Treatment of Embrittlement Uncertainties and Performance Monitoring

Mark Kirk · 3 October 2024

1. Background

Professor Najmedin (Najm) Meshkati of the Diablo Canyon Independent Safety Commission (DCISC) noted the following quotation from page 31 of a report by the Nuclear Regulatory Commission (NRC) staff [NRC 2022] (***emphasis*** added):

> *"The risk-informed analysis in Section 6 highlights the synergistic effect of the ETC and performance monitoring on RPV integrity. Although the probability of RPV rupture from Reference 17 remains generically low, the impact of the embrittlement* ***uncertainty*** *on adequate safety margins and combined with insufficient* ***performance monitoring****, impact the staff's confidence in the RPV integrity and challenge their finding of reasonable assurance of safety in long-term operation. To restore confidence in long-term RPV integrity, regulation and guidance changes are necessary to implement use of an accurate ETC and ensure continued performance monitoring through surveillance capsule testing."*

Motivated by these statements, Professor Meshkati asked two questions:

1. **Regarding "uncertainty"**: Please state all the underlying assumptions in the [Part 2 Rpt] uncertainty characterization and their rationale and treatment method in the analyses.
2. **Regarding "performance monitoring"**: What does sufficient "performance monitoring" in this context entail? What are its criteria? What is the standard industry practice for such? How can they be ascertained/assessed? How does [the Diablo Canyon Power Plant] do such performance monitoring and determine its adequacy? What constitutes "insufficient" (performance monitor) and how can it be assessed?

This document will address these questions. The remainder of the document is organized as follows.

- Section 2 summarizes the information in [NRC 2022] and from [SECY-22-0019] that gave rise to the paragraph quoted by Professor Meshkati. This summary provides better context for and understanding of quoted statement from [NRC 2022].
- Section 3 addresses question #1 concerning uncertainty.
- Section 4 addresses question #2 concerning performance monitoring.

- Section 5 summarizes the information from Sections 3 and 4 and discusses implications for Diablo Canyon Unit 1

2. Recent NRC Documents

Document Summary

The quotation cited by Professor Meshkati from [NRC 2022] provides the technical basis for the NRC staff's request to the Commission, in March 2022, for permission "*to conduct rulemaking to amend the reactor pressure vessel (RPV) embrittlement and surveillance requirements in Title 10 of the Code of Federal Regulations (10 CFR) Part 50*" [SECY-22-0019].

Concerning embrittlement prediction, [SECY-22-0019] states that, if approved, the rulemaking would:

> *"revise the RPV embrittlement and surveillance requirements in 10 CFR Part 50. The rulemaking would revise Appendix H, "Reactor Vessel Material Surveillance Program Requirements," to 10 CFR Part 50 to include additional surveillance testing requirements for long-term operation and a revised fluence function fit (either a new embrittlement trend curve (ETC) or an update to existing trend curves) in the applicable regulations and implementing guidance for all materials that will experience high neutron fluence levels."*

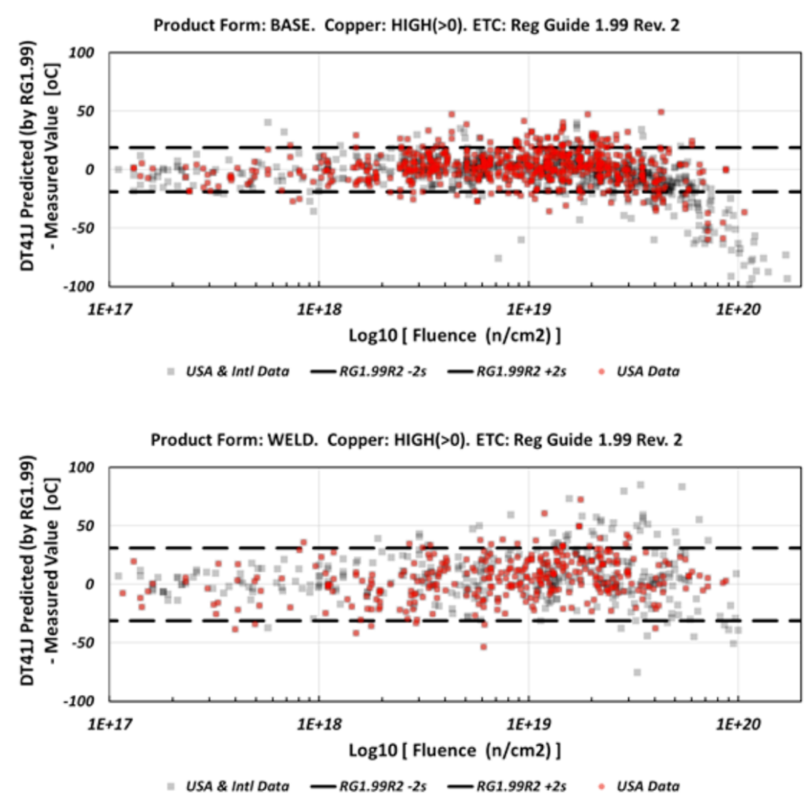
[SECY-22-0019] further states that two reasons motivations for these revisions:

- Underprediction by the Embrittlement Trend Curve (ETC): *"This revision is necessary because of an underprediction of RPV material neutron embrittlement with the high fluences that will be reached at some pressurized-water reactor (PWR) plants in long-term operation (e.g., beyond 60 years of operation)."*
- Deferral of Surveillance Capsule Removal and Testing: *"Under current regulatory practice, licensees can defer, and some have deferred, surveillance capsule testing required by Appendix H to 10 CFR Part 50 that is intended to confirm embrittlement predictions from the ETC model."*

The following paragraphs provide additional information on these two topics:

- Embrittlement: The NRC staff performed a detailed review of the continued adequacy of the Regulatory Guide 1.99 embrittlement trend curve [RG1.99R2], which was documented in [Widrevitz 2019] and is referenced by both [NRC 2022] and [SECY-22-0019]. This review documented several inadequacies with the [RG1.99R2] ETC when its predictions were compared to ΔT_{41J} data obtained by surveillance program testing. Most significantly, as shown in Figure 1 the Staff identified that the RG1.99R2 ETC begins to under-predict the measured magnitude of embrittlement (ΔT_{41J}) to a larger and larger degree as fluence increases. The staff identified other aspects of the [RG1.99R2] procedure that could be improved, including a revision of the current approach to credibility evaluation which, is not optimal for lower copper steels.

Figure 1 – The prediction error of the [RG1.99R2] ETC (vertical axis) for base metals (top) and welds (bottom) plotted as a function of fluence. Each dot on the figure is represents a surveillance measurement while the horizontal dashed lines are the [RG1.99R2] ± 2σ tolerance bounds. This figure is copied from Figure 2-6 of the NRC staff's detailed evaluation of the [RG1.99R2] ETC [Widrevitz 2019]. Despite its labeling the data points shown are for all values of copper in the NRC's surveillance database.



- Surveillance Monitoring: As was described for the DCISC in Section 3.2.3.2 of the [Part 1 Rpt] and in Section 2.3 of the [Part 2 report] (and reinforced in [NRC 2022]), the only legal **requirements** for surveillance monitoring pertain to the original 40-year licensing term. These are established by 10CFR50-AppH], which incorporates [ASTM E185-82][1] by reference. The NRC has issued no requirements, only **guidance**, for operation beyond 40 years (see [NUREG-0800], [NUREG-1801], and [NUREG-2191V2]). This lack of requirements for operation beyond 40-years

[1] If a plant was built in a year before 1982, [10CFR50-App H] states that the surveillance requirements of earlier editions of ASTM E185 are applicable. See Section 2.3 of the [Part 2 Rpt] for a further explanation.

coupled with the language[2] in [ASTM E185-82] has enabled some licensees to, with complete legality, delay repeatedly the withdrawal and testing of their last capsule during periods of license renewal, and to not test supplemental capsules. [NRC 2022] provides graphs illustrating several examples of plants that have delayed capsule withdraw, one of which appears here as Figure 2 (this plant is not Diablo Canyon Unit 1). The highest fluence surveillance data available for this plant corresponds to approximately 25 years of operation, however the plant has operated for 52 years. Thus, for over half of its operating life the fluence on the RPV inner diameter of this plant has exceeded that of the highest available plant-specific surveillance data. In [NRC 2022] the NRC staff state that no legal means currently exist to force this plant to test its last capsule because the plant remains in compliance with [ASTM E185-82].

Figure 2 – Illustration from [NRC 2022] of how one plant has delayed withdrawal of its last capsule.

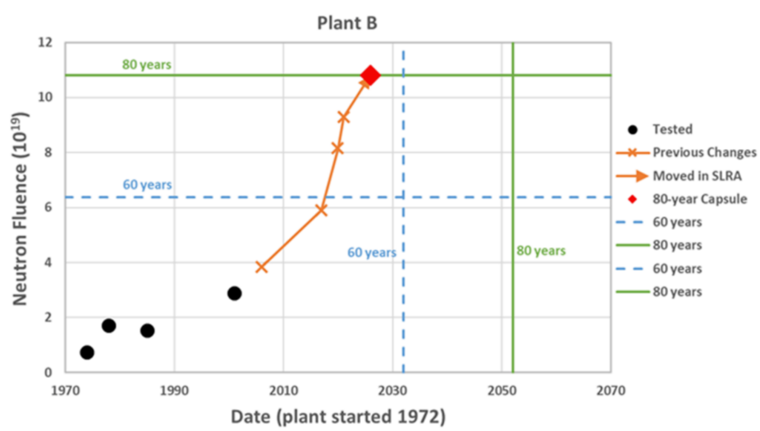


Figure 8 Capsule withdrawal history for Plant B, where the withdrawal of the capsule originally designed to apply to 40 years has been delayed multiple times and is currently credited to address 80 years of plant operation, and the highest-fluence data represent about 25 years of plant operation

The NRC staff concluded in [NRC 2022] that these shortcomings do not pose a current safety risk and do not comprise the current integrity of RPVs, stating:

"Using a risk-informed approach, the staff has determined that the combined effect of the underprediction and testing deferral practice could impact the

[2] The NRC explains this situation in [SECY-22-0019] as follows: "*Licensees are able to delay their final capsule from their original 40-year program because the ASTM E 182-82 standard was not developed to explicitly account for design lives beyond 40 years. ASTM E 185-82 states that the last capsule can be withdrawn when the capsule fluence is "not less than once or greater than twice the peak end-of-life vessel fluence." Therefore, delaying the final capsule from the original 40-year program to the initial period of extended operation, and then again to the subsequent period of extended operation is still in conformance with the ASTM standard.*"

staff's long-term confidence in the integrity of the RPV for certain plants—that is, about 10 years from now. These circumstances recommend change to the current regulations to ensure the safety margins and performance monitoring necessary to provide reasonable assurance that RPV integrity will be maintained over the extended operating lifetime of each plant. The staff's analysis confirms that reactors operating today provide reasonable assurance of adequate protection of public health and safety. This issue is a potential long-term concern that may only affect certain plants and does not compromise the current integrity of the RPVs."

The [SECY-22-0019] rulemaking request has been before the Commission since March 2022. So far, no action has been taken.

Impact of these NRC Staff Concerns on Diablo Canyon Unit 1

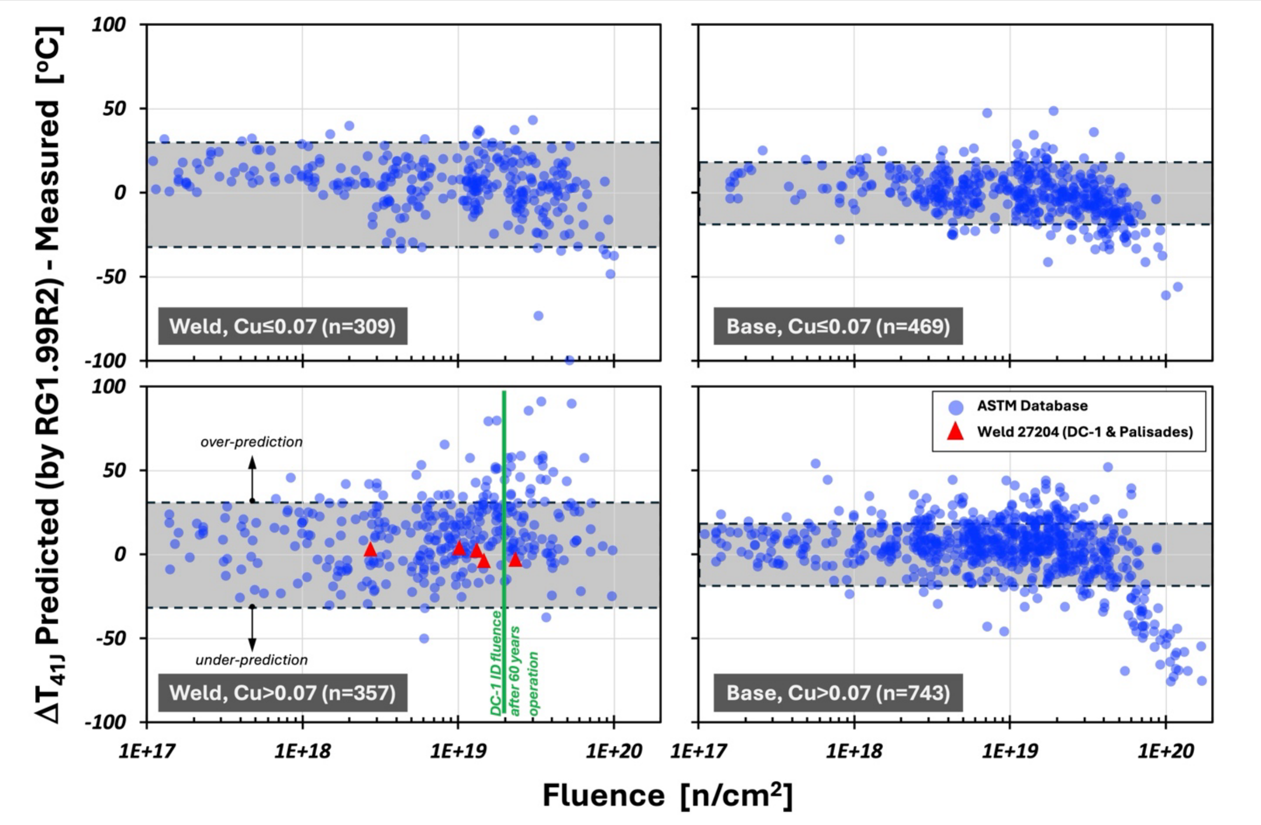
While the information in [Widrevitz 2019], [NRC 2002], and [SECY-22-0019] indicate a general need to update NRC guidance and rules concerning embrittlement trend prediction and surveillance monitoring, the concerns expressed in these documents do not adversely impact the operating safety of Diablo Canyon Unit 1 for the following reasons:

- <u>Embrittlement prediction</u>: Further analysis of the NRC's database, which they obtained from [ASTM Adjunct] appears as Figure 3. Figure 3 distinguishes the data by both product form (welds appear in the left column, base metals in the right) as well as by copper content (lower copper content steels appears in the upper row, higher copper content in the lower row). Figure 3 shows that the NRC-identified tendency towards ΔT_{41J}under-prediction, which is non-conservative and thus undesirable, is most clearly manifest by available data from base materials above a fluence of ≈3-4×10^{19} n/cm^2. Conversely, higher copper content welds (lower left graph in Figure 3) have ΔT_{41J} values that are, if anything, somewhat over-predicted by [RG1.99R2]. The limiting material in Diablo Canyon Unit 1 is a high copper (Cu=0.20 weight percent) weld. Data from weld wire heat 27204 are overlaid on the lower-left graph, as is the predicted end-of-60-year fluence value for Diablo Canyon Unit 1. These data illustrate that for the ΔT_{41J}measurements of the Diablo Canyon Unit 1 weld are well predicted by the NRC's methodology, and that the 60-year inner diameter fluence of Diablo Canyon Unit 1 is 2×10^{19} n/cm^2, which is well below the fluence at which the data show that the predictions of the [RG1.99R2] ETC become non-conservative.
- <u>Surveillance monitoring</u>: As documented Section 2.3 of the [Part 2 Rpt], Diablo Canyon Unit 1's Capsule B has been delayed several times. However, this situation differs markedly from the NRC's example of Figure 2. For Diablo Canyon Unit 1, the maximum fluence of the surveillance data from the Unit 1 surveillance program exceeds the 40-year RPV fluence. When combined with similar data on weld wire heat 27204from the Palisades "sister plant," the Diablo Canyon Unit 1 data set

extends to a RPV fluence well beyond 60 operating years. Thus, while the example plant from Figure 2 has had a RPV fluence above its maximum surveillance fluence for over 25 years, this is not the situation for Diablo Canyon Unit 1. The Diablo Canyon Unit 1 RPV fluence has been, and continues to be, well bounded by surveillance data, demonstrating the sufficiency of Diablo Canyon Unit 1's performance monitoring (i.e., surveillance) program.

Thus, the general concerns on embrittlement prediction non-conservatism and on inadequate surveillance monitoring expressed in [Widrevitz 2019], [NRC 2002], and [SECY-22-0019] do not apply to Diablo Canyon Unit 1.

Figure 3 – Assessment of the effects of product form and copper content on the accuracy of [RG1.99R2] ΔT_{41J} predictions. The shaded region between the dashed lines represents the ±2σ tolerance bounds on the [RG1.99R2] ΔT_{41J} predictions.



3. Treatment of Embrittlement Uncertainty

The statement from [NRC 2022] quoted by Professor Meshkati (see page 1) expresses concern about "*the impact of the **embrittlement uncertainty** on adequate safety margins,*" not about uncertainty in general. Since this concern about embrittlement uncertainty treatment in [RG1.99R2] was one of two major motivators for the [SECY-22-

0019] rulemaking request, this discussion focuses on embrittlement uncertainty and how it is addressed by [RG1.99R2].

Section 2.4.2 of the [Part 1 Rpt] described in detail the NRC's procedures from the PTS rule [10CFR50.61] and from [RG1.99R2] to address embrittlement uncertainty (that is, uncertainty in the estimation of ΔT_{41J}]. That procedure is used to calculate the adjusted reference temperature (ART), which was described in Section 3.1 of the [Part 2 Rpt][3]. For ΔT_{41J} these procedures stipulate that the standard deviation of ΔT_{41J}have the following values:

	Not Credible Data Set	Credible Data Set
Base Metals	9.4 °C	4.7 °C
Weld Metals	15.6 °C	7.8 °C

The margin term that is part of the ART formula (see equation (3-1) in the [Part 2 rpt]) uses these values to ensure that ART values conservatively over-estimate measured ΔT_{41J} data. The shaded regions of the graphs in Figure 3 of this document represent ±2σ bounds using the not-credible σ values from the above table (the not-credible σ values are appropriate to use in Figure 3 because the data in that figure are not adjusted for plant-specific effects). At the lower fluences relevant to Diablo Canyon Unit 1 up to 60-years of operation these ±2σ bounds enclose an appropriate percentage of available data. At higher fluences too much of available data falls below the -2σ tolerance bound, which is not conservative. As discussed earlier, this observation has motivated the NRC staff's request for rulemaking, but the deficiencies of the [RG1.99R2] ETC at these high fluences is not pertinent to any operational condition at Diablo Canyon Unit 1 through at least 60-years.

In Section 4.3 of [Widrevitz 2019] the NRC evaluated the appropriateness of the reduction in σ value that is permitted for credible data sets. The NRC analysis focused on sets of surveillance data of sufficient size to enable estimating σ with reasonable accuracy. Figure 4-9 in that report shows that for all weld data sets examined the 7.8 °C value for σ given in the above table over-estimates the value of σ determined directly from the data (most base metal values were likewise over-estimated by the 4.7 °C value). Thus, the NRC's analysis in [Widrevitz 2019] demonstrates the appropriateness and conservatism of existing guidance for credible data sets.

In summary, for the conditions of interest at Diablo Canyon Unit 1 through 60-years of operation, the NRC's [RG1.99R2] procedure includes an appropriate treatment of embrittlement (ΔT_{41J}) uncertainties. For plants that will operate at higher fluences the NRC has identified that both their uncertainty treatment and embrittlement trend predictions

---

[3] Estimates of ART calculated by this consultant appear as the curves labeled ART in Figure 2 and in Figure 3 of the [Part 2 Rpt]. All ART curves in these figures include the uncertainty treatment stipulated in [RG1.99R2].

require improvement, which is why the staff made a rulemaking request [SECY-22-0019] to address these issues.

4. Performance Monitoring

This section will first describe how performance monitoring of the RPV in a nuclear power plant is conducted. Next the question of "sufficient" performance monitoring is addressed. Finally, the sufficiency of performance monitoring at Diablo Canyon Unit 1 is discussed.

How Performance Monitoring is Now Conducted for the RPV

RPV performance monitoring has two components:

- **Surveillance capsules** contain samples of the base and weld material used in RPV fabrication. These samples monitor the embrittlement of the RPV beltline materials over the reactor's operating lifetime. The capsules are withdrawn over the life of the plant to measure the progressive embrittlement of these samples. Because the capsules are placed inside the reactor, they typically accumulate irradiation dose, and therefore embrittlement, faster than the RPV itself. This acceleration of the damage rate provides advance information, often many years in advance, on the future embrittlement of the reactor, which provides time for regulatory review and compensatory actions if they are needed.
- **Ultrasonic (UT) Inspections** of the welds and adjacent base metal in the RPV shell region near the active core are performed procedures of Section XI of the ASME Boiler and Pressure Vessel Code. These inspections do not monitor embrittlement because embrittlement does not manifest as flaws of a size detectable by UT. Rather these inspections monitor for increases in defect size due to cracking mechanisms such as fatigue. UT inspections are required by the NRC [10CFR50.55a] every 10-years[4] to look for indications of flaws in and near the structural weld seams.

These two data sources provide plant-specific evidence on the evolving condition of the RPV. However, these data are not used alone when assessing the current and future suitability of the RPV for continued service.

- Embrittlement data on ΔT_{41J} and % USE DROP is generated by testing Charpy V-notch samples from the **surveillance capsules** following ASTM requirements [ASTM E23]. The ΔT_{41J} and % USE DROP data, which quantify the effect of embrittlement on the RPV steels, is used as described in Section 2.4 of the [Part 1 Rpt] along with an embrittlement trend curve (ETC) to predict future embrittlement.

[4] If plant-specific justification is provided by the utility the UT inspection interval may be extended to 20-years [WCAP-16168].

Ideally the maximum fluence from available surveillance data exceeds fluence at the inner diameter of the RPV wall (this is the case for Diablo Canyon Unit 1). It is the combination of the plant-specific surveillance data along with the ETC that forms the performance monitoring program, which is used to account for the effects of embrittlement on fracture toughness. This estimate is used as input to fracture mechanics equations to ensure that fracture toughness remains adequate (that is, larger than the driving force to fracture) during both routine operations [ASME-SC-XI App-G] and for postulated accidents like PTS [10CFR50.61]. Beyond these required forecasting procedures, it should be noted that after more than half a century of surveillance data collection by the industry there is now considerable data available (> 2,000 ΔT_{41J} values have been aggregated by ASTM [ASTM Adjunct]) as well as a better understanding of embrittlement trends, as expressed by more modern ETCs than RG1.99R2 [ASTM E900-15, JEAC4201, Soneda 2013]. In this situation, and especially for plants like Diablo Canyon Unit 1 that already have many years of service, collection of surveillance data typically becomes more of a check on expectations than a revelation of new and completely unexpected trends. Especially for lower fluence plants like Diablo Canyon Unit 1 there are not usually surprises when new surveillance data are obtained.

- As described in Section 3.3.2.2 of the [Part 1 Rpt], NRC regulations and the ASME Code require a **UT inspection** of the reactor vessel beltline welds and adjacent base materials once every 10-years, see [10 CFR 50.55a] paragraph (g). These periodic inspections monitor flaws that may exist within the RPV beltline to determine if they increase in size over time. The 10-year inspection interval was established at the beginning of electricity production by nuclear power, a time when there was very little experience concerning crack growth rates. Now decades of operational experience demonstrates that there are no time-dependent cracking mechanisms that significantly increase the size of any original fabrication defects that may remain in the RPV:
    - Concerning stress corrosion cracking, Section 3.3.3.2 of [NUREG-1806], which was also summarized in Section 3.3.3.2 of the [Part 1 Rpt] to the DCISC, describes how chemistry control of the primary coolant keeps oxygen levels low. This keeps the electrochemical potential sufficiently low to inhibit SCC initiation and growth.
    - Concerning fatigue, low operational stresses and long operating cycles combine to limit the amount of fatigue crack growth in the RPV shell over the lifetime of the plant to extremely small values. Between 2005-2011, the PWR Owner's group performed research to justify extension of the ISI interval from 10 to 20 years [WCAP-16168], accounting for the effects of fatigue. This risk-informed approach used the NRC's FAVOR model to demonstrate that the increase of RPV fracture risk resulting from small increases in crack depth due to fatigue lie well within the NRC's guidelines as expressed in Regulatory Guide 1.174 [RG 1.174]. Plants wishing to extend their ISI interval following this approach need to submit an exemption request to the NRC.

Thus, the UT exams of the vessel beltline have become, for all practical purposes, a re-examination of a static condition every 10 or (if approved) 20 years. Occasionally the exams will find a "new" flaw in a location where none was previously recorded; however, this is typically caused by an increase of inspection quality and accuracy over time. Regardless of if the UT inspection is performed every 10- or 20-years, any indications found during these inspections must remain below the threshold of the very small flaws allowed by the ASME Section XI flaw acceptance tables or be demonstrated to be acceptable using the conservative flaw evaluations procedures of the Code.

What Constitutes Sufficient Performance Monitoring?

Sufficient performance monitoring is defined by the work of consensus codes and standards organizations as endorsed, supplemented, and amended by the applicable regulatory authority. In the United States sufficient surveillance monitoring is required by the NRC in [10CFR50-AppH], which incorporates [ASTM E185-82] and earlier editions by reference (see Section 2.1.1 of the [Part 1 Rpt] for details). Likewise, sufficient RPV inspections are defined in [ASME SC-XI], which is incorporated into NRC regulations by reference, see [10 CFR 50.55a] paragraph (g). Section 3.3.2.2 of the [Part 1 Rpt] further describes the requirements for and rationale behind current ultrasonic in-service inspection requirements.

Sufficiency cannot be defined by one person's opinion. The consensus and regulatory processes just described ensure that the perspective of relevant stakeholders and subject matter experts are included in the determination of the required performance monitoring process. What is considered sufficient and consequently required need not remain fixed over time. Operating experience and research findings from the world-wide nuclear community are continuously considered by ASTM, ASME, and by NRC in their efforts to keep their standards and regulations up-to-date. Current processes remain adequate but can also be improved as evidenced by the information in [SECY-22-0019] as well as by continuous improvement efforts by both ASTM and ASME. For example, ASTM standards must be reviewed for adequacy and updated, if needed, roughly every 5 years [Kirk 2018]. If evidence from either operating experience or research suggested a need to augment current RPV performance monitoring requirements this would manifest as new revisions to the applicable documents.

Finally, it should be noted that there may not always be complete agreement between regulatory requirements and the guidance of ASTM and ASME. This occurs because regulatory requirements establish what is needed to maintain safety while the ASTM and ASME guidance represents the best consensus concerning the current state of practice.

Assessment of Performance Monitoring at Diablo Canyon Unit 1

As discussed in Section 2, the concerns expressed in [SECY-22-0019] and in [NRC-2022] about "*insufficient performance monitoring*" relate to plants with low lead factor capsules who have delayed their last capsule withdraw to the point that the highest fluence at which they have measured ΔT_{41J} data from surveillance no longer exceeds the current peak fluence on the inner diameter of the RPV wall. As was illustrated in Figure 2, some plants have been in this situation for decades. This is not the condition of Diablo Canyon Unit 1, where the maximum fluence from the surveillance data collected as part of the Unit 1 surveillance program exceeds the 40-year RPV fluence. When combined with similar data on weld wire heat 27204 from the Palisades "sister plant," the Diablo Canyon Unit 1 data set extends to a RPV fluence well beyond 60 operating years. Based on the comments made by NRC in [SECY-22-0019] and in [NRC-2022] Diablo Canyon Unit 1 cannot be construed to have a surveillance capsule program that provides "*insufficient performance monitoring*." The Diablo Canyon Unit 1 RPV fluence has been, and continues to be, well bounded by already measured surveillance data, demonstrating the sufficiency of Diablo Canyon Unit 1's surveillance program.

The NRC concerns expressed in [SECY-22-0019] and in [NRC-2022] about "*insufficient performance monitoring*" made no mention of ultrasonic examination. Prior to 2014 PG&E made the required in-service inspections of the Diablo Canyon Unit 1 RPV on a 10-year interval. In 2014 PG&E made a plant-specific request for an interval extension to 20-years [DCL-14-074], which was granted by the NRC. Thus, the UT inspections of the Diablo Canyon Unit 1 RPV are sufficient.

5. Summary and Implications for Diablo Canyon Unit 1

In [NRC 2022] and in [SECY-22-0019] the NRC staff indicated that the NRC's surveillance monitoring and embrittlement prediction requirements may need to be updated to address the current state of knowledge. Specifically, the staff expressed concerns that the embrittlement predictions of [RG1.99R2] may under-estimate embrittlement at high fluences and that the surveillance requirements of [10CFR50-AppH] may allow too long of an interval between surveillance capsule withdrawals for some plants licensed for operation beyond 40 years. The information summarized in this document demonstrates that these concerns are not pertinent to Diablo Canyon Unit 1. The limiting weld in the Diablo Canyon Unit 1 RPV is well predicted by the NRC's [RG1.99R2] embrittlement trend curve and the maximum fluence that the Unit 1 RPV will reach after 60-years is only 2×10^{19} n/cm^2, which is far below the fluence at which the NRC's [RG1.99R2] formula begins to under-predict embrittlement. Additionally, while there have been deferrals to the removal of Capsule B from the Diablo Canyon Unit 1 RPV these deferrals have not compromised the sufficiency of the performance monitoring program. Surveillance data for the limiting weld in Diablo Canyon Unit 1 is already available to a fluence beyond that the RPV will experience after 60-years of operation.

6. Citations

10CFR50.55a	Codes and Standards, https://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-0055a.html
10CFR50-AppH	Reactor Vessel Material Surveillance Program Requirements, https://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-apph.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.
ASTM E23	ASTM E23-23, Standard Test Methods for Notched Bar Impact Testing of Metallic Materials, https://www.astm.org/e0023-23a.html.
ASTM E185-82	Standard Practice for Design of Surveillance Programs for Light-Water Moderated Nuclear Power Reactor Vessels. The current version, E185-21, appears here: https://www.astm.org/e0185-21.html
ASTM E900-15	Standard Guide for Predicting Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials, the current version appears here: https://www.astm.org/e0900-21.html
ASTM Adjunct	"Adjunct for ASTM E900-15: Technical Basis for the Equation used to Predict Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials," A Product of ASTM Subcommittee E10.02, on the Behavior and Use of Nuclear Structural Materials, Work Package WK34875, 18 September 2015, https://www.astm.org/adje090015-ea.html.
DCL-14-074	PG&E Letter DCL-14-074 to US Nuclear Regulatory Commission, "ASME Section XI Inspection Program Request for Alternative RPV-U1-Extension to Allow Use of Alternate Reactor Inspection Interval," 18 August 2014, NRC ADAMS No. ML14230A618.
JEAC4201	Japan Electric Association Code, "Method of Surveillance Tests for Structural Materials of Nuclear Reactors," JEAC 4201 -2007, (2013 addenda)
Kirk 2018	M. Kirk, R. Hardies, and M. Gordon, "A Historical Review of the Development of ASTM Standard Practices E185 and E2215 from the 1960s to 2016," International Review of Nuclear Reactor Pressure Vessel Surveillance Programs, ASTM STP1603, W. L. Server and M. Brumovsky, Eds., ASTM International, West Conshohocken, PA, 2018, https://doi.org/10.1520/STP1603-EB.
NRC 2022	"Impacts of Embrittlement on Reactor Pressure Vessel Integrity from a Risk Informed Perspective," NRC ADAMS ML21314A228, March 8, 2022, https://www.nrc.gov/docs/ML2131/ML21314A228.pdf.

NUREG-0800	Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition, https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0800/index.html.
NUREG-1800	Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1800/index.html.
NUREG-1801	Generic Aging Lessons Learned (GALL) Report, NUREG-1801, Revision 2, Nuclear Regulatory Commission, December 2010, https://www.nrc.gov/docs/ML1034/ML103490041.pdf
NUREG-1806	M. EricksonKirk, M. Junge, W. Arcieri, B.R. Bass, R. Beaton, D. Bessette, T.H.J. Chang, T. Dickson, C.D. Fletcher, A. Kolaczkowski, S. Malik, T. Mintz, C. Pugh, F. Simonen, N. Siu, D. Whitehead, P. Williams, R. Woods, and S. Yin, "Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61)," United States Nuclear Regulatory Commission, NUREG-1806, August 2007, https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1806/index.html
NUREG-2191V2	Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report, https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr2191/index.html.
Part 1 Rpt	Kirk report to DCISC, Part 1. https://www.dcisc.org/download/library/an-evaluation-of-the-status-of-diablo-canyon-unit-1/report-part-1-addressing-public-concerns-75.pdf
Part 2 Rpt	Kirk report to DCISC, Part 2. https://www.dcisc.org/download/library/an-evaluation-of-the-status-of-diablo-canyon-unit-1/report-part-2-evaluation-of-diablo-canyon-unit-1-embrittlement-76.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.
RG 1.174	Regulatory Guide 1.174, Revision 3, "An approach for using probabilistic risk assessment in risk-informed decisions on plant-specific changes to the licensing basis," January 2018, https://www.nrc.gov/docs/ML1731/ML17317A256.pdf.
SECY-22-0019	Letter to The Commissioners from Daniel H. Dorman, "Rulemaking Plan for the Revision of Embrittlement and Surveillance Requirements for High-Fluence Nuclear Power Plants in Long-Term Operation," SECY-22,0019, 8 March 2022, NRC ADAMS ML21314A215, https://www.nrc.gov/docs/ML2131/ML21314A194.html.

Soneda 2013	Soneda, N., Nakashima, K., Nishida, K., and Dohi, K., “Modification of Embrittlement Correlation Method of Reactor Pressure Vessel Steels – Calibration to the Surveillance Test Data at High Fluences,” CRIEPI Report Q12007, March 2013 (in Japanese), http://criepi.denken.or.jp/jp/kenkikaku/report/detail/Q12007.html.
WCAP-16168	B. Bishop, C. Boggess, and N. Palm, “Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval,” Westinghouse Electric Company, WCAP-16168-NP-A Revision 3, October 2011.
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.