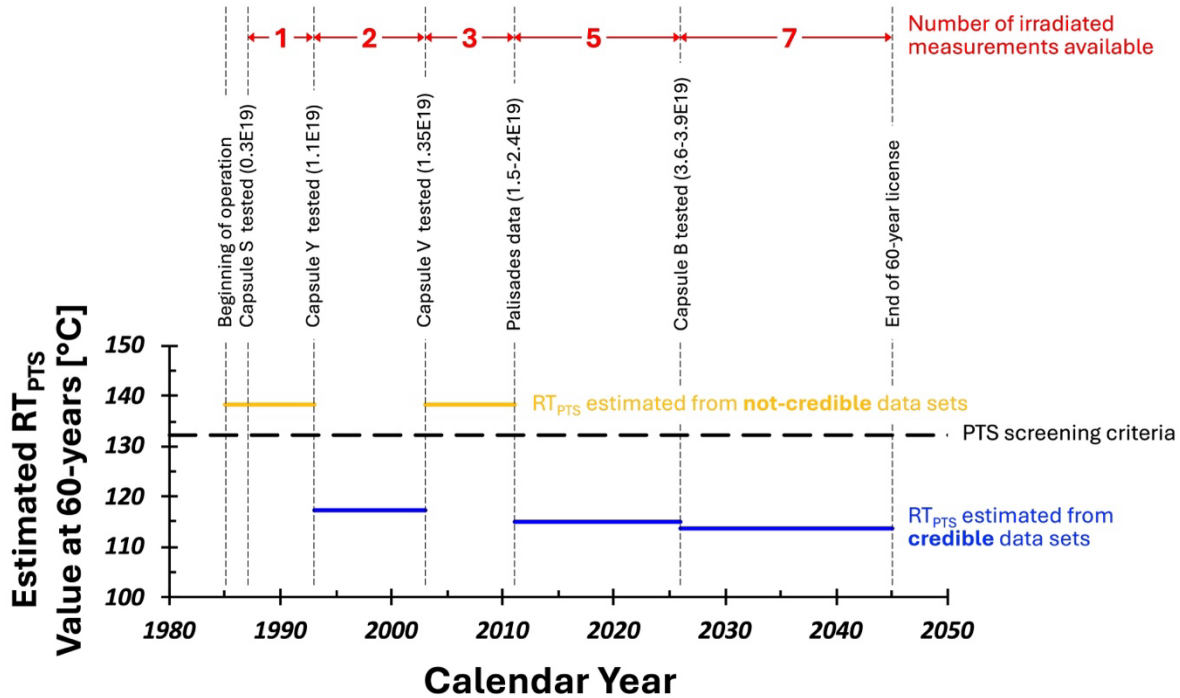


Figure 10. Evolution of data for weld wire heat 27204 over time and resultant estimates of RT_{PTS} for Unit 1.

5.3. Supplemental Evaluation using Draft ASME Code Case N-914

To provide check on the conservatism of the NRC assessment methods, this section includes an analysis of the Unit 1 data using a more up-to-date method for embrittlement prediction now under development by Section XI of the ASME Code. This effort, undertaken by the ASME Section XI Working Group on Operating Plant Criteria (WGOPC), aims to provide a method for estimation of embrittlement for use in Code calculations [Kirk 2026, MRP-462]. The method, designated as Code Case (CC) N-914, has been under development since 2019. Since that time, the CC has undergone multiple revisions to improve clarity and comprehensiveness and has been peer-checked by WGOPC members to ensure that multiple engineers following the CC instructions produce the same results. The CC is now being balloted at ASME; final passage is expected by 2027.

Relative to the NRC's method, CC N-914 offers the following advantages:

- It will be a consensus approach based on the diverse input and experience of ASME Code members.
- It will use a modern embrittlement trend curve [ASTM E900, Kirk 2023] that has been developed and checked against an up-to-date database of surveillance information having over 2,000 experimental observations (this compares with less than 200 experimental observations for [RG1.99R2]). The ASTM database spans the full range of conditions characteristic of today's operating fleet of nuclear power plants, for example:

- Fluence from 1×10^{17} to 2×10^{20} n/cm².
- Irradiation temperature from 255 to 300 °C.
- Copper content from nil to 0.4 weight percent.
- Nickel content from nil to 1.7 weight percent.
- The adjustment allowed for plant-specific data in terms of trend fitting and margin reduction is based on an extensive statistical analysis of surveillance data.
- A credibility assessment of the NRC-type is not performed. Rather, the plant-specific surveillance data is taken as the best exemplar of the embrittlement response of the RPV caused by exposure to neutron radiation.
- A means is provided to use Charpy V-notch data along with direct measurements of fracture toughness transition temperature (T_0) made using [ASTM E1921]. Such measurements accurately reflect the fracture toughness of the steel without the correlation uncertainties associated with impact tests using the Charpy V-notch specimen.

While still under development, the CC N-914 approach represents a more up-to-date methodology than does the NRC's, which was last revised in the late 1990s using a database of 177 ΔT_{41J} measurements over the fluence range from $\sim 7 \times 10^{17}$ to $\sim 8 \times 10^{19}$ [Randall 1986]. Use of the CC N-914 approach for weld wire heat 27204 data provides a means to check the outcome of the NRC's method.

For the CC N-914 evaluation, the Charpy data for weld wire heat 27204 summarized in Table 4 are again used¹³ and are augmented by fracture toughness transition temperature (T_0) data reported for this weld wire heat in [MRP-127] (see Table 6). Figure 11 shows the result of this analysis, which can be summarized as follows:

- RT_{PTS} at 60 operating years: 102.4 °C (216.3 °F)
- Comparison to PTS screening criteria of 132.2 °C: 29.8 °C below (53.7 °F below)
- Fluence when PTS screening criteria is reached: 6.5×10^{19} n/cm²
- Years to reach the PTS screening criteria: approximately 183 (see footnote 14)

The Margin in the CC-N914 varies somewhat with fluence. On Figure 11 the Margin appears graphically as the vertical distance between the dashed and solid blue curves.

¹³ All data from Table 4 are used *with the exception* of the generic unirradiated RT_{NDT} value. The CC approach recognizes that plant and weld wire heat specific values of T_{41J} and T_0 , both of which are available in the unirradiated condition, are more reliable measures of the fracture toughness of the weld in Unit 1 than is a generic average based on data from other plants.

¹⁴ The approximate fluence to years conversion factor of 0.0355 n/cm² / year is again used.

Table 6. Information on T_0 values for weld wire heat 27204 tested in Palisades capsule SA-60 [MRP-127] that were determined by testing single edge-notch bend specimens following [ASTM E1921].

Fluence $\div$ 1×10^{19} n/cm^2	# specimens tested	# specimens meeting E1921 validity criteria	T_0 reported in [MRP-127] ($^{\circ}\text{C}$)	T_0 from Consultant's re- calculation ($^{\circ}\text{C}$)
0	9	9	-93.0	-92.5
1.61	11	11	-43.0	-42.0

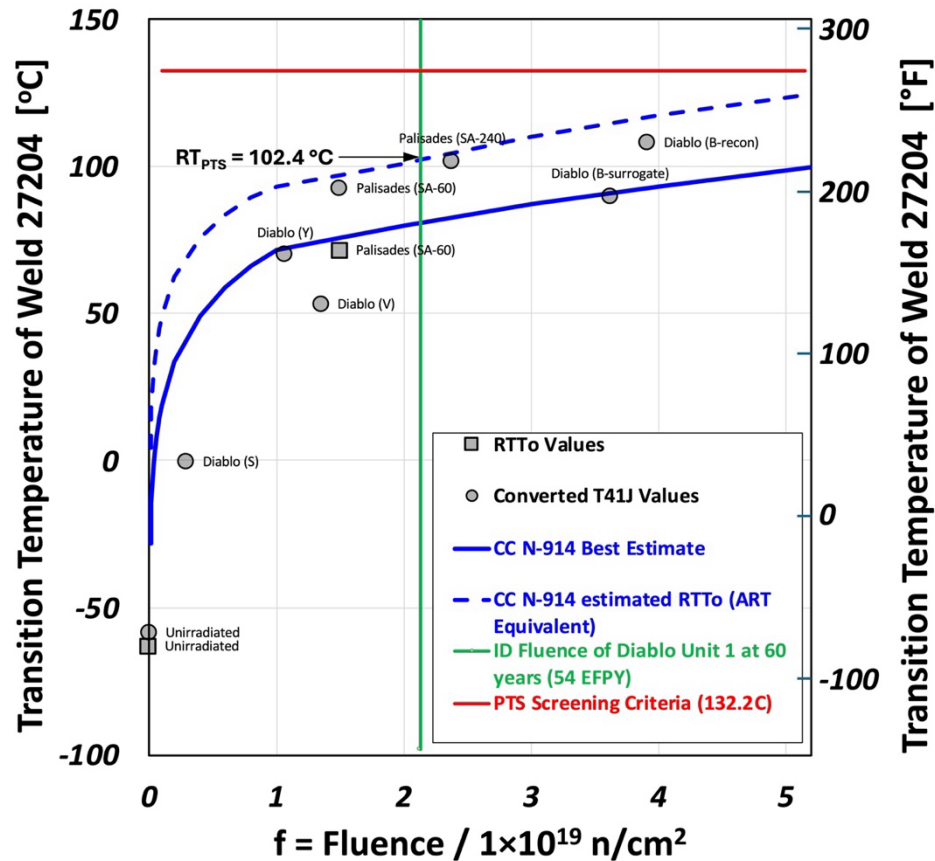


Figure 11. PTS assessment of Unit 1 using the method of draft ASME CC N-914.

5.4. Summary

Analysis of the new measurements of Charpy samples from weld wire heat 27204 irradiated in Capsule B and reported in [WCAP-19173] confirms, using the NRC's required methodology, that Unit 1 remains below the PTS screening criteria at the end of its currently licensed 60-years of operation, and that it would take approximately 118 years to reach the level of embrittlement associated with the screening criteria. This forecast differs slightly (a reduction of 1.2°C in the RT_{PTS} value) from the prior forecast for Unit 1, made in 2011, that did not include the Capsule B data. Relative to the 2011 forecast the update reported here has a

slightly lower value of RT_{PTS} and, consequently, a slightly longer predicted time before the PTS screening criteria is exceeded.

The predictions of the NRC's method were checked using a more up-to-date method now being developed by Section XI of the American Society of Mechanical Engineers and designated as CC N-914. This assessment, which used all of the available Charpy data, including that from Capsule B, as well as fracture toughness data that is available for weld wire heat 27204, confirmed the conservatism of the NRC analysis.

6. USE Analysis

6.1. Evaluation Using the Required NRC Method

Figure 12 shows the four data for the Diablo Canyon Unit 1 surveillance weld analyzed according to the NRC's approach¹⁵, which was summarized in Section 3.3. The four circles show the data available prior to 2025 from capsules S, Y and V along with the new datum from Capsule B, which is labeled. The red curve shows the [RG1.99R2] USE prediction for weld wire heat 27204 without consideration of any USE data from Unit 1; this prediction is expected to lie below available data. The black curve adjusts the red curve to the data as required by the NRC's procedure¹⁶. The new data from Capsule B lies on this assessment curve established by data from Capsule Y, but does not alter it. The analysis results are as follows for the 60-year ¼-T fluence of 1.27×10^{19} n/cm²:

- Percent drop = 32.8% (rounded to 33% on the figure).
- Irradiated USE at 60 years = Unirradiated USE (118.8J) $\times$ (1-0.328) = 79.8J (rounded to 80J on the figure).

¹⁵ As discussed in [Kirk Part 2], USE data from samples other than the surveillance sample are not considered by the NRC's process, so these data are not used on Figure 12. Data from the sister and surrogate welds will be considered as part of the supplemental evaluation performed in Section 6.2.

¹⁶ As explained in Section 2.4.3 of [Kirk Part 1], unlike the analysis for ΔT_{41J} , there is no explicit uncertainty treatment associated with the %drop or Δ USE estimates according to the approach used here, which is as described by [RG1.99R2] and continues to be accepted by the NRC. Contemporary NRC evaluation of the circa-1980s [RG1.99R2] procedure for USE demonstrates that these procedures bound roughly 80% of available data [Widrevitz 2019]. The NRC continues to accept the screening procedure described in [RG1.99R2] and used here at least in part because evaluations of plants that fail the 68J screening are consistently evaluated as being acceptable following the more detailed and accurate assessment using elastic-plastic fracture mechanics described in [RG1.161].

These results, which comply with the NRC's 68 J screening criteria, are indicated on the figure by blue lines. This analysis also indicates that the 68J screening criteria would be reached at a $\frac{1}{4}T$ fluence of 3.9×10^{19} n/cm², which exceeds the fluence at the end of the 60-year licensed lifetime by more than a factor of three.

6.2. Supplemental Evaluation Using an Expanded Data Set

Figure 13 shows the seven USE measurements available for weld wire heat 27204, which includes the four data for the Diablo Canyon Unit 1 surveillance weld, two data from the sister plant weld, and one datum from the surrogate weld. Consideration of USE values from the sister and surrogate weld are not required by NRC procedures for USE evaluations. Nevertheless, this analysis was performed to estimate the USE after 60 years using all available data for weld wire heat 27204. As documented in a 2024 memo to the DCISC [Kirk 2024], analysis of this combined data set requires that the unirradiated Charpy data from the surveillance and the sister/surrogate welds be combined and fit together, which produces an unirradiated USE of 131.4J¹⁷.

The red curve shows the [RG1.99R2] USE prediction for weld wire heat 27204 without consideration of any measured USE data. The black curve adjusts the red curve based on the data as required by the NRC procedure. The Capsule B data from both the surveillance and surrogate welds lie above this assessment curve, whose position is established by one of the measurements from the sister plant. The analysis results are as follows for the 60-year $\frac{1}{4}T$ fluence of 1.27×10^{19} n/cm²:

- Percent drop = 43.4% (rounded to 43% on the figure).
- Irradiated USE at 60 years = Unirradiated USE (131.4J) $\times$ (1-0.434) = 74.4J (rounded to 74J on the figure).

These results, which comply with the NRC's 68 J screening criteria, are indicated on the figure by the blue lines. This analysis also indicates that the 68J screening criteria would be reached at a $\frac{1}{4}T$ fluence of 2.3×10^{19} n/cm², which exceeds the fluence at the end of the 60-year licensed lifetime by a factor of 1.8.

6.3. Summary

Analysis of the new measurements of Charpy samples from weld wire heat 27204 irradiated in Capsule B and reported in [WCAP-19173] confirms, using the NRC's required methodology, that Unit 1 remains above the USE screening criteria at the end of its currently licensed 60-years of operation. The predictions of the NRC's method were checked using a data set consisting of all measurements from weld wire heat 27204, including both data from the sister weld and surrogate weld; a data set that is 75% larger than that required by the NRC. This

¹⁷ This USE value was reported incorrectly in [Kirk 2024] as 132.4J due to a typographical error. The error has been corrected in the analysis reported herein

supplemental analysis also shows that Unit 1 remains above the USE screening criteria at the end of its currently licensed 60-years of operation.

Specific values estimated by these analyses are as follows:

- Required NRC analysis
 - USE at $\frac{1}{4}T$ fluence = 80J after 60-years of operation, 12J above the screening criteria.
 - Unit 1 will not reach the USE screening criteria until a fluence the exceeds 3-times the value at 60-years for operation.
- Supplemental analysis
 - USE at $\frac{1}{4}T$ fluence = 74J after 60-years of operation, 6J above the screening criteria.
 - Unit 1 will not reach the USE screening criteria until a reaching a fluence that is 1.8 times the value at 60-years for operation

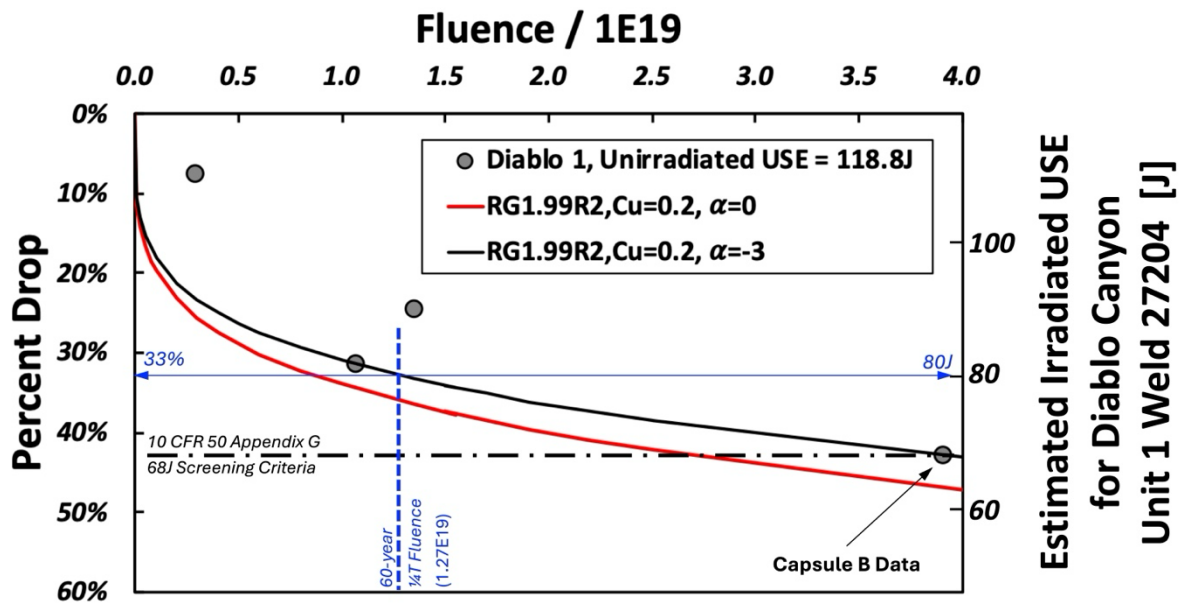


Figure 12. USE evaluation using only weld wire heat 27204 surveillance data from Diablo Canyon Unit 1.

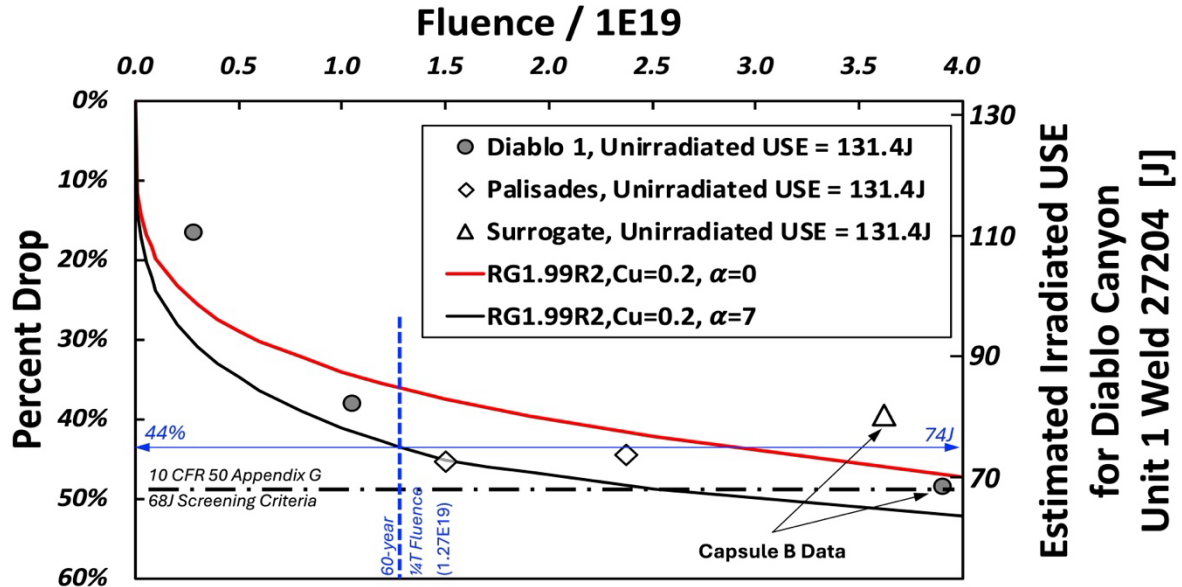


Figure 13. USE evaluation using weld wire heat 27204 surveillance data from Diablo Canyon Unit 1 supplemented by data from the Palisades sister welds and from the surrogate weld, which are both also heat 27204.

7. Assessment of these Results Relative to Possible Regulatory Changes, and Options for Unit 1

7.1. Impact of Possible Regulatory Changes on Unit 1

Informed by technical evaluations performed in [Widrevitz 2019] and in [Widrevitz 2023], the NRC staff requested permission from the NRC Commissioners to initiate a rulemaking aimed at updating NRC guidance on embrittlement prediction and surveillance monitoring [SECY-22-0019]. The Commission gave permission for this rulemaking to proceed three years later [NRC 2025b]. The NRC recently published a letter and white paper describing the NRC staff's current thinking on how they will execute the rulemaking plan described in SECY-22-0019] in a manner consistent with the Commissions direction described in [NRC 2025b] In general, the effort described in SECY-22-0019 aims to bring NRC guidance and requirements on embrittlement trend prediction and on surveillance monitoring into better alignment with current technical understandings and practices, a needed effort because current NRC guidance and requirements on these matters have not been updated for decades.

In [SECY-22-0019] and [NRC 2026b] the NRC staff identified three technical topics that could be affected by this rulemaking. In the following the NRC statements are reviewed and the impact on Unit 1 is assessed:

- **Embrittlement Trends Prediction**
 - Summary of NRC Statements: The NRC has expressed its intention to update its guidance on embrittlement trend prediction. Specifically, the NRC stated that it is considering revising its PTS rule (10CFR50.61) to *"require the use of an NRC-approved ETC augmented with the appropriate margins at high neutron fluences (6×10^{19} n/cm²)"* and that *"the staff is*

considering an update to RG 1.99, Rev. 2, such that the use of the ETC in E900-15 would be approved for all neutron fluences up to 2×10^{20} n/cm², while the ETC in RG 1.99, Rev. 2, would be approved only for neutron fluence levels less than 6×10^{19} n/cm²." These changes are needed to correct a tendency of the [RG1.99-R2] ETC to under-predict values of ΔT_{41J} at such high fluence values.

- Impact on Diablo Canyon Unit 1: If this change is made it will have no impact on Unit 1. The projected fluence for Unit 1 after 60 years of operation is at most $\approx 2.2 \times 10^{19}$ n/cm² while the highest surveillance capsule fluence is $\approx 3.9 \times 10^{19}$ n/cm². Both of these fluence values are below the fluence of 6×10^{19} n/cm², above which the NRC anticipates making changes to its regulatory guidance. Thus, the NRC's changes, if made, will have no impact on Unit 1 for a license term of 60-years.
- **Surveillance Monitoring**
 - Summary of NRC Statements: The NRC has expressed its intention to update its guidance concerning the required withdrawal schedule for surveillance capsules. Specifically, the NRC has stated that *"during periods of extended operation, the requirements would be revised to ensure that data from additional surveillance capsule(s) with a neutron fluence equal or greater than the peak RPV neutron fluence for extended operation is tested at an appropriate periodicity (e.g., every 20 years) to ensure adequate performance monitoring."* This change is being considered to ensure adequate performance monitoring for periods of multiple license renewals.
 - Impact on Diablo Canyon Unit 1: As illustrated in Figures 9 and 12, with the testing of Capsule B Unit 1 now has three measurements at a fluence value exceeding the anticipated fluence after 60-years of operation. Thus, on a fluence basis there is no need for further capsule withdrawals, and indeed Unit 1 has satisfied all current monitoring requirements. An NRC-caused need for Unit 1 to test another capsule would only occur if (1) the testing periodicity adopted by the NRC is significantly less than 20-years and (2) the NRC takes no account of capsule fluence as compared to vessel fluence in developing its updated capsule withdrawal requirements. Thus, these changes could impact Unit 1, but the likelihood of this outcome is, in the opinion of this consultant, low.
- **Use of Direct Fracture Toughness (Master Curve)**
 - Summary of NRC Statements: The NRC has expressed its intention to *"consider allowing the use direct fracture toughness measurements to demonstrate the integrity of the RPV."* Specifically, the NRC has stated that Master Curve approaches may be considered in the future without the need for case-specific regulatory review.

- Impact on Diablo Canyon Unit 1: If this change is made it would have no operational impact for the foreseeable future (i.e., for 60-years of operation) on Unit 1. As documented in Section 5.3 of this report, Master Curve data is already available for weld wire heat 27204. Use of these data long with Charpy data in an analysis following the guidance of draft CC N-914 reduces the RT_{PTS} value compared to that estimated using current NRC requirements. Thus, if this change is implemented it would improve the perceived safety of Unit 1.

The NRC has not yet initiated the rulemaking discussed in [SECY-22-0019, NRC 2025b, NRC 2026b]. Prior experience with the time needed for rulemaking suggests that the changes discussed in these documents are unlikely to impact the Code of Federal Regulations until 2027 or later.

7.2. Options for Unit 1

From the viewpoint of regulatory compliance there is no need for Unit 1 to take any further actions to demonstrate RPV integrity at the current time, nor is there currently an NRC requirement to withdrawal and test another capsule before the end of the current 60-year license term.

As reviewed in the previous section, the possible rule changes discussed by the NRC in [SECY-22-0019, NRC 2025b, NRC 2026b] are unlikely to adversely impact Unit 1. The only possible impact, albeit unlikely in the opinion of this consultant, is that the NRC may change its rules to require the withdrawal and testing of another capsule before the end of Unit 1's current 60-year license term. In this situation Capsule A (now in standby status) and Capsules C and D (now withdrawn and stored) could be considered for testing as all contain samples of weld wire heat 27204. In all cases these capsules would likely need to be moved or re-inserted to a higher lead factor location so that they could provide data at a relevant fluence.

Moving beyond the NRC's regulatory requirements, a number of options exist across a broad range of difficulty and expense that could be undertaken to increase public confidence that the level of embrittlement in Unit 1 is not detrimental to safe operation. The lowest risk and most effective of these options is, in my opinion, to fabricate from samples of weld 27204 (that is: either unirradiated samples from archive materials or irradiated samples from prior capsule testing) additional fracture toughness specimens for testing according to [ASTM E1921] and analysis following draft CC N-914, as was reported here in Section 5.3. These samples would grow the available data set and, thereby, increase confidence in the conclusions.

Other options could be discussed if it is the DCISC's desire to do so.

8. Summary

In April 2025, surveillance Capsule B was removed from Unit 1 at the Diablo Canyon Nuclear Power Plant and sent to the Westinghouse hot cell laboratory in Pittsburgh Pennsylvania for testing. The results of this testing, which was reported in June 2026, included two new measurements of embrittlement made using Charpy V-notch specimens from weld wire heat 27204. Welds made from heat 27204 are the most sensitive to neutron irradiation embrittlement of the various steels from which the beltline of the reactor pressure vessel in Unit 1 was fabricated and, as such, most limit the operation of that Unit.

This report independently confirms the “credibility” analysis of the Charpy V-notch data set for weld wire heat 27204 reported by Westinghouse in Appendix D of [WCAP-19173]. This data set includes the two new measurements from Capsule B. These measurements were performed over a range of neutron exposure spanning conditions characterizing the beginning of Unit 1 operation in the late 1980s to conditions that will not be reached until long after expiration of its current operating license in 2045.

This report used the new measurements on weld wire heat 27204 along with existing data on this weld wire heat to assess the status of Unit 1 with respect to the NRC’s screening criteria on pressurized thermal shock (PTS) as expressed in [10CFR50.61] and with respect to the NRC’s screening criteria on upper shelf energy (USE) as expressed in [10CFR50-AppG]. The results were as follows:

- **Pressurized Thermal Shock**
 - NRC Method: An analysis performed using the NRC’s required analysis procedures as expressed in [10CFR50.61] determined that Unit 1 is forecast to remain 18.4 °C (33.2 °F) below the PTS screening criteria after 60-years of operation and, therefore, remains in compliance with those requirements through that time. The analysis also forecast that the embrittlement of weld wire heat 27204 will not reach the screening criteria until a neutron exposure nearly twice that expected in Unit 1 after 60-years of operation. These results are very

similar to prior PTS assessments of weld wire heat 27204 that were based on five measurements, which have been available since 2011 and serve as the current licensing basis

- Supplemental Analysis: Additionally, a supplemental analysis was performed using a more up-to-date analysis method as described in draft ASME Code Case N-914; this method is currently under review by the American Society of Mechanical Engineers. Use of CC N-914 provides an independent check on the conclusions based on the NRC's analysis method. The CC N-914 analysis used fracture toughness data on weld wire heat 27204 that is not used by the NRC. The CC N-914 analysis assessed Unit 1 to be 29.8 °C (53.7 °F) below the NRC's PTS screening criteria, which confirms the conservatism of the NRC analysis.
- **Upper Shelf Energy**
 - NRC Method: An analysis was performed using the new measurements of Charpy samples from weld wire heat 27204 irradiated in Capsule B and reported in [WCAP-19173]. Using the NRC's required methodology, this analysis predicts that Unit 1 will have a USE after 60-years of 80J, which is 12J (8.9 ft-lb) above the NRC's USE screening criteria and, thus, complies with this NRC requirement. The analysis also forecasts that the USE of weld wire heat 27204 will not reach the screening criteria until it reaches a neutron exposure over three times that expected in Unit 1 after 60-years of operation.
 - Supplemental Analysis: The predictions of the NRC method were checked using a data set consisting of all measurements from weld wire heat 27204, including both data from the sister weld and surrogate weld. Together these data make a set that is 75% larger than that required by the NRC. This supplemental analysis predicts that Unit 1 will have a USE after 60-years of 74J, which is 6J (4.4 ft-lb) above the NRC's USE screening criteria and, thus, complies with NRC requirements. The analysis also forecasts that the USE of weld wire heat 27204 will not reach the screening criteria until it reaches a neutron exposure of approximately 1.8-times that expected in Unit 1 after 60-years of operation.

Finally, this report reviewed possible future changes to embrittlement and surveillance guidelines that have been recently discussed by the NRC to assess their potential impact on Unit 1. These include a possible change to the NRC's method of forecasting embrittlement, a possible change in surveillance requirements after the original 40-year license term, and the possible allowance by the NRC for plants to use fracture toughness data in their assessments. The impact of these changes on Unit 1, were they to occur, is as follows:

- There is no impact of the possible change in embrittlement formula because the NRC does not anticipate changing this formula except for fluence values much higher than Unit 1 will experience in its first 60-years of operation.
- The impact of changes to surveillance requirements on Unit 1 is more difficult to assess because the NRC has provided limited information about its future intentions. If future requirements consider the relationship of capsule fluence to reactor fluence then no more capsules should be required of Unit 1 because Capsule B was withdrawn and

tested at a fluence over 2-times the predicted 60-year fluence. However, if the NRC's new capsule withdrawal schedule considers only the interval in years between surveillance capsule withdrawals then another capsule may be required before the end of the 60-year license. If this is the case Unit 1 has three capsules all containing the limiting weld wire heat that could be used to satisfy such a testing requirement.

- In what the NRC has written the use of fracture toughness data has been expressed as an option, not a requirement. The supplemental PTS analysis reported here used fracture toughness data for weld wire heat 27204 along with draft ASME CC N-914. This use of fracture toughness data demonstrated a greater distance between the assessed RT_{PTS} value for Unit 1 and the NRC's screening criteria.

9. Citations

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