Impacts of Embrittlement on Reactor Pressure Vessel Integrity from a Risk-Informed Perspective

Final Report

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Executive Summary

U.S. Nuclear Regulatory Commission (NRC) regulations on reactor pressure vessel (RPV) integrity, together with the associated codes and standards, are designed to function synergistically to provide reasonable assurance that RPV integrity will be maintained over the operating lifetime of each plant. Within these regulations, the material toughness predicted by the embrittlement trend curve (ETC) model[1] of Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," issued May 1988, is used to demonstrate that margin to prevent brittle fracture of the RPV[2] is maintained both in normal operation, as defined by Appendix G, "Fracture Toughness Requirements," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," and during pressurized thermal shock (PTS) events, as defined by 10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events." In conjunction, the regulations contain requirements for performance monitoring through surveillance programs to demonstrate that the generic ETC model predictions adequately describe the properties of critical plant-specific RPV materials over the entire reactor operating lifetime.

However, the existing RG 1.99 ETC model, which was developed in the mid-1980s, has characteristics that manifest as underprediction of RPV material neutron embrittlement under the high fluences that would be reached at multiple pressurized-water reactor (PWR) plants when operated beyond 60 years. Furthermore, the amount of the underprediction increases with increasing fluence. In parallel, licensees are allowed to defer, and many have deferred, surveillance capsule testing that is intended to confirm the embrittlement predictions from the ETC model. This report documents a holistic, risk-informed evaluation of RPV integrity that adheres to the principles of RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," to assess the coupled impacts of the RG 1.99 ETC underprediction of RPV material neutron embrittlement at high fluences and the trend of decreasing performance monitoring.

A prior assessment of the RG 1.99 ETC, documented in the technical letter report TLR-ES/DE/CIB-2019-2, "Assessment of the Continued Adequacy of Revision 2 of Regulatory Guide 1.99," dated July 31, 2019, first verified and quantified the general tendency of the ETC to increasingly underpredict fracture toughness as fluence increases, starting at a fluence of approximately 3×10^{19} neutrons per square centimeter (n/cm^2) and becoming statistically significant at $6\text{x}10^{19}$ n/cm^2. (Sixty percent of currently operating PWRs are projected to surpass 3×10^{19} n/cm^2 within 80 years of operation, while 25 percent are projected to surpass 6×10^{19} n/cm^2 within 80 years of operation.[3]) For the evaluation documented in this report, the NRC staff determined that the ETC model of American Society for Testing and Materials (currently known as ASTM International) (ASTM) E900-15, "Standard Guide for Predicting Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials," provided

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1 This ETC is also found in 10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events."

2 Margin hereinafter means "adequate margin," i.e., margin that conforms to RG 1.99, Revision 2.

3 Fifty percent of PWRs in the United States are projected to surpass 6×10^{19} n/cm^2 within 100 years of operation.

sufficiently accurate predictions of existing surveillance capsule data, particularly at high fluences, such that it could be used to assess the safety implications of the RG 1.99 ETC underpredictions. The staff used a targeted sample of approximately 200 individual materials from 21 plants to assess these potential safety implications. The sample focused on high-fluence plants, with some plants added to represent other critical material characteristics.

To evaluate the risk significance of the RG 1.99 embrittlement underpredictions in relation to a PTS event, the staff used methods and analyses consistent with those that supported the development of the alternative PTS rule in 10 CFR 50.61a, "Alternate fracture toughness requirements for protection against pressurized thermal shock events." The staff projected licensing-basis fluences for the targeted sample plants to 80 years of operation and estimated the through-wall crack frequency (TWCF) for each plant's limiting material using available plant-specific information. The estimated TWCF was found to be more than one order of magnitude below the criterion of 1×10^{-6}/year (yr) for all plants in the targeted sample.

The staff performed a scoping study to quantitatively assess the risks associated with using the RG 1.99 ETC to determine normal operating conditions. The probability of RPV failure was estimated as a function of the amount of underprediction by the RG 1.99 ETC for two separate postulated flaws and transients associated with leak testing, cooldown operations following the pressure-temperature limit curve, and actual plant cooldown transients. The scoping study demonstrated that, for embrittlement shift values less than the maximum embrittlement underprediction for the targeted sample plants, the expected TWCF for each transient studied was below 1×10^{-6}/yr for 80 years of operation. However, there is significant uncertainty in extending these generic findings to individual plants.

Because the risk calculations contained large uncertainties, the staff also assessed the impact of these issues on the safety margins for normal operating conditions and the adequacy of performance monitoring requirements. The staff concluded that, compared to an ETC giving accurate embrittlement predictions, the RG 1.99 ETC may produce pressure-temperature limits that are less conservative for normal operating conditions and may provide a reduction in margin to brittle fracture due to the underprediction of embrittlement at high fluence. In addition, if performance monitoring is not conducted as intended throughout periods of extended operation, the uncertainties in the analyses are amplified. In long-term operation, these large analysis and monitoring uncertainties may further erode the safety margins that are inherent in the requirements of Appendix G to 10 CFR Part 50.

1. Introduction

U.S. Nuclear Regulatory Commission (NRC) regulations on reactor pressure vessel (RPV) integrity of existing and new light-water reactors, together with the associated codes and standards, are designed to function synergistically to provide reasonable assurance that RPV integrity will be maintained over the operating lifetime of each plant. The regulations encompass the RPV lifecycle, addressing fabrication, preservice inspection and testing, inservice inspection and testing, monitoring of material property changes during operation, and changes to operational requirements based on these material property changes. The current regulatory framework, established over 40 years ago, was intended to be conservative to compensate for existing uncertainties. Over time, as knowledge of the factors governing RPV integrity has evolved, understanding of the nature and significance of many of the conservatisms associated with the regulatory framework has also improved. This foundational knowledge has helped in assessing issues that have challenged RPV integrity during this time.

While other degradation factors may impact RPV integrity, this paper focuses on time-dependent degradation of RPV material properties due to neutron radiation. This neutron damage, i.e., embrittlement, increases the ductile-to-brittle transition temperature and thus reduces the fracture toughness of the RPV material. The NRC regulatory framework addresses such degradation through (1) the use of an embrittlement trend curve (ETC) to predict the level of embrittlement, as described in Regulatory Guide (RG) 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials,” issued May 1988 [Ref. 1], and (2) monitoring of plant-specific embrittlement through an RPV material surveillance program in accordance with Appendix H, “Reactor Vessel Material Surveillance Program Requirements,” to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, “Domestic Licensing of Production and Utilization Facilities.” Recently, the NRC staff has been assessing the safety significance of underpredictions of material fracture toughness calculated according to the RG 1.99 ETC. The material toughness predicted by the ETC is used to determine pressure-temperature (P-T) operational limits and demonstrate that adequate margin exists to protect against pressurized thermal shock (PTS) events.[1]

In parallel, the staff has been assessing the requirements in Appendix H to 10 CFR Part 50 in relation to surveillance needs for plant operation beyond 60 years. Some licensees are planning to test a single additional surveillance capsule at a fluence associated with the end of their proposed 80-year license, regardless of the time elapsed since their most recent surveillance testing, and regardless of the difference between the fluence level of the most recently tested capsule and the current vessel fluence. The current framework does not require testing of this last capsule, leading to a lack of surveillance monitoring over the entire operation of the plant, especially at high fluences.

Consistent with precedent and practice, the staff has demonstrated (see Appendix A for details) that these two issues individually have low generic risk significance. However, these two issues are coupled and perform different, yet supporting, functions within the regulatory framework. In

[1] Reference 11 discusses the impacts of embrittlement on the upper-shelf energy; it finds them to be minimal, because of the conservative nature of the models.

particular, plant monitoring through surveillance testing provides confidence that the generic material embrittlement trends predicted by the RG 1.99 ETC are valid for the plant-specific materials of interest. Therefore, to fully understand the potential safety impact of these two issues, it is necessary to consider their effects jointly.

This risk-informed holistic evaluation of RPV integrity in light-water reactors assesses the impact of potential RG 1.99 ETC underpredictions and decreasing performance monitoring (e.g., RPV material surveillance) taking into consideration the five principles of risk-informed decisionmaking found in RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" [Ref. 2]. Because risk-informed evaluations have demonstrated that the generic risk of each issue individually is not significant, the expectation is that significant risk can result only from a plant-specific confluence of decreased performance monitoring (e.g., lack of surveillance capsule testing at the high fluence levels typical of the end-of-license conditions) and characteristics that elevate the underpredictions of the RG 1.99 ETC, especially at high fluences. The evaluation considers representative plant-specific combinations of fluence, material properties, and surveillance capsule withdrawal schedules over a presumed 80 years of operation. This evaluation also assumes implicitly that future plant operation and capacity factors remain consistent with current industry practice, so that fluence values can be projected to the end of the plant's operating life.

Section 2 of this report describes the philosophical underpinnings of RPV integrity assessments and notes the applicable regulatory requirements. Section 3 summarizes the staff's quantitative assessment of the risk significance of RG 1.99 ETC underpredictions and industry activities related to ETC development and surveillance programs. It also discusses the uncertainties associated with both the staff's RG 1.99 ETC evaluation and the extension of surveillance withdraw periods. Section 4 addresses the adequacy of plant-specific surveillance programs in the periods of extended operation (PEOs) to 60 years and subsequently to 80 years. Section 5 evaluates how the potential RG 1.99 ETC underpredictions coupled with decreased surveillance testing jointly affect safety margins. Section 6 provides an evaluation of the situation according to the five principles of risk-informed decisionmaking in RG 1.174. Finally, Section 7 summarizes the staff's analyses.

2. Background

Preventing catastrophic RPV failure has long been a cornerstone of nuclear reactor research and regulation [Ref. 3], as such a failure exceeds the design requirements for engineered core cooling systems. Therefore, there is a mosaic of regulations focused on preventing RPV failure, including several relevant general design criteria in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 [Ref. 4]. The regulations most pertinent to this evaluation, however, are Appendix G, "Fracture Toughness Requirements," to 10 CFR Part 50 [Ref. 5]; 10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events" [Ref. 6]; and Appendix H to 10 CFR Part 50 [Ref. 7]. Appendix G to 10 CFR Part 50 specifies fracture toughness requirements for the RPV, including the RPV material's minimum fracture toughness on the upper shelf (i.e., the temperature regime where failure occurs in a ductile manner), minimum operating temperature requirements, and P-T limits that apply over the RPV's operating life. The P-T limits, in particular, are intended to maintain adequate margins during normal operating conditions with continued embrittlement of the RPV; these limits must therefore be adjusted to higher temperatures as the plant ages. The requirements in 10 CFR 50.61 and the voluntary alternative of 10 CFR 50.61a, "Alternate fracture toughness requirements for protection against pressurized thermal shock events" [8], provides requirements applicable to pressurized-water reactor (PWR) licensees to demonstrate that the RPV's material toughness remains acceptable over the operating period to guard against PTS events. Appendix H to 10 CFR Part 50 provides the surveillance program requirements; for practical implementation, it cites several editions of the American Society for Testing and Materials (ASTM; currently known as ASTM International) standard ASTM E185, all from 1982 or earlier [Ref. 9]. Appendix A gives more information on these regulations and their interrelationship.

As indicated in Section 1, radiation embrittlement of the RPV is a significant aging concern,[2] and ETC models are used to assess the effects of such embrittlement on the RPV's fracture toughness. Activities have been ongoing to make ETC models more accurate and ensure that they adequately represent the critical RPV materials as embrittlement continues to increase beyond 80 years of operation. The plant-specific validation of embrittlement trends provided by surveillance programs was originally intended to provide data for 40 years of operation. Consequently, recent activities have focused on obtaining information for up to 80 years of operation. Appendix A gives a more detailed summary of recent and ongoing industry activities for improving ETC models and extending surveillance programs.

The underpredictions of the RG 1.99 ETC model and the decreased availability of fracture toughness data from the surveillance programs required by Appendix H to 10 CFR Part 50 are the latest of numerous issues observed domestically and internationally in the last 10 years that have raised questions about their effects on RPV integrity. These issues include the previously unanalyzed risks associated with small surface-breaking flaws (SSBFs), quasilaminar hydrogen cracking, nonconservatisms in Branch Technical Position 5-3, and carbon macrosegregation. The NRC staff has assessed the generic risk associated with each of these issues individually, consistent with RG 1.174 principles. However, it has not considered possible combined or synergistic effects due to potential interactions among these issues. Such interactions are not expected to significantly alter the generic risks or the conclusions from the individual evaluations, but they may have a significant impact on specific plants. Section 3.6 and Appendix A give more details on these prior analyses and the associated uncertainties.

3. Staff Evaluation of Implications of Regulatory Guide 1.99 Underpredictions

As a followup to an NRC periodic review of the adequacy of RG 1.99 [Ref. 10], the staff performed a comprehensive review to evaluate the continued adequacy of the RG for the operating fleet and new light-water reactor builds. This review found potential safety-significant issues in the prediction of embrittlement at high fluences (such as those experienced in license renewal PEOs), the potential rejection of credible surveillance data, and continued reliance on the ETC model trend prediction even when surveillance data indicate a different trend [Ref. 11].

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2 Other degradation, such as thermal embrittlement and stress corrosion cracking, may also impact RPV integrity, but this paper is focused on the impact of radiation embrittlement.

In addition, the steel specifications for some small modular reactors now being considered have operational and compositional conditions (in particular operating temperatures) lying at the edge of, or beyond, those used in the development of RG 1.99. These technical observations led to the staff evaluation documented in this section.

3.1. Regulatory Guide 1.99 Embrittlement Underpredictions at High Fluence

As described in Section 2 and Appendix A, RG 1.99 describes methods that may be used to predict the effects of radiation embrittlement of RPVs. Specifically, neutron irradiation of the RPV steel results in material property changes, making the steel more brittle (e.g., increasing in the ductile-to-brittle transition temperature) and potentially susceptible to rapid failure under high stress. As described in Reference 11, the most recent version of RG 1.99 was published in 1988 and was expected to be updated and refined as more data became available. The evaluation documented in Reference 11 assessed all aspects of RG 1.99, including the analysis methodology for predicting embrittlement behavior in RPV steels based on the results from testing of surveillance capsules to measure the transition temperature shift at 41 joules (30 foot-pounds), or ΔT_{41J}.

In Reference 11, the RG 1.99 ΔT_{41J} ETC was assessed using the BASELINE dataset recently developed by ASTM, as described in Appendix A. The predicted embrittlement shift for the surveillance materials can be compared directly to the measured embrittlement shift as shown in Figure ***1***. In this figure, the abscissa (X-axis) is the specimen fluence, and the ordinate (Y-axis) is the difference between the predicted and measured embrittlement shift. An ordinate value of zero indicates a perfect prediction of embrittlement behavior. The gray symbols that are in Figure 1 represent surveillance data from international reactors and the red symbols that are represent surveillance data from the U.S. only. Predictions that are too high (conservative) may cause undue plant burden by unnecessarily narrowing the operating window of P-T limits or increasing the required hydrostatic leak testing temperature. More importantly, predictions that are too low (nonconservative) may lead to operation below the safety margins required in Appendix G to 10 CFR Part 50 and in 10 CFR 50.61 and 10 CFR 50.61a.

The estimates provided by RG 1.99 appear to underpredict embrittlement at fluence levels approaching 3×10^{19} neutrons per square centimeter (n/cm^2) to 6×10^{19} n/cm^2 (E > 1 megaelectron volt). For base metals, this is evident from the U.S. data and corroborated by the international data. However, no conclusion can be drawn for weld metals because the data are too sparse.[3] Also, a significant proportion of both U.S. and international data (approximately 19 percent) fall outside of the two-sigma standard deviation bounds shown in Figure ***1***. This result indicates that the prescribed standard deviation in RG 1.99 is smaller than the standard deviation in the ASTM BASELINE dataset (i.e., the ASTM BASELINE standard deviation is about 20 percent larger than that of RG 1.99). Consequently, the RG 1.99 ETC provides a less accurate prediction than the guidance indicates.

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3 The selection of limiting material with regard to radiation damage is plant dependent. Across the U.S. fleet, there is an approximately equal distribution of weld and base limiting materials. Of the 21 plants in the targeted sample described in Section 3.3, 12 were weld-limited and 9 were base-metal-limited.

Another point of interest is that greater uncertainty is inherent in the predictions as the data become sparse at fluences greater than 3×10^{19} n/cm^2 to 4×10^{19} n/cm^2. As mentioned in Appendix A, EPRI's PWR Supplemental Surveillance Program (PSSP) aims to develop data at these high fluence levels to better understand the uncertainty in the embrittlement trend predictions. The 27 additional data points from the PSSP are expected to be available after 2028.

The potential for underpredicting ΔT41J may affect the safe operation of plants. As fluence increases, the potential to underpredict ΔT41J also increases. To provide some context, 60 percent of the current operating reactors are projected to surpass 3×10^{19} n/cm^2 within 80 years of operation while 25 percent are projected to surpass 6×10^{19} n/cm^2 within 80 years of operation (Figure 2-6 of [Ref. 11]).

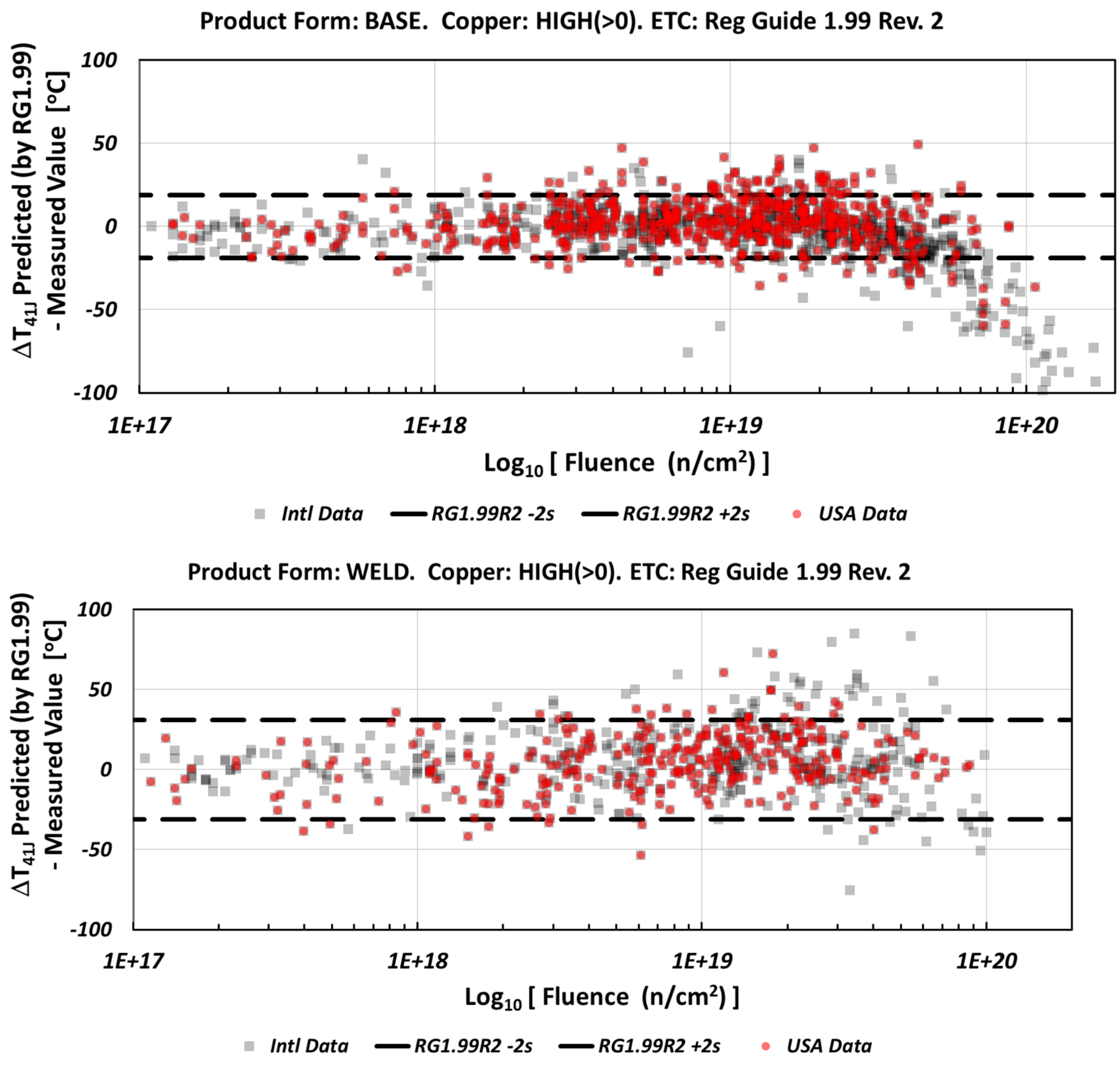


Figure 1 Embrittlement predictions using RG 1.99: top, ΔT_{41J} for base metals; bottom, ΔT_{41J} for weld metals; two standard deviations plotted from RG 1.99 values

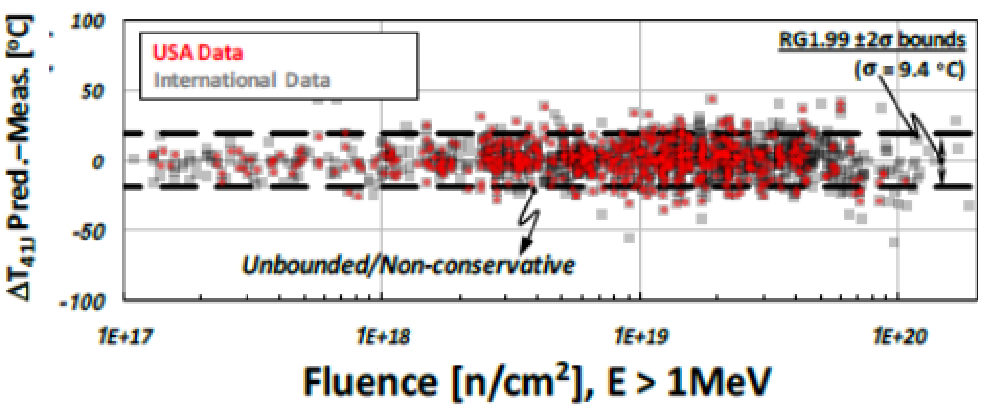
3.2. Embrittlement Trend Curve Assessment

Recognizing that the RG 1.99 ETC underpredicts the surveillance data at high fluences (Section 3.1), the NRC staff performed a statistical assessment of the accuracy of the RG 1.99 ETC and other ETC models using the most recent BASELINE ASTM E900 dataset discussed in Appendix A [Ref. 12]. The staff determined that the ETC from the 2015 version of ASTM E900, “Standard Guide for Predicting Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials” (the ASTM E900-15 ETC), provided the most accurate characterization of this database. Specifically, ASTM E900-15:

- Produced more accurate predictions of surveillance data at high fluence ($> 3\times10^{19}$ n/cm^2) than other similar ETCs.
- Performed better than other similar ETCs with respect to *t*-test results for all material inputs.

The improved accuracy of the ASTM E900-15 ETC results from the use of a larger dataset, including U.S. surveillance data from between 2004 and 2012, that other ETCs do not incorporate. The ASTM E900-15 ETC is expected to predict embrittlement more accurately in a broader band of temperatures than other ETCs.

Figure 2 shows the results in Figure ***1*** recreated using the ASTM E900-15 ETC. From this figure it is clear that the ASTM E900-15 ETC predictions for these materials are more accurate than those of the RG 1.99 ETC.



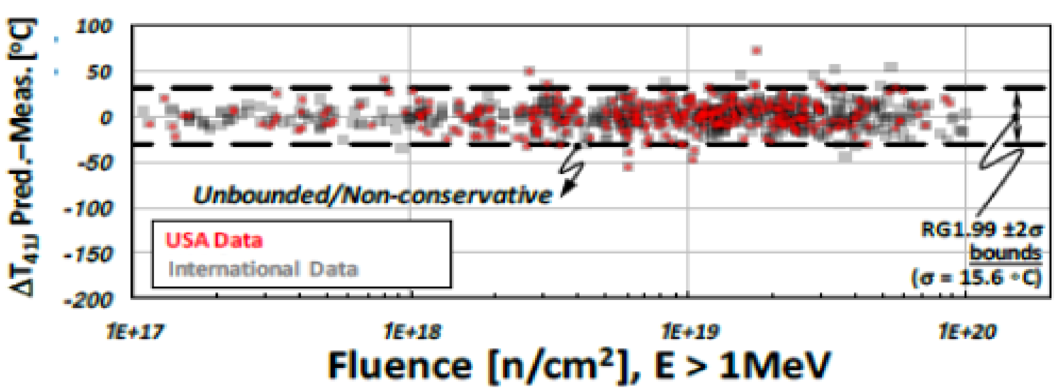


Figure 2 Embrittlement predictions using ASTM E900-15: top, ΔT_{41J} for base metals; bottom, ΔT_{41J} for weld metals; two standard deviations plotted from RG 1.99 values

Because of these results, the staff used the ASTM E900-15 ETC model to represent existing surveillance information in its evaluations of the plant-specific implications of continued use of the RG 1.99 ETC, which are discussed in subsequent sections.

3.3. Plant Selection for the Regulatory Guide 1.99 Targeted Sample

The NRC staff recognized that use of an alternative ETC to correct the underpredictions described in Section 3.1 may affect the operating fleet by resulting in an increased adjusted reference temperature (ART), which is used to calculate P-T limits in accordance with Appendix G to 10 CFR Part 50. An increase in ART shifts the P-T limits and decreases the allowable operating window for heatup and cooldown transients. To understand the potential changes, the staff performed a fleet impact study on a targeted sample of 21 reactors to determine the amount by which correction of the underpredictions would change the licensing-basis ART and RT_{PTS} values (i.e., the reference temperature calculated as required in 10 CFR 50.61 or 10 CFR 50.61a). As described in the previous section, the ASTM E900-15 ETC has been shown to predict the embrittlement behavior of RPV steels more accurately for fluence levels up to 1×10^{20} n/cm^2; therefore, this was the model used in this study. Correspondingly, the staff defined the "embrittlement shift delta" (ESD) as the difference in ART between the RG 1.99 ETC and the ASTM E900-15 ETC. From the number and magnitude of the ESD values, the staff determined qualitatively whether the use of the alternative ETC would increase or decrease burden.

The targeted sample for the fleet impact study comprised 21 reactors and approximately 200 individual materials. It included mostly plants having relatively high projected end-of-license peak neutron fluences (mainly older PWRs), with a few plants representing other data subsets, such as boiling-water reactors (BWRs) and low-copper materials, for completeness. The staff confirmed that the sample spanned the full copper and nickel chemistry range of the operating fleet. Reference 12 gives the details of the plants chosen.

Figure 3 shows the distribution of ESDs for the targeted sample plants as a function of neutron fluence for both the RPV inner diameter (ID) and the quarter-thickness (1/4T) locations.[4] There is a visible trend toward higher ESDs as fluence increases. The ID location tends to have higher ESDs, which is not surprising since neutron fluences are higher at the ID. The maximum ESD is around 120 degrees Fahrenheit (F) on the ID and 100 degrees F at the 1/4T location.

4 These locations were chosen to correspond to the Appendix G and 10 CFR 50.61a analysis locations.

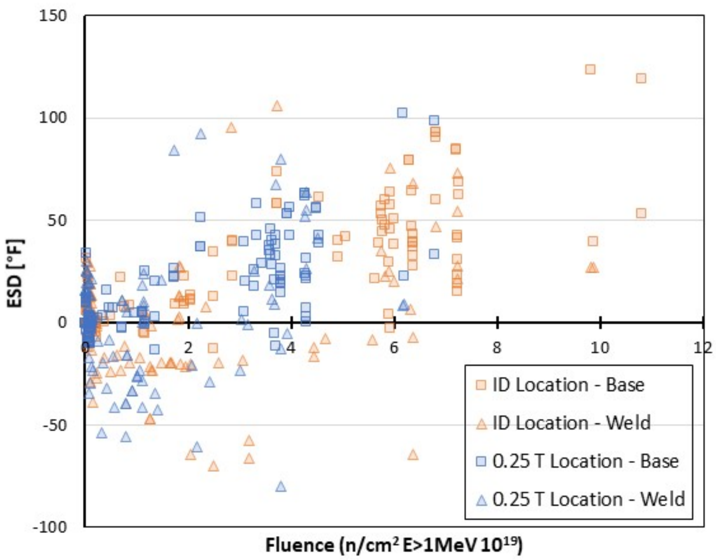
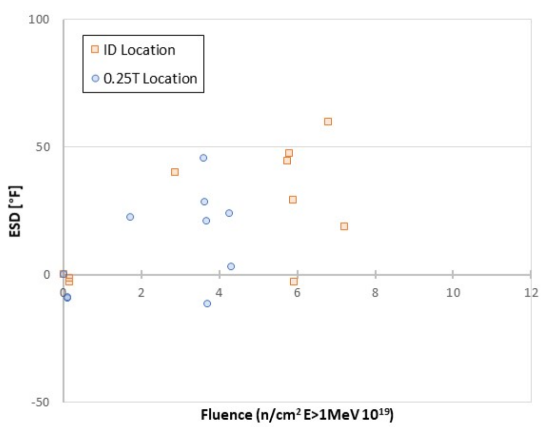
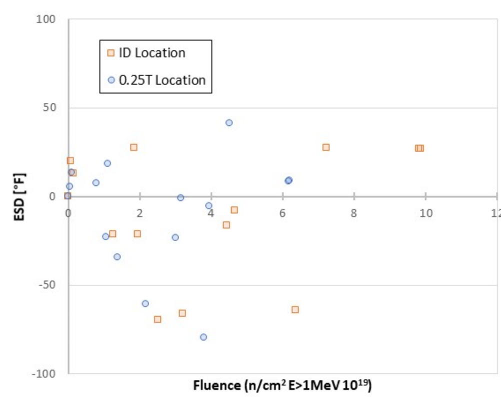


Figure 3 Distribution of ESD versus fluence for all materials in targeted sample

Figure 4 shows the distribution of ESDs for only the limiting materials (i.e., those with the highest ART or RT_{PTS} for a given reactor at the 1/4T location or the ID, respectively). For the base materials there is a similar trend of ESD increasing with increasing fluence, while for the weld materials, there appears to be little trend with fluence. The maximum ESD is about 60 degrees F for the base materials and about 40 degrees F for the weld materials. Use of the ASTM E900-15 ETC changed which material was limiting for 20 percent of the plants in the targeted sample, but this did not affect the conclusions on the trends of ESD with fluence.



(a)



(b)

Figure 4 Distribution of ESDs versus fluence for limiting materials only: (a) base, (b)weld

The results of the fleet impact study show the following:

- There is a tendency for the limiting material reference temperatures to increase, particularly for base metals. The trend is not evident for welds.
- Reference temperatures tend to increase more at the ID location than at the 1/4T location.
- Many weld materials see reductions in reference temperature at fluences below 4×10^{19} n/cm^2.
- Only a few plant limiting materials may have increases in reference temperatures of over 50 degrees F, mainly for base metals at fluences of 6×10^{19} n/cm^2 or greater.

3.4. Pressurized Thermal Shock Evaluation Summary

To assess how a more accurate ETC would affect PTS evaluations for the targeted sample plants, the staff used predictions of through-wall crack frequency (TWCF) due to PTS events. Using the methodology developed in the technical basis for 10 CFR 50.61a [Ref. 13], the staff conducted a series of probabilistic fracture mechanics analyses to develop a relationship between the maximum RT_{NDT} (RT_{max}) (for axial welds (AW), circumferential welds (CW), forgings (FO), and plates (PL)) and the 95th-percentile TWCF ($TWCF_{95\text{-total}}$). (The 10 CFR 50.61a rule uses the 95th-percentile TWCF as the acceptance criterion in order to produce conservative RT_{max} screening limits.) The relationship is as follows:

$$TWCF_{95-TOTAL} = [\alpha_{AW} TWCF_{95-AW} + \alpha_{PL} TWCF_{95-PL} + \alpha_{CW} TWCF_{95-CW} + \alpha_{FO} TWCF_{95-FO}],$$

where

$$TWCF_{95-AW} = \exp\{5.5198 \ln(RT_{MAX-AW} - 616) - 40.542\}\beta,$$

$$TWCF_{95-PL} = \exp\{23.737 \ln(RT_{MAX-PL} - 300) - 162.38\}\beta,$$

$$TWCF_{95-CW} = \exp\{9.1363 \ln(RT_{MAX-CW} - 616) - 65.066\}\beta,$$

$$TWCF_{95-FO} = \exp\{23.737 \ln(RT_{MAX-FO} - 300) - 162.38\}\beta + \eta\{1.3 \times 10^{-137} 10^{0.185 RT_{MAX-FO}}\}\beta,$$

and the values of α, β, and η are given in Table 1 (taken from Reference 13).

Table 1 PTS Parameter Definitions

Factor	Condition	Equation
Stuck-Open Valves α	$RT_{MAX\text{-}xx} \leq 625R$	$\alpha_{xx} = 2.5$
	$625R < RT_{MAX\text{-}xx} < 875R$	$\alpha_{xx} = 2.5 - \frac{1.5}{250}(RT_{MAX\text{-}xx} - 625)$
	$RT_{MAX\text{-}xx} \geq 875R$	$\alpha_{xx} = 1$
Vessel Thickness β	$T_{WALL} \leq 9\ ½$ -in	$\beta = 1$
	$9\ ½ < T_{WALL} < 11\ ½$ -in	$\beta = 1 + 8 \cdot (T_{WALL} - 9\ ½)$
	$T_{WALL} \geq 11\ ½$ -in	$\beta = 17$
Sub-Clad Cracks η	Forging is compliant with Regulatory Guide 1.43	$\eta = 0$
	Forging not compliant with Regulatory Guide 1.43	$\eta = 1$

After projecting the targeted sample licensing-basis fluences to 72 effective full-power years (EFPYs) and using licensing-basis chemistry information to calculate the RT_{MAX} values, the staff used the equations shown above to predict the TWCF for each material. For this evaluation, the RT_{MAX} calculations used three ETCs: RG 1.99, ASTM E900-15, and EONY[5] [Ref. 14]. The maximum value of RT_{MAX} for each product form was used in the above equations and the results are shown in Table 2. From this table, for all materials in the targeted sample, the maximum 95th-percentile TWCF was well below the acceptance criterion of 1×10^{-6}/year (yr).

5 The EONY ETC is shown here for reference, since it is the ETC used in 10 CFR 50.61a.

Table 2 PTS Evaluation Results*

Unit	Total $TWCF_{95\text{-}total}$ at 72 EFPYs		
	RG 1.99 RT_{MAX}	ASTM E900 RT_{MAX}	EONY RT_{MAX}
A	$<1\times10^{-10}$	$<1\times10^{-10}$	6.3×10^{-8}
B	3.7×10^{-7}	1.1×10^{-9}	2.6×10^{-8}
C	4.6×10^{-10}	1.6×10^{-9}	6.4×10^{-9}
D	$<1\times10^{-10}$	4.2×10^{-10}	2.4×10^{-9}
E	$<1\times10^{-10}$	2.9×10^{-10}	1.5×10^{-9}
F	$<1\times10^{-10}$	$<1\times10^{-10}$	2.7×10^{-10}
G	2.0×10^{-10}	3.0×10^{-10}	1.4×10^{-10}
H	6.9×10^{-9}	$<1\times10^{-10}$	1.2×10^{-10}
I	$<1\times10^{-10}$	$<1\times10^{-10}$	1.0×10^{-10}
J	6.8×10^{-10}	1.2×10^{-10}	$<1\times10^{-10}$
K	$<1\times10^{-10}$	$<1\times10^{-10}$	$<1\times10^{-10}$
L	$<1\times10^{-10}$	$<1\times10^{-10}$	$<1\times10^{-10}$
M	$<1\times10^{-10}$	$<1\times10^{-10}$	$<1\times10^{-10}$
N	$<1\times10^{-10}$	$<1\times10^{-10}$	$<1\times10^{-10}$
O	$<1\times10^{-10}$	$<1\times10^{-10}$	$<1\times10^{-10}$
P	$<1\times10^{-10}$	$<1\times10^{-10}$	$<1\times10^{-10}$
Q *	$<1\times10^{-10}$	$<1\times10^{-10}$	$<1\times10^{-10}$
R *	$<1\times10^{-10}$	$<1\times10^{-10}$	$<1\times10^{-10}$
S *	$<1\times10^{-10}$	$<1\times10^{-10}$	$<1\times10^{-10}$
T *	$<1\times10^{-10}$	$<1\times10^{-10}$	$<1\times10^{-10}$
U	$<1\times10^{-10}$	$<1\times10^{-10}$	$<1\times10^{-10}$

* Shaded rows correspond to BWR plants.

3.5. Probabilistic Fracture Mechanics Scoping Study

The staff used Version 16.1 of the Fracture Analysis of Vessels, Oak Ridge (FAVOR), code [Ref. 15, 16] to perform a quantitative assessment of the RPV failure risks associated with a set of normal operating events. It first computed risks while retaining the RG 1.99 ETC to determine the normal-operation PWR and BWR P-T limits and leak test curves for operation to 80 years, as described in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” Appendix G, Paragraph G-2215. After establishing these baseline conditions, the staff assessed the effect of potential RG 1.99 ETC underpredictions by computing the probability of crack initiation and RPV failure for various transients as a function of the ESD, assuming (consistent with all other analyses) that the ASTM E900-15 ETC most accurately estimates RPV embrittlement after 80 years of operation.

The probability of RPV failure was assessed for two flaw types: (1) a 1/4T ID surface flaw with a surface crack length-to-depth ratio of 6 to 1, and (2) the SSBF whose crack tip penetrated through the stainless steel cladding into the ferritic RPV metal (i.e., 0.03T PWR flaw depth or

0.04T BWR flaw depth), with various surface crack length-to-depth ratios. For each combination of reactor type, flaw type, and ESD value, the following transients were studied:

- BWR and PWR cooldown following the operational P-T limit curve (using a uniform cooldown rate of either 100 degrees F/hour (hr) or 50 degrees F/hr)
- BWR plant cooldown following the saturation curve
- BWR plant performing leak test following P-T limit curves (using a uniform cooldown rate of either 40 degrees F/hr or 100 degrees F/hr at the end of the leak test)
- PWR plant following cooldown curves for 42 actual plant cooldowns and leak tests

For each scenario, the staff used FAVOR to calculate the conditional probability of crack initiation (CPI) and the conditional probability of through-wall crack failure (CPF). In particular, CPF was used as a conservative screening metric instead of core damage frequency or large early-release frequency, which are more commonly used in probabilistic risk assessment. The use of CPF as a risk surrogate was considered appropriate for a generic evaluation of ESD effects, and CPF values below 1×10^{-6} were deemed risk insignificant. Appendix A and Reference 17 give more details on the FAVOR inputs, the analysis assumptions, the approach adopted to develop the model plants and apply the ESD, the plant loading transients, and the analysis of results.

Figure 5 shows typical results for a PWR analysis scenario, with CPI and CPF calculated as a function of ESD for both 1/4T and 0.03T flaws (SSBF) in a vessel that is cooling down following the P-T limit curve at the maximum allowable cooldown rate of 100 degrees F/hr. For a negative or small positive ESD, the CPF in all cases is well below 1×10^{-6}. However, as ESD increases, the CPF increases monotonically and smoothly. All such PWR and BWR scenarios show similar trends, differing principally in the actual CPI or CPF values calculated for a particular ESD. In Figure ***5***, for example, the CPF of the 1/4T flaw increases more rapidly with ESD than that of the 0.03T flaw, and it generally bounds the 0.03T results. However, the CPF of the 0.03T flaw is bounding in other scenarios.

Table 3 summarizes the results of the FAVOR scoping runs. The scoping study demonstrates that the CPF for realistic BWR heatup and cooldown on the saturation curve is generally low, regardless of the ESD. During leak testing, the CPF for BWRs only exceeds 1×10^{-6} for ESD values over 100 degrees F. The CPF increases slightly with higher loading rates for the transients considered. Based on the earlier targeted sample evaluation [Ref. 12] and the filtered surveillance capsule data [Ref. 18], it is not expected that the ESD for any BWR plant will exceed 100 degrees F at 80 years of operation.

Figure 5 CPI and CPF results for a transient that follows the P-T limit curve

However, for operation on the P-T limit curve at the highest allowed cooldown rate, the PWR CPF exceeds 1×10^{-6} for both shallow and 1/4T flaws at ESD values over 20–50 degrees F. At least a few plants are predicted to have ESD values over 50 degrees F after 80 years of operation, based on both the targeted sample evaluation and filtered surveillance capsule results. The CPF was generally low for the actual PWR transients studied, although it was almost always higher for the SSBF than for the 1/4T flaw.

The scoping study was also invaluable for identifying potentially risk-significant operational characteristics of an operating plant. The highest failure probability for deeper ID surface flaws occurs near the beginning of the P-T limit cooldown curve, where operating pressure can be held while cooling is initiated. Conversely, for SSBF, the highest failure probability occurs near the end of the P-T limit cooldown curve, when the cladding interface stresses are relatively high and some repressurization is allowed.

The following example illustrates this point. For a scenario created to better represent an actual transient (i.e., with faster pressure decrease than required by ASME at the beginning of the cooldown, then repressurization at lower temperature as allowed by the P-T limit curve), the CPF values were significantly less than those given in Table 3 for a transient following the P-T curve for a postulated 1/4T flaw. The CPF values for a postulated SSBF were consistent with those for a 1/4T flaw for a transient that followed the entire P-T limit cooldown curve (Table 3).

Table 3 Summary of FAVOR Scoping Runs

Transient Type	SSBF	1/4T Flaw	Additional context[6]
BWR P-T Limit Cooldown	CPF $\leq 1\times10^{-6}$ for all ESDs	CPF $\geq 1\times10^{-6}$ for ESD > 40 °F	BWRs must cool down on saturation curve, so cooldown on licensed limits is not plausible.
BWR Saturation Cooldown	CPF $\leq 1\times10^{-6}$ for all ESDs	CPF $\leq 1\times10^{-6}$ for all ESDs	
BWR Leak Test, Cooldown Rate ≤ 50 °F/hour	CPF $\leq 1\times10^{-6}$ for all ESDs	CPF $\geq 1\times10^{-6}$ for ESD > 100 °F	Additional information is desired to determine whether high cooldown rates are possible, or ASME BPVC action will be pursued to prohibit this scenario.
BWR Leak Test, Cooldown Rate > 50 °F/hour	CPF $\leq 1\times10^{-6}$ for all ESDs	CPF $\geq 1\times10^{-6}$ for ESD > 100 °F	
PWR P-T Limit Cooldown	CPF $>1\times10^{-6}$ for ESDs ≥ 50 °F	CPF $> 1\times10^{-6}$ for ESD ≥ 20 °F	Additional information on event frequencies is desired to confirm TWCF< 1×10^{-6} /year.
PWR Cooldown, Actual Transients	CPF $< 1\times10^{-6}$ for most transients	n/a	

The failure risk associated with an actual cooldown transient therefore depends on how closely the transient approaches these higher-risk locations of the P-T limit curve, in conjunction with the probability that cracks of the corresponding types exist. However, it is expected that the combined frequency with which such cracks occur and a cooldown approaches a high-risk portion of the P-T limit curve is much less than 1/yr (Appendix A, Section A.3). Therefore, the expected generic TWCF for all the transients analyzed should be much less than 1×10^{-6}/yr, at least for ESD values below 100 degrees F, which bounds the ESD values calculated for the targeted sample. However, for plants that have only limited, low-fluence surveillance data, ESD values could exceed 150 degrees F by 80 years of operation, which means the TWCF could exceed 1×10^{-6}/yr. Sections 3.6, 4, and 5 give more information on the possible development and safety impact of these higher-risk conditions.

3.6. Uncertainties Associated with the Staff's Evaluation

The staff's quantitative evaluation of the RG 1.99 ETC underpredictions at high fluence indicates that the expected generic risk is not significant; for example, the highest increases in RT_{NDT} for the targeted sample plants are below 50 degrees F. However, as discussed previously, it is difficult to extend this finding to specific plants because of the relatively large ESD values possible at some plants, coupled with existing analysis uncertainties in the scoping study. This section discusses some of these analysis uncertainties, the additional uncertainties associated with previous observations on RPV integrity (Appendix A), and the role of

6 Information for the benefit of the reader

performance monitoring to provide assurance that the plant-specific impact of these uncertainties is not significant.

3.6.1. Probabilistic Fracture Mechanics Scoping Study Uncertainties

As indicated in Section 3.5, it is appropriate to use a CPF of 1×10^{-6} as a conservative screening criterion for evaluating the generic risk associated with ESD values (i.e., ETC underpredictions), given that the corresponding generic TWCF is also expected to be less than 1×10^{-6}/yr. However, a plant-specific TWCF is more difficult to quantify, or appropriately bound, because of large differences in fabrication and operational practices (discussed further in Appendix A) that ultimately affect the TWCF. Recall that the TWCF is the product of the transient frequency, the probability of having a flaw, and the CPF. As detailed in Appendix A, there are unquantified uncertainties associated with the frequency of a challenging cooldown transient, the probability of having a critical flaw, and in the CPF estimates themselves. The impact of these uncertainties is that the TWCF could vary by several orders of magnitude across the fleet. While there is no evidence that the TWCF exceeds 1×10^{-6}/yr at any particular plant, the combined effects of ETC underprediction and insufficient surveillance monitoring, as detailed later, erode the safety margin and degrade confidence that this metric is upheld.

3.6.2. Uncertainties Associated with Recent Reactor Pressure Vessel Integrity Issues

As discussed in Section 2 and Appendix A, since other factors previously studied in relation to RPV integrity (e.g., SSBFs, hydrogen cracking, Branch Technical Position 5-3, and carbon macrosegregation) were evaluated generically and independently, it is challenging to assess their plant-specific impacts in conjunction with the potential of the RG 1.99 ETC underprediction at high fluences. Ideally, the fabrication, inspection, and operational history of the plant, as well as the plant-specific system constraints affecting its operation, would be known. This information would permit analysis of each RPV using actual information on its material toughness and flaw distribution, which would be coupled with the plant's loading history and system operation to incorporate loading constraints. Only then could the plant's quantitative risk due to RPV failure be clearly quantified.

Such an evaluation would require significant resources to be tenable. However, in all the factors of concern, the fracture toughness properties of plant-specific RPV materials are a fundamental consideration. A more accurate characterization of these properties could support an engineering assessment to provide reasonable assurance that adequate plant-specific margin remains in spite of the combined effects of these issues.

3.6.3. Impact of Performance Monitoring on Plant-Specific Embrittlement Predictions

The purpose of the plant-specific surveillance data required by Appendix H to 10 CFR Part 50 is to capture unique behavior due to plant-specific characteristics that may not be adequately represented by the generic data used in developing ETC predictions. In essence, the plant-specific surveillance data validate that the generic ETC accurately predicts the plant's behavior and give licensees time to adjust their P-T operating conditions and assess the significance of PTS challenges (for PWRs). Ideally, the generic embrittlement trends in the complete database of materials would represent the overall behavior of every plant within the population; however, this may not always be the case. Figure **6** presents the difference between the embrittlement shift predicted using the RG 1.99 ETC and the measured embrittlement shift from surveillance data, as a function of fluence. An ordinate value of zero represents a perfect prediction by the ETC. The small black squares represent both U.S. and international base metal surveillance data, while the solid colored symbols represent plant-specific surveillance data for three U.S. plants. For one case (Plant 1), the prediction becomes more conservative as fluence increases, for another (Plant 2) the amount of underprediction increases with fluence, and for the third (Plant 3) the prediction is always conservative. A horizontal line fit through the plant-specific data would indicate that the trends are properly predicted (although with a positive or negative bias). It is not known why the plant-specific trends differ, but RG 1.99 provides guidance that the ETC embrittlement predictions be adjusted if the plant surveillance data are deemed credible. This is accomplished by curve-fitting the plant-specific data using the RG 1.99 fluence function (see Section **Error! Reference source not found.** for an example). This curve fit is used to make future embrittlement predictions, until more data become available.

As described in Section 3.1, the overall database suggests that the RG 1.99 ETC underpredicts embrittlement at high fluence. For some plants (e.g., Plants 1 and 3 in Figure **6**), the available surveillance data suggest a different trend from that for the overall database (i.e., that the RG 1.99 ETC is accurate). If no further surveillance data are obtained for these plants, the licensees may erroneously assume the RG 1.99 predictions continue to be appropriate, and may continue to operate the plants accordingly, even as the fluence increases beyond the levels covered by the existing surveillance data; not realizing that this may not be an accurate representation of the actual material trend at high fluence. In these cases, (Plants 1 and 3 and similar plants) licensees may underestimate embrittlement shifts by up to 180 degrees F (100 degrees Celsius), significantly reducing the margins expected in their P-T limit curves. Continued acquisition of plant-specific embrittlement data at high fluence is the only effective way to validate or monitor the performance of the ETC predictions, limit prediction uncertainty, and avoid plant-specific extrapolation errors of embrittlement data.

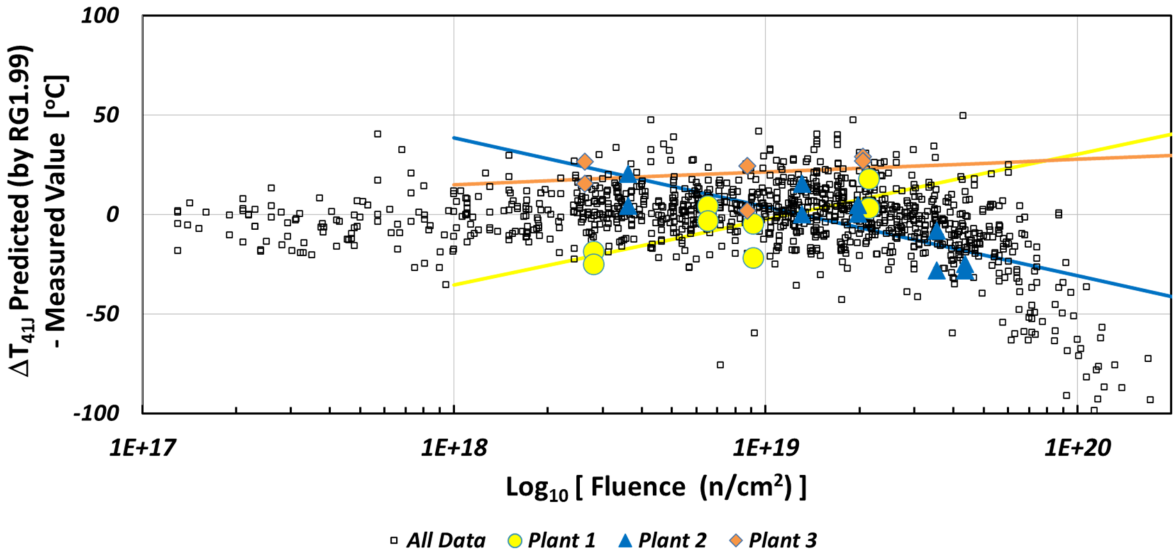


Figure 6 Illustration of plant-specific data compared against the complete database for base metals

4. Adequacy of Surveillance Programs for Plant Operation beyond 60 Years

Appendix H to 10 CFR Part 50 incorporates by reference ASTM E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," as the most recent standard to govern the design and implementation of RPV material surveillance programs. This standard, issued when plants were early in their initial 40-year license periods, did not consider the potential for longer operating periods.

The NRC has not revised Appendix H to 10 CFR Part 50 to account for extended plant operation beyond 40 years, either by incorporating by reference a more recent standard that addresses extended plant operation, or by including explicit provisions in the regulation. Additionally, as described below, ASTM E185-82 has several provisions related to the capsule withdrawal schedule that can lead to increased uncertainty in monitoring of RPV embrittlement. Finally, as described in Administrative Letter (AL) 97-04, "NRC Staff Approval for Changes to 10 CFR Part 50, Appendix H, Reactor Vessel Surveillance Specimen Withdrawal Schedules," dated September 30, 1997 [Ref. 19], surveillance capsule withdrawal schedule "changes that conform to ASTM E185 require only staff verification of such conformance."

Because Appendix H to 10 CFR Part 50 does not specifically treat operation beyond 40 years, licensees with 60-year or 80-year operating licenses have maintained their surveillance programs in conformance with Appendix H to 10 CFR Part 50, as supplemented by the following license renewal guidance: (1) NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," Revision 0, issued July 2001 [Ref. 20], Revision 1, issued September 2005 [Ref. 21], and Revision 2, issued December 2010 [Ref. 22], for plants operating to 60 years; and (2) NUREG-2191, "Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report," issued July 2017 [Ref. 23], for plants operating to 80 years. For example, the GALL-SLR report (in Section XI.M31, "Reactor Vessel Material Surveillance") states, "This program includes withdrawal and testing of at least one capsule addressing the subsequent PEO with a neutron fluence of the capsule between one and two times the peak neutron fluence of interest at the end of the subsequent PEO." For plants with existing data that cover this fluence range, licensees do not need to gather additional data, since they have already characterized the behavior of their material over the planned operating period. However, since this guidance is not a regulatory requirement, licensees that commit to the withdrawal of one capsule to meet this provision may later change their commitment and still be consistent with the regulations and their current licensing basis (CLB); any change in the withdrawal schedule requires prior NRC approval in accordance with Appendix H to 10 CFR Part 50, but this approval is controlled by AL 97-04 and is limited to verification that the changes conform to the ASTM standard.

The relevant provisions of ASTM E185-82 are all contained in the capsule withdrawal schedule in Table 1 of the standard (see Appendix A). From that table, the second-to-last capsule in each schedule (column) is listed for withdrawal at 15 EFPYs or, in accordance with Footnote B, "at the time when the accumulated neutron fluence of the capsule corresponds to the approximate EOL [end-of-life] fluence at the reactor vessel inner wall location, whichever comes first." Because the standard mixes a firm withdrawal time (15 EFPYs) and a performance-based time, the capsule could be withdrawn either at the 32-EFPY fluence, or, depending on the capsule lead factor, at around the 15-EFPY level. Footnote E of Table 1 states that the last capsule in the program is withdrawn at "not less than once or greater than twice the peak EOL vessel fluence"; it also states, "This capsule may be held without testing following withdrawal." Based on the first part of the footnote, ASTM E185-82 allows a licensee to delay withdrawal of a capsule that was originally intended to address 40 years of operation (or 32 EFPYs per the standard) until its fluence is close to the 80-year peak RPV fluence (specifically, 64 EFPYs). Combined with the provisions for the second-to-last capsule, this allows for a substantial gap in capsule fluence and testing time. Furthermore, based on the second part of Footnote E, the last capsule could be withdrawn, at a fluence that represents between 32 and 64 EFPYs, and then held without testing. In this scenario, the available surveillance data could cover as little as 15 EFPYs of plant operation, as compared to a potential 72 EFPYs for 80 years of plant operation. And these changes would still be considered to conform to ASTM E185-82.

ASTM E185-82 defines the plant "end of life" (EOL) as the "design lifetime in years," and the withdrawal schedule refers to a design life of 32 EFPYs (40 years times a capacity factor of 80 percent). Several recent safety assessments of surveillance capsule schedule changes have effectively interpreted the ASTM E185-82 "design life" as equal to the plant license period, e.g., 40 years for the initial license, 60 years for a renewed license, or 80 years for a subsequently renewed license. Thus, the provisions in Table 1 of ASTM E185-82 could permit withdrawal of the last capsule when it reaches the fluence at 120 years (for a plant with a renewed license) or 160 years (for a plant with a subsequently renewed license). Given the second part of Footnote E to the table, the licensee may be able to continue operating the plant with renewed licenses and either never withdraw this capsule or withdraw it and hold it without testing.

Appendix H to 10 CFR Part 50 specifies that the NRC must approve the withdrawal schedule before implementation (III.B.3). However, as noted earlier, AL 97-04 states, "[Schedule] changes that conform to ASTM E185 require only staff verification of such conformance."

This position instructs the staff to perform a conformance review to ASMT E185-82 in lieu of a detailed technical evaluation, such as that performed for a license amendment request, to verify whether the schedule change is appropriate (for example, with respect to long gaps in operating time and neutron fluence between the prior capsule test and the proposed change in the withdrawal schedule for the last capsule). As explained above, a licensee could have fluence data applying only to the early part of plant operation (e.g., 15 EFPYs or about 20 years of operation), change its schedule so as to withdraw the last capsule only when it reaches 64 EFPYs, and then hold the capsule without testing. This would conform fully to ASTM E185-82, and so in accordance with AL 97-04, the staff could approve the schedule to its conformance to the standard, without performing a detailed technical evaluation of this scenario.

Some licensees have periodically delayed withdrawal of their last capsule so that it matches the peak RPV fluence at the end of the currently licensed operating period, generally for a 60-year renewed license or an 80-year subsequently renewed license. In some cases, the withdrawal of the last capsule, initially intended for 40-year fluence levels, has been delayed multiple times, with the capsule in essence "triple counted" to address first 40-year and then also 60- and 80-year fluence levels. These repeated delays in withdrawal have sometimes created large gaps in time and fluence between the second-to-last and the last capsule. At present, if licensees choose (consistent with ASTM E185-82) to delay withdrawal of these "last capsules" and ultimately hold them without testing, then they may fail to gather the plant-specific surveillance data at 80-year high fluence levels needed to validate current ETC estimates.

The examples below show actual or planned capsule withdrawal schedules and the data gaps that can result. As a starting point and base case, Figure **7** illustrates the history of a plant with a renewed license for operation to 60 years, for which surveillance testing has been spaced to provide data throughout the plant's operating life, including at the 40-year and 60-year peak RPV fluence levels. This represents the ideal implementation of ASTM E185-82 and Appendix H to 10 CFR Part 50, which are intended to enable the monitoring of plant-specific changes in RPV fracture toughness properties due to the variability in the behavior of reactor vessel steels caused by long-term exposure to the neutron radiation and temperature environment.

Figure 8–10 show cases where the withdrawal of the original 40-year capsule has been delayed multiple times to address the maximum RPV fluence at 80 years of plant operation. The plants in Figure 8 and Figure 9 have been approved for subsequent license renewal for operation to 80 years, whereas the plant in Figure 10 has a renewed license for 60 years. In the case of Figure 10, the last-tested capsule represents about 25 years of plant operation, and the plant is nearing 50 years of operation. The plants in Figure 9 and Figure 10 have data at approximately 30 and 40 years of operation, respectively. In Figure 9, the withdrawal of the last capsule has been delayed sequentially to address fluences for 60 and 80 years of operation. In Figure 10, the withdrawal of the last capsule was initially delayed to address the fluence for 60 years of operation, then delayed further by a small timeframe to a fluence that approximates 80 years of operation.

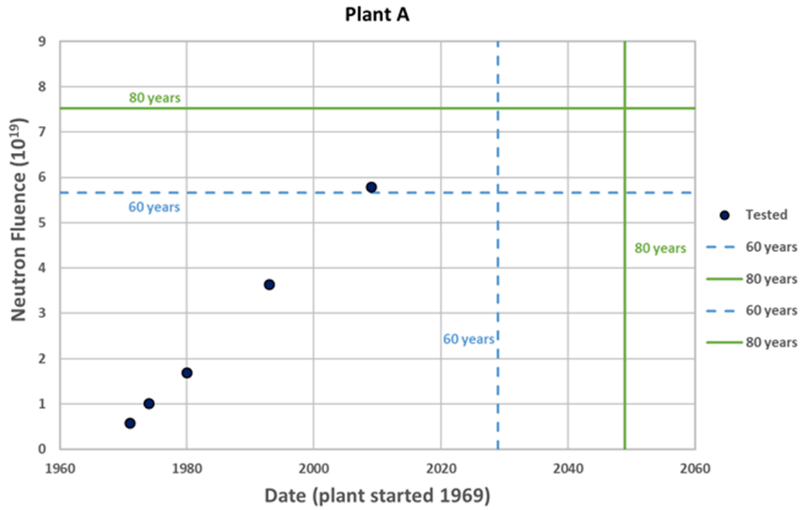


Figure 7 Capsule withdrawal history for Plant A, indicating periodic withdrawal and testing of capsules throughout plant operation, with capsules tested at the 40-year and 60-year peak RPV fluence levels

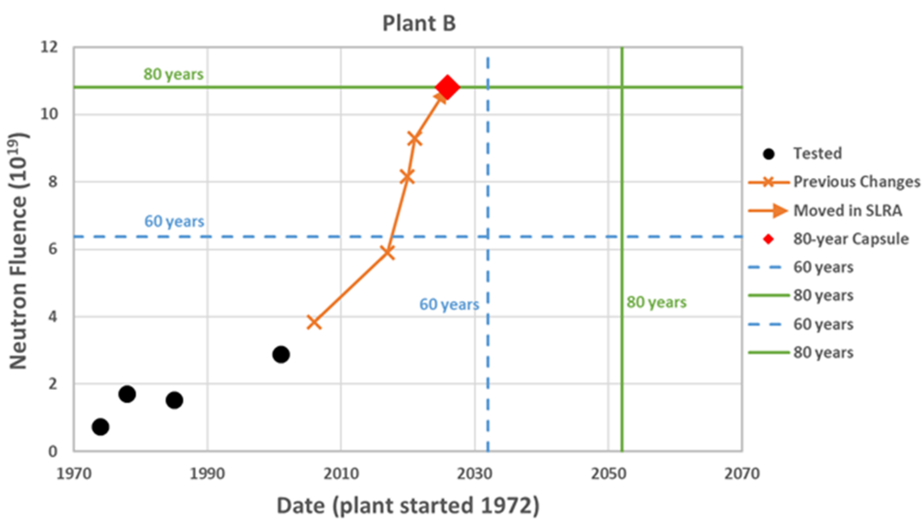


Figure 8 Capsule withdrawal history for Plant B, where the withdrawal of the capsule originally designed to apply to 40 years has been delayed multiple times and is currently credited to address 80 years of plant operation, and the highest-fluence data represent about 25 years of plant operation

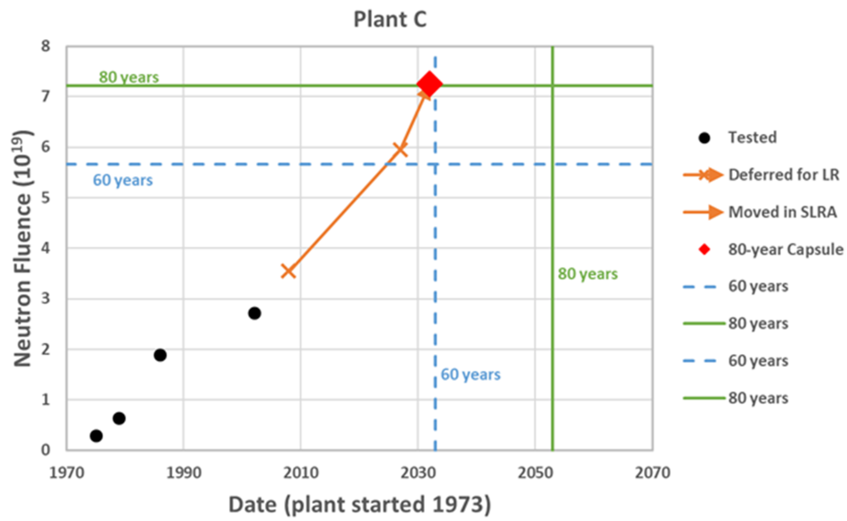


Figure 9 Capsule withdrawal history for Plant C, where the withdrawal of the capsule originally designed to apply to 40 years has been delayed sequentially to address 60 years and then 80 years of plant operation

As these figures show, under the current regulatory structure in Appendix H to 10 CFR Part 50, combined with the provisions of ASTM E185-82 and AL 97-04, plants may repeatedly delay capsule withdrawals in PEOs and potentially hold the last capsule without testing, which strictly limits their ability to periodically monitor embrittlement as dictated by Appendix H to 10 CFR Part 50. In such cases, the limited availability of surveillance data that is available, when combined with the permissible exclusion of future testing, would prevent plant-specific verification of the adequacy of the embrittlement trends from RG 1.99, even for cases where the plant will experience fluence levels above 1×10^{20} n/cm^2 during and beyond the subsequent PEO.

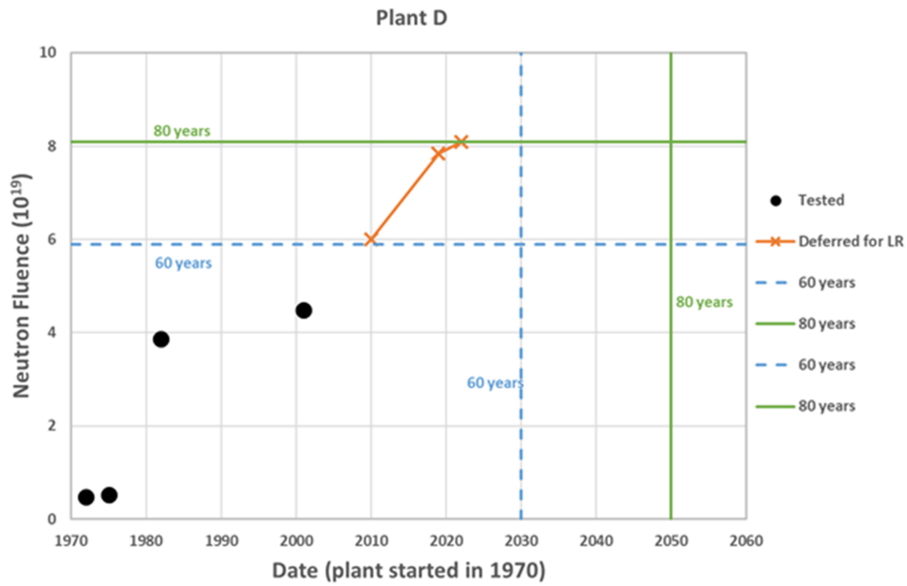


Figure 10 Capsule withdrawal history for Plant D, where the capsule originally designed to apply to 60 years is now credited to address 80 years of plant operation

Another way to interpret the surveillance capsule data is to plot the plant-specific surveillance capsule fluences along with the projected plant fluence levels at 60 and 80 years of operation, using the format of Figure ***1***. As illustrated in Figure ***11*** for Plant B (introduced in Figure 8), the current capsule data (shown by **green lines**, with the capsule withdrawal dates indicated) have been acquired at fluence levels where RG 1.99 has been shown to give reasonably accurate predictions; the final capsule (whose withdrawal and testing of which has been deferred multiple times) is scheduled for testing at a fluence that (1) bounds the plant's 80-year fluence, and (2) is on the part of the curve where RG 1.99 is likely to underpredict embrittlement, based on prior data. The projected 60-year fluence level (shown in **blue lines**, with the 60-year operation date indicated) is near where RG 1.99 ETC underpredictions begin to appear, and the 80-year fluence level (also shown in **blue lines**, with the 80-year operation date indicated) is essentially the same as the planned fluence for testing of the last capsule. If this capsule is held without testing, as permitted by AL 97-04 together with Appendix H to 10 CFR Part 50 and ASTM E185-82, the licensee would be projecting its 80-year embrittlement trends using only the available data, which neither bound the plant's 80-year fluence nor adequately model the embrittlement as a function of fluence. It is therefore essential to test this capsule to ensure that the plant is accurately predicting the RPV embrittlement at 80-year fluence levels.

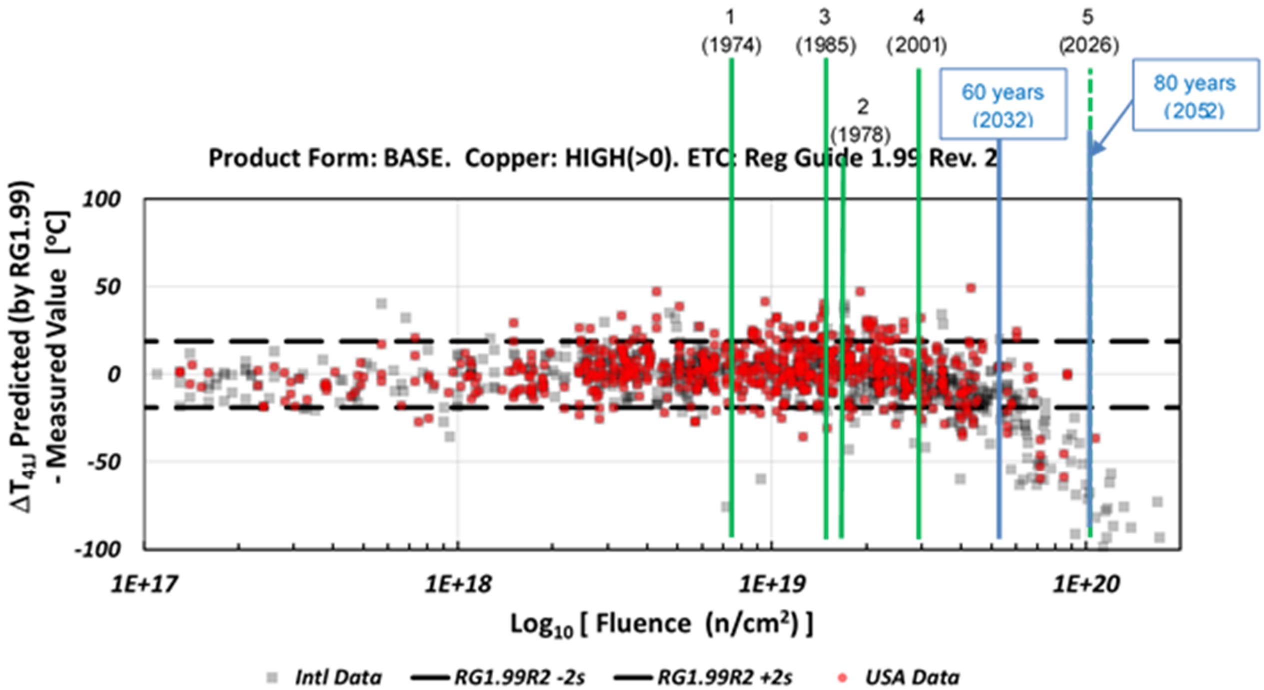


Figure 11 Another view of the history of Plant B, where the capsule originally designed to apply to 40 years has been delayed multiple times and is currently credited to address 80 years of plant operation, while the highest-fluence data available represent about 25 years of plant operation; green lines indicate surveillance capsules fluences, and blue lines indicate peak RPV fluences at 60 and 80 years of operation

5. Impacts on Margins for Normal Operation

As discussed in Section 2 and Appendix A, the regulations are intended to work synergistically to provide reasonable assurance of RPV integrity, in part through the establishment and maintenance of adequate safety margins. RG 1.174 delineates the role and importance of maintaining adequate safety margins in the risk-informed decisionmaking process [Ref. 2]. The general premise associated with maintaining adequate safety margins is that licensing-basis changes should not compromise the fundamental safety principles that are the basis of plant design and operation (i.e., activities such as maintenance, testing, inspection, and qualification). Therefore, the plant's CLB is the reference point for judging whether a proposed change to this basis maintains adequate safety margins. The effects of the proposed change should be assessed through an engineering evaluation, with the objective to verify that (1) the codes and standards or their NRC-approved alternatives are met, and (2) safety analysis acceptance criteria in the plant-specific CLB (e.g., the final safety analysis report, supporting analyses) are met, or proposed revisions provide adequate margin to account for uncertainty in the analysis and data.

To ensure RPV integrity, the plant's CLB requires, in part, conformance with Appendices G and H to 10 CFR Part 50 and with 10 CFR 50.61 (for PWRs). The staff's evaluation (Section 3) indicates that the biggest expected impact on safety margin is associated with the Appendix G requirements. Therefore, as stipulated in RG 1.174, a plant's Appendix G CLB is the reference point for judging whether adequate safety margins are maintained. What follows is an engineering evaluation, consistent with maintaining adequate safety margins per RG 1.174, to consider how plant-specific inaccuracies in the current ETC predictions, coupled with less-frequent testing and a lack of high-fluence data, may decrease the Appendix G safety margin while increasing its uncertainty as a plant age. The evaluation aims to determine whether adequate Appendix G margins are maintained in light of these factors.

The calculation of P-T limits for normal operating conditions is inherently conservative because of several underlying assumptions, described in Sections 2, 3.5, and 3.6.1 of this document. The conservative nature of the P-T limit curve required in Appendix G to 10 CFR Part 50 implicitly defines the safety margin needed for adequate protection as described in Section 3 of this report. An additional margin between the plant operating conditions and the licensed P-T limits arises from the low-temperature overpressure protection system and other operational constraints. This section describes how the underprediction of embrittlement due to the RG 1.99 ETC and lack of plant-specific surveillance testing may impact these safety margins.

Figure 12 shows an example. In this figure, the ordinate represents the change in RT_{NDT} with embrittlement, and the abscissa represents the specimen fluence level. The solid blue symbols represent the surveillance data measured by a currently operating reactor, with the last data point representing a surveillance capsule that was tested after the plant had operated for 25 EFPYs.

The blue line in the figure represents the best fit through the plant surveillance data using Regulatory Position 2.1 of RG 1.99. The solid orange line represents the prediction based on the plant-specific material chemistry, using Regulatory Position 1.1 of RG 1.99. Clearly, the plant surveillance data suggest that the RG 1.99 ETC (the solid orange line) overpredicts the trends for this plant. The dashed orange line corresponds to the solid orange line minus twice the standard deviation required from RG 1.99. However, since surveillance data are available only for an early period of operation and a limited fluence range, it is unknown whether the blue line truly represents the future embrittlement behavior for this plant, especially in the high fluence range (e.g., above 6×10^{19} n/cm^2).

The solid green line, representing a curve fit of the overall U.S. surveillance data at high fluence, is meant to address the underprediction described in Section 3.1, while the dashed green lines correspond to the green line plus and minus twice the standard deviation. Note that the standard deviation around the green line increases with fluence, and beyond 9×10^{19} n/cm^2 it is extrapolated, since high-fluence data are limited; this makes the trend more uncertain at higher fluence levels.

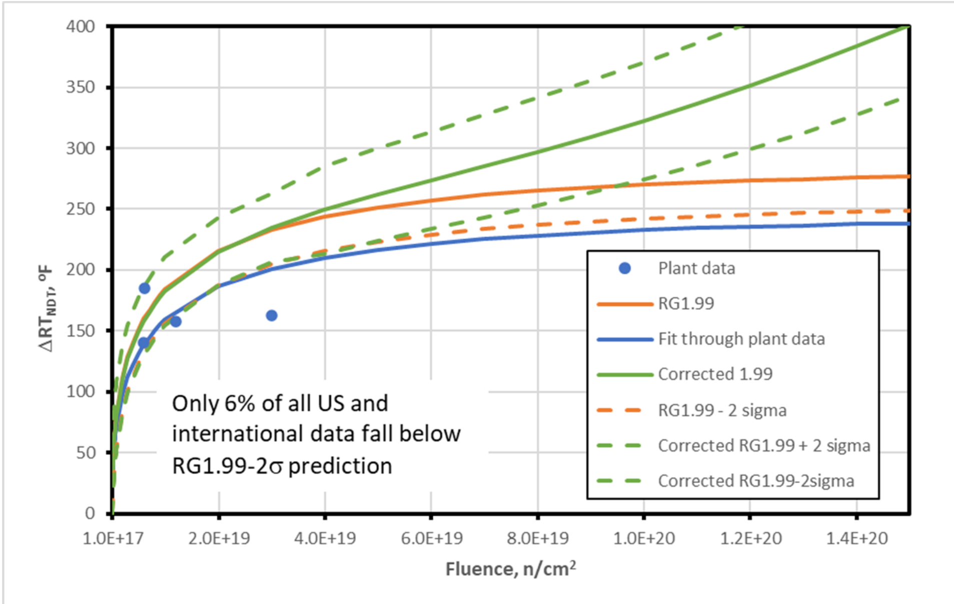


Figure 12 Predictions of embrittlement shift

From Figure 12, it is not immediately apparent how the underprediction of RT_{NDT} with increasing fluence affects the plant's operating behavior. The data suggest that at a fluence of 1×10^{20} n/cm^2, the underprediction in □RT_{NDT} could range from about 50 to 150 degrees F (blue line to dashed green lines). As described in Section 3.5 and Reference 17, this change could increase the CPF and TWCF by more than two orders of magnitude, possibly making certain unanalyzed plant-specific transients a safety concern.

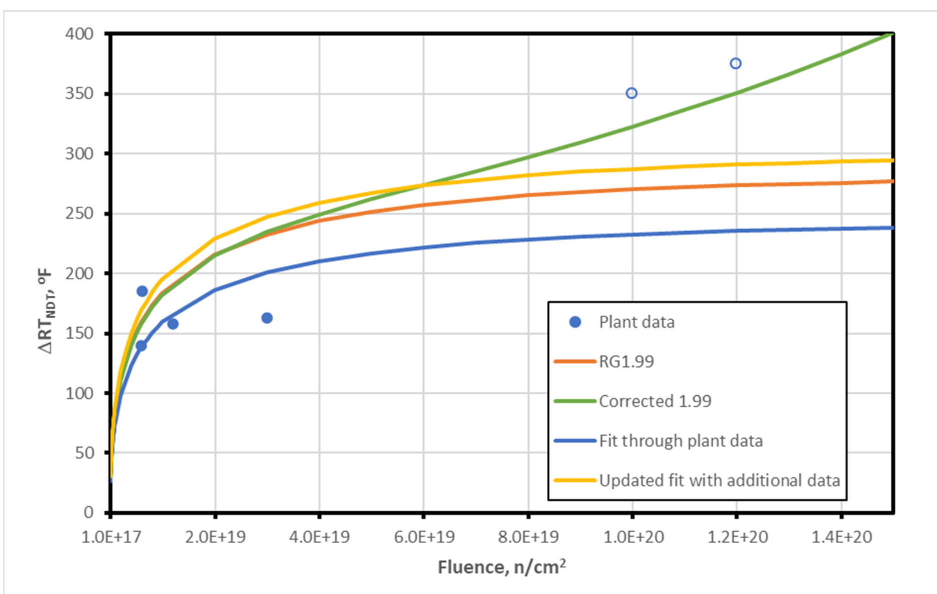


Figure 13 Predictions of embrittlement shift with additional data

Figure 13 shows hypothetical additional surveillance data obtained for this plant at high fluence (that follow the adjusted embrittlement trend shown in Figure ***12***), together with a fit of the data using Regulatory Position 2.1 of RG 1.99. In this figure, the open blue symbols represent the hypothetical data, which follow the green curve, and the yellow curve represents the fit through all the plant data (solid and open blue symbols) using Regulatory Position 2.1 of RG 1.99. The other curves are the same as in Figure ***12***. While the additional data elevate the embrittlement trend fit, the use of the fluence function from Regulatory Position 2.1 of RG 1.99 still results in large differences between the actual material behavior (blue symbols) and the predicted material behavior (yellow curve). In fact, in some cases, the difference is over 56 degrees F, which in accordance with RG 1.99 would make the corresponding data noncredible, leading the licensee to use the orange curve as the ETC[7] and thus underpredict the actual embrittlement even more severely.

Because the current procedure in RG 1.99 is to fit the plant-specific surveillance data to the fluence function of the ETC, the shape of the function becomes important for proper embrittlement prediction. As shown in Figure ***14***, the fluence function begins to change slope at approximately 3×10^{19} n/cm^2 and reaches a maximum at about 2×10^{20} n/cm^2. This behavior occurs because the developers of the fluence function did not have sufficient data to properly fit the function within this high fluence range, and likely did not envision its use at such high fluence levels. If high-fluence surveillance data are used to determine plant-specific embrittlement behavior, this fluence function requires modification at high fluence to prevent the underprediction illustrated in Figure ***13***.

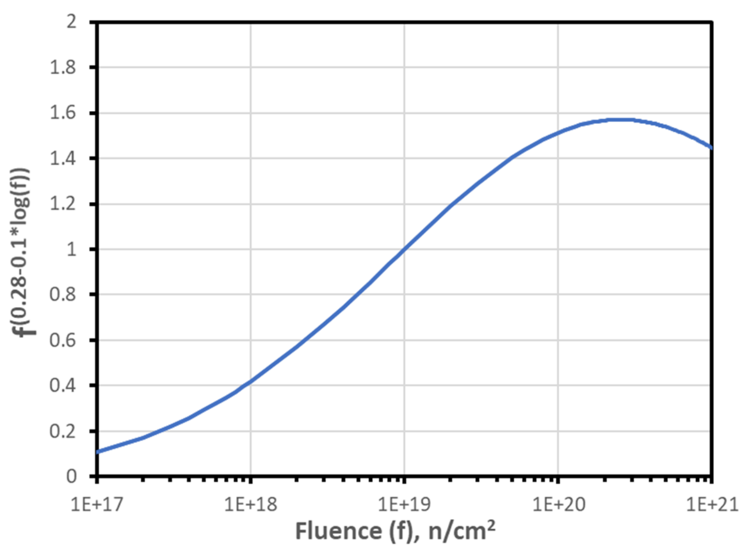


Figure 14 RG 1.99 fluence function

As explained earlier, this level of underprediction in embrittlement may not significantly affect the TWCF. However, it has a clear impact on safety margins. These margins are illustrated in Figure ***15***, with the P-T curves at a high embrittlement level compared to the typical operating window. The blue curve represents the P-T structural limit, where RPV failure would be expected. The green curve represents the allowable P-T limits using Appendix G to 10 CFR Part 50 and accurate predictions of the embrittlement. The gap between the blue and green curves represents an adequate margin, as intended by the regulations. The orange curve represents the P-T limits calculated with Appendix G to 10 CFR Part 50 and the RG 1.99 ETC,

7 In some situations, other methods have been used and approved for determining whether data are credible.

which underpredicts embrittlement at high fluence. The actual margin to failure is defined by the conservative nature of the P-T calculation and the accuracy of the embrittlement prediction. The gap between the blue and orange curves represents the reduced margin due to the underpredictions by the RG 1.99 ETC at high fluence levels. This reduction in the margin occurs because of inadequate accounting for the underprediction in the RG 1.99 ETC at high fluence levels typical of 80 years of plant operation, coupled with the potential unavailability of plant-specific surveillance data to verify the adequacy of the embrittlement trends assumed for the RPV. To re-establish the margin defined by Appendix G to 10 CFR Part 50 would require corrected embrittlement estimates.

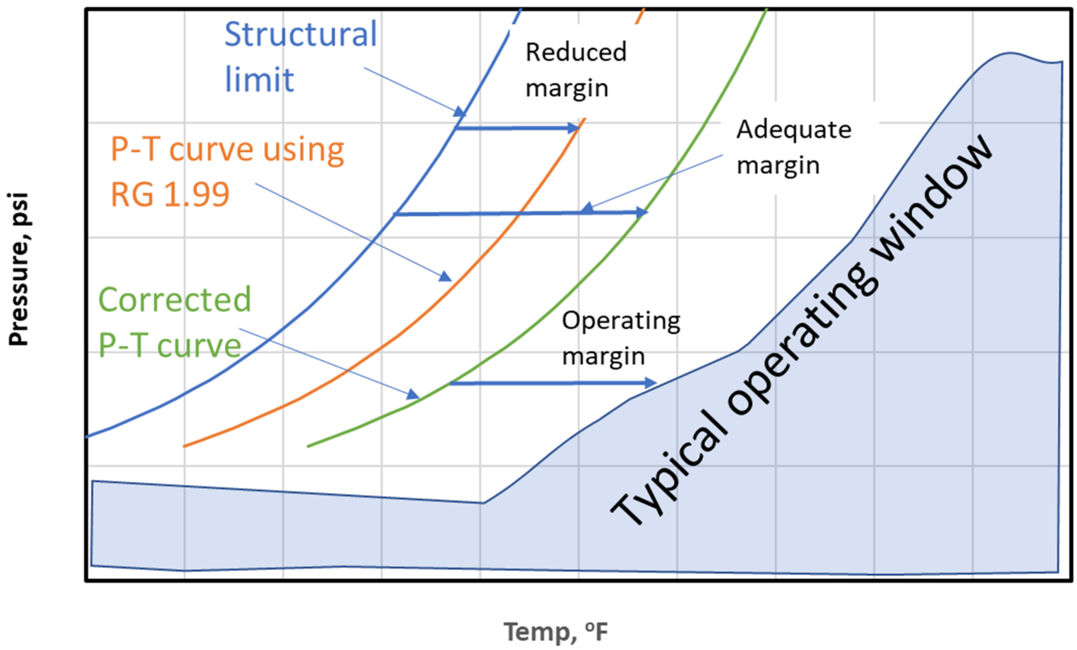


Figure 15 Notational illustration of P-T curve margin

Unfortunately, the reduction in margin is difficult to quantify. Because the level of conservatism in the P-T calculations using Appendix G to 10 CFR Part 50 was deemed appropriate for adequate protection, the reduction in the safety margin is characterized by embrittlement underpredictions and the associated uncertainty. As described earlier, when surveillance data are limited, the currently assumed embrittlement trends cannot be verified (see Figure ***12***), and the uncertainty due to RG 1.99 ETC underpredictions can overwhelm the safety margins.

6. Risk-Informed Evaluation

As described in Section 1, the purpose of this paper is to assess the safety significance of two interdependent phenomena: the underprediction of RPV embrittlement arising from the use of the ETC in RG 1.99 (and 10 CFR 50.61) at high fluence levels, and a potential lack of future plant-specific surveillance data for operation beyond 60 years. The staff structured the assessment in terms of the five principles of risk-informed decisionmaking embedded in both RG 1.174 and LIC-504, “Integrated Risk-Informed Decision-Making Process for Emergent

Issues," dated May 30, 2014 [Ref. 24], which contains staff guidance for evaluating and communicating risk-informed decisions.

The following sections assess the issues described in this paper in relation to each of these five principles of risk-informed decision making.

6.1. Principle 1: Compliance with Existing Regulations

The pertinent regulations, described in Section 2 and Appendix A of this report, include the following:

- Appendix G to 10 CFR Part 50
- Appendix H to 10 CFR Part 50
- 10 CFR 50.61
- 10 CFR 50.61a
- 10 CFR 50.55a, "Codes and standards"
- 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for light-water nuclear power reactors for normal operation" (invokes Appendices G and H to 10 CFR Part 50)

Assessment of compliance with the existing regulations is not necessary since the decision under consideration involves changes to the regulations or guidance (most likely to the allowable P-T curves and to plant-specific surveillance programs). Plants are currently meeting the regulations; the issue is that these regulations (Appendix H to 10 CFR Part 50) and the associated guidance (RG 1.99) may not ensure safety margins consistent with their original intent, in particular for high fluence plants.

6.2. Principle 2: Consistency with the Defense-in-Depth Philosophy

To assess how an issue might degrade defense in depth, it is important to understand how it affects the balance among the layers of defense. The aspect of defense in depth that underprediction of embrittlement and lack of surveillance data may affect is "barrier integrity." The reactor coolant pressure boundary is one of three independent fission product release barriers in a U.S. plant. The NRC has determined that acceptable failure probabilities for RPV integrity are a 95-percent TWCF of less than 1×10^{-6}/yr for PTS events [Ref. 25]. The same criteria can be applied to normal operating conditions assuming the frequency of the transient is known; for example, actual cooldown transients have a frequency of approximately 1/yr. When the cooldown transient frequency is difficult to determine (e.g., for a cooldown along the P-T limit), a surrogate criterion of CPF less than 1×10^{-6} is reasonable. The PTS evaluations summarized in Section 3 of this paper demonstrate that the 95-percent TWCF for PTS is less than 1×10^{-6}/yr for operation to 80 years. On the other hand, for normal operating conditions, under certain cooldown conditions (along the P-T curve) and when the ESD exceeds

100 degrees F (for BWR leak tests), the calculated CPF values are greater than 1×10^{-6}. However, for BWRs, the ESD is not expected to exceed 100 degrees F for operation to 80 years of operation (owing to generally lower fluence levels), and for PWRs, the frequency of occurrence of a transient following the P-T curve is very low. Therefore, these issues will not impact the barrier integrity and is consistent with the defense-in-depth philosophy. However, additional analyses and considerations may be needed to determine whether these issues sufficiently erode defense in depth for operation beyond 80 years.

6.3. Principle 3: Maintenance of Adequate Safety Margins

As described in Section 5, RG 1.99 underpredictions of embrittlement and a lack of plant-specific surveillance data at high fluence can impact the safety margins to RPV failure. According to Appendix G to 10 CFR Part 50, these margins to brittle failure are defined by the conservative nature of the Appendix G analyses coupled with accurate predictions of embrittlement due to irradiation. Effective surveillance monitoring during the entire operating period of a plant provides assurance of accurate predictions of embrittlement. Under the existing regulations, a plant may have no limiting material data points, or possibly only one, at high fluence; this circumstance may cause large uncertainty in embrittlement predictions, depending on plant-specific circumstances. Furthermore, an accurate ETC that appropriately models high fluence data trends adds assurance that the embrittlement is well predicted and provides more accurate interpolation and extrapolation of the surveillance data. Therefore, the use of the RG 1.99 ETC, which is known to underpredict embrittlement at high fluence, and the lack of planned surveillance data at high fluence, means that the safety margins are degraded commensurate with the level of underprediction in embrittlement.

6.4. Principle 4: Demonstration of Acceptable Levels of Risk

As described in Section 3, the staff conducted generic analyses to predict the levels of risk due to the underprediction of embrittlement at high fluence. These analyses demonstrated that for PTS events, the 95-percent TWCF is less than 1×10^{-6}/yr for operation to 80 years. For normal operating conditions, the CPF values calculated were below 1×10^{-6} for operation to 80 years except under the following circumstances. First, for cooldown transients that follow the licensed P-T curve, the CPF exceeded 1×10^{-6} when the ESD was more than 20 degrees F for 1/4T flaws and more than 50 degrees F for SSBFs. However, these flaws and transients are expected to occur with sufficiently low frequency that the calculated generic TWCF would be less than 1×10^{-6}/yr. (It should be noted that these analyses may not bound all plant-specific circumstances and do not consider plant-specific sources of uncertainty.)

For BWR leak tests, the calculations produced a CPF above 1×10^{-6} for ESD greater than 100 degrees F. However, based on the targeted sample evaluation in Section 3 and the filtered capsule data, no BWR plant is expected to have an ESD greater than 100 degrees F within 80 years of operation.

Additional analyses may be needed to determine whether acceptable generic risk is maintained for operation beyond 80 years.

6.5. Principle 5: Implementation of Defined Performance Measurement Strategies

As demonstrated in Section 4, plant-specific surveillance data at high fluence may not follow the trends extrapolated from the RG 1.99 ETC. Per Appendix H to 10 CFR Part 50, the purpose of a surveillance program is to monitor plant-specific RPV embrittlement behavior and verify that the RG 1.99 embrittlement trends are appropriate. Because Appendix H to 10 CFR Part 50 was originally developed at a time when operation beyond 40 years was not considered and it references an ASTM standard that does not call for the testing of surveillance capsules at high fluence, it is possible that few or no high-fluence plant-specific surveillance data will be available for plants operating to 80 years or beyond. Thus, adequate performance monitoring is not ensured under the current regulatory framework.

7. Summary of Risk-Informed Analysis

Based on the data and analyses presented in this paper, the staff has high confidence that currently operating plants remain safe and recent licensing actions remain valid. However, for long-term operation, the eventual degradation of safety margins and the potential lack of performance monitoring for the RPV, the most safety-significant passive component in the plant, are of concern. Even with the lack of operating experience and the past calculations that demonstrate low risk significance, the impact of the uncertainty described in this paper on the adequate safety margins and insufficient performance monitoring eventually will challenge reasonable assurance of adequate protection for long term operation. The RG 1.99 ETC (also given in 10 CFR 50.61) appears to provide adequate predictions of embrittlement to about 6×10^{19} n/cm^2, which is adequate for the many U.S. plants that will not reach this fluence level in their projected operating lives. However, in the long term, these models will increasingly underpredict embrittlement. Current projections suggest that up to 25 percent of the current U.S. units will surpass 6×10^{19} n/cm^2, and 10 percent will surpass 8×10^{19} n/cm^2, within 80 years of operation. Because the ETCs considered in this paper (RG 1.99 and ASTM E900) are empirically based, it may be necessary to update the formulations as higher fluence data become available; furthermore, as illustrated in Reference 12, some of the guidance in RG 1.99 (e.g., the surveillance data credibility criteria) may be inadequate.

Also, without appropriate performance monitoring, it is very difficult to adequately account for embrittlement in high-fluence plants. Because some licensees have tested capsules only early in the plant's operating life (e.g., representing much less than half of an 80-year operating period), their data are too limited to be extrapolated reliably to high fluence levels (see Figure ***12***). The resulting uncertainties are compounded by the underpredictions of the RG 1.99 ETC: although the ETC of RG 1.99 (and 10 CFR 50.61) is reasonably accurate at low fluence, extrapolation can still be of concern as the embrittlement trends early in operation may not continue throughout plant operation (see Figure **6**). Periodic performance monitoring is necessary to obtain adequate data to verify embrittlement trends later in a plant's operating life. In addition, to confirm data credibility and incorporate plant-specific data correctly within the ETC model, a proper fit is needed for datasets that include high-fluence data.

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.

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A. Appendix A: Background Information on Reactor Pressure Vessel Structural Integrity

A.1. Regulatory Requirements

A.1.1. Appendix A (General Design Criteria) and Appendix G to 10 CFR Part 50

In the event of an accident, the three principal barriers to fission product release are the reactor coolant system, which includes the reactor pressure vessel (RPV); the reactor fuel cladding; and the containment vessel(s). These barriers are intended to be independent and to provide defense in depth against fission product release. The U.S. Nuclear Regulatory Commission (NRC) regulations associated with each barrier provide reasonable assurance that they will independently fulfill their intended functions over the lifetime of the plant during both normal operation and design-basis accidents scenarios.

There is a mosaic of related regulatory requirements that specifically govern RPV structural integrity. Appendix A, “General Design Criteria for Nuclear Power Plants,” to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, “Domestic Licensing of Production and Utilization Facilities,” contains several related criteria [Ref. 1]. General Design Criterion (GDC) 10, “Reactor design,” requires that RPV design provide appropriate margin to ensure that fuel design limits are not exceeded during normal operation and anticipated operational occurrences. GDC 14, “Reactor coolant pressure boundary,” requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested to have extremely low probabilities of abnormal leakage, rapidly propagating failure, and gross rupture. GDC 31, “Fracture prevention of reactor coolant pressure boundary,” requires that the RPV be designed with sufficient margin to ensure that the vessel behaves in a nonbrittle manner and to minimize the probability of rapidly propagating fracture during both normal operation and postulated accident scenarios. GDC 31 also requires that the design reflect consideration of service temperatures and other conditions of the materials under operating and postulated accident conditions, as well as consideration of the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady-state, and transient stresses, and (4) size of flaws. Finally, GDC 32, “Inspection of reactor coolant pressure boundary,” requires that the RPV be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leak tight integrity, and an appropriate material surveillance program.

The pre-service requirements associated with these general criteria (i.e., those related to design, fabrication, erection, and pre-service testing) are practically fulfilled by adherence to American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code Section III, “Rules for Construction of Nuclear Facility Components,” Division 1, and, for a few plants, its predecessors [Ref. 2, 3, 4]. The requirement in GDC 14 for testing during operation is fulfilled, in part, through the inservice examination and inspection requirements of ASME BPVC, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” Division 1, “Rules for Inspection and Testing of Components of Light-Water-Cooled Plants” [Ref. 5].

Sections III and XI of the ASME BPVC are both required by 10 CFR 50.55a, “Codes and standards” [Ref. 6].

Specific requirements to address these general criteria over the life of the plant are provided within several other regulations. Appendix G, “Fracture Toughness Requirements,” to 10 CFR Part 50 specifies RPV fracture toughness requirements to provide adequate safety margins during normal operation, including anticipated operational occurrences and system hydrostatic tests, over the RPV’s service lifetime [Ref. 7]. The use of Appendix G to 10 CFR Part 50 is mandated by 10 CFR 50.60, “Acceptance criteria for fracture prevention measures for light-water nuclear power reactors for normal operation” [Ref. 8]. Appendix G to 10 CFR Part 50 specifies requirements for the RPV material’s minimum fracture toughness on the upper shelf (i.e., the temperature regime where failure occurs in a ductile manner), minimum temperature requirements, and pressure-temperature (P-T) limits that apply over the RPV’s operating life. The P-T limits, in particular, are intended to maintain adequate margins throughout the plant’s life. This objective requires that the P-T limits be adjusted to higher temperatures as the RPV experiences neutron embrittlement. P-T limit curves are explicitly calculated using ASME BPVC, Section XI, Appendix G [Ref. 5]. An equivalent margins analysis is performed in accordance with ASME BPVC, Section XI, Appendix K [Ref. 5], to evaluate materials that do not meet the upper-shelf requirements in Appendix G to 10 CFR Part 50. The equivalent margins analysis is reviewed and approved by the NRC. Again, Appendices G and K to ASME BPVC, Section XI, are both approved for use within 10 CFR 50.55a.

A.1.2. Pressurized Thermal Shock

In the early 1980s, the NRC became aware of the possibility, in pressurized-water reactors (PWRs), of a transient causing severe overcooling (i.e., thermal shock) concurrent with or followed by significant pressure in the RPV [Ref. 9]. Dubbed “pressurized thermal shock” (PTS), this transient was recognized as posing the most significant challenge to RPV integrity in PWRs, as it could cause rapid, or brittle, RPV failure. The PTS rule, 10 CFR 50.61, “Fracture toughness requirements for protection against pressurized thermal shock events” [Ref. 10], contains requirements and a method for demonstrating that the RPV’s material toughness remains acceptable to guard against PTS throughout the licensing period. The simplest way to demonstrate applicability, which all licensees currently follow, is to show that the RPV’s PTS reference temperature (which represents the material toughness at the plant’s end-of-license condition) is less than established screening limits.

The implementation of low-neutron-leakage reactor cores along with thermal shields to protect the RPV from gamma radiation, starting in the 1980s, helped decrease the rate at which the RPV’s material toughness was degrading with service time due to radiation embrittlement [Ref. 11]. Even so, some licensees found it challenging to meet the 10 CFR 50.61 toughness screening limits through the end of their licensing periods. A large-scale, risk-informed evaluation of PTS challenges led to the development of 10 CFR 50.61a, “Alternate fracture toughness requirements for protection against pressurized thermal shock events” [Ref. 12], which provides a risk-informed relaxation of the 10 CFR 50.61 screening limits, but requires that licensees conduct a one-time inspection of the RPV beltline region to demonstrate that the flaw density, distribution, and types are consistent with the flaw assumptions used in developing the technical basis for 10 CFR 50.61a [Ref. 13].

A.1.3. Regulatory Guide 1.99

The embrittlement trend curve (ETC) model in Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," issued May 1988 [Ref. 14], is part of the fabric of both Appendix G to 10 CFR Part 50 and 10 CFR 50.61, as they both require that the fracture toughness values used in the analyses must account for the effects of neutron radiation. The RG 1.99 ETC model is embedded in and required by the rule in 10 CFR 50.61. While Appendix G to 10 CFR Part 50 does not require the use of a specific ETC model, RG 1.99 is the approved guidance to account for embrittlement effects; in Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," dated July 12, 1988, the NRC staff stated that licensees should use RG 1.99 in all P-T limits and PTS analyses unless they could justify an alternative method [Ref. 15]. Hence, all licensees use RG 1.99 to determine their plant-specific P-T limits. The rule in 10 CFR 50.61a also requires the use of an ETC model, different from the RG 1.99 ETC, that was deemed to be the best available model at the time of the 10 CFR 50.61a rulemaking.

All ETC models have the same function within the rules: they are used to predict the fracture toughness of the RPV material at each plant. The PTS rules (10 CFR 50.61 and 10 CFR 50.61a) use the end-of-license embrittlement condition of the RPV (for PWRs only), whereas the P-T limits of Appendix G to 10 CFR Part 50 are typically updated periodically to ensure that they bound the current embrittlement condition of the RPV. For each potentially limiting material, the fracture toughness is predicted from that material's chemical composition, together with (for PTS) the end-of-license fast neutron fluence (where fast neutrons are defined as neutrons with energies greater than 1 megaelectron volt), or (for Appendix G P-T limits) a specific future neutron fluence. Data from credible surveillance testing are used to verify the accuracy of this prediction. If necessary, the licensee may adjust the ETC model to appropriately represent the surveillance data, or, if using 10 CFR 50.61a, may propose alternative end-of-license toughness values for staff approval using the surveillance data and not the ETC model.

A.1.4. Reactor Pressure Vessel Material Surveillance Program Requirements of Appendix H to 10 CFR Part 50

The regulation at 10 CFR 50.60 mandates that licensees meet the requirements of Appendix H, "Reactor Vessel Material Surveillance Program Requirements," to 10 CFR Part 50 [Ref. 16]. Appendix H first became effective on August 16, 1973. The introduction of the 1973 version of Appendix H stated, "These data will permit the determination of the conditions under which the vessel can be operated with adequate margins of safety against fracture throughout its service life." The 1973 version of Appendix H also stated that surveillance programs shall comply with American Society for Testing and Materials (ASTM; currently known as ASTM International) E185-73, "Standard Recommended Practice for Surveillance Tests for Nuclear Reactor

Vessels" [Ref. 17], although Appendix H modified some aspects of the standard, for example by providing specific capsule withdrawal schedules.

Beginning with a rule change in 1983, Appendix H incorporated by reference certain versions of ASTM E185, but none later than 1982. ASTM E185 incorporates the placement of samples of RPV materials into surveillance "capsules," which are inserted into the RPV and exposed to the same thermal and radiation environment as the RPV during plant operation. When properly located, the samples receive a higher neutron flux than the RPV itself, resulting in a "lead factor,"[1] so that the data provide an assessment of the future condition of the RPV. Periodic withdrawal and testing of the capsules enable monitoring of the embrittlement of the RPV material. ASTM E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels" [Ref. 18], describes the capsule withdrawal schedule in Table 1 of the standard as follows: "[T]he withdrawal schedule is in terms of effective full-power years (EFPY) of the vessel with a design life of 32 EFPY." When ASTM E185-82 was issued, plant operations targeted an availability factor of 80 percent, and 32 EFPYs corresponded to a design life and operating period of 40 years; operation beyond the initial 40-year license was not under general discussion in the technical community and was therefore not considered in the ASTM standard.

Since that time, the NRC staff has periodically considered updating Appendix H to incorporate the most recent editions of the ASTM standard, but has not ultimately pursued this option. For example, a 2019 analysis concluded that the use of 2016 editions of relevant standards (e.g., ASTM E185-16 [Ref. 19] and ASTM E2215-16 [Ref. 20]) was "suboptimal," since numerous conditions on the use of the standards "would be necessary to offset the unnecessary burden without a corresponding benefit to public health and safety and the environment." Thus, incorporation by reference of these standards was not recommended or pursued [Ref. 21].

A.2. Reactor Pressure Vessel Structural Integrity—Current Understanding and Ongoing Embrittlement Prediction and Surveillance Activities

The RPV's structural integrity is determined by the nexus of the applied loading challenges, the existence of cracks that could lead to a breach, and the RPV's fracture toughness properties. The earliest reactor design requirements provided significant margin to protect against both known and then unknown loading challenges. The RPV fabrication, preservice, and quality assurance provisions were intended to ensure that materials with significant flaws would not be placed in service. Since the first plants were constructed, the loading challenges and flaw distributions have been further evaluated and are now both reasonably well understood. Normal operation and accident loads have been assessed through thermal-hydraulic modeling that has been validated through large-scale experiments [e.g., Ref. 22, 23, a 24]. Flaw distributions have been assessed at each plant through ongoing inservice inspection, and extensive research has also been conducted to better understand the fabrication flaws that may exist in RPV materials that are not subject to inservice inspection [Ref. 25, 26]. Most

1 In some cases, capsule placement enables the samples to receive lower fluence levels than the RPV wall, thus producing a "lag factor."

importantly, loads and flaw distributions are expected to be relatively stable over time, notwithstanding significant operational changes such as “flexible operation” or load-following.

When the earliest U.S. plants were built, little was publicly known about how radiation embrittlement could decrease RPV fracture toughness. Now, after over 50 years of laboratory research augmented and validated by surveillance capsule testing, the effects of radiation embrittlement are much better understood [Ref. 27]. The mechanisms that lead to radiation embrittlement have been explained and linked to the important plant-specific causal factors such as the RPV material’s chemical composition, neutron fluence, and temperature; furthermore, both the mechanisms and the causal factors have been correlated with their effect on the fracture toughness of RPV materials.

At present, there is no quantitative physical model that adequately explains the relationship between the causal factors and the material’s fracture toughness. The relationship is therefore understood through empirical ETC models instead. The reliability of any empirical model is only defined and appropriate for use within the context of the scope, quantity and quality of the underlying data used to develop the model. Periodic assessment is therefore needed to ensure that the model appropriately addresses new data that subsequently becomes available, or new models should be developed.

RPV material surveillance programs are an essential complement to the ETC models. Their purpose is to periodically monitor changes in fracture toughness, in part to validate the general empirical ETC predictions using plant-specific embrittlement data. If necessary, the ETC is shifted to provide a best fit of the plant-specific data, so that future predictions better reflect plant-specific embrittlement characteristics. The combination of an accurate ETC model and plant-specific surveillance provides confidence the RPV toughness remains adequate during continued plant operation.

The current ETC models and surveillance programs were originally intended to provide plant-specific validation of embrittlement trends only to 40 years of plant operation. However, with subsequent license renewal (SLR) already approved for some plants, and more applications expected, recent activities have focused on providing embrittlement and surveillance information to 80 years of operation. These activities have focused on improving the accuracy of ETC models and ensure that they adequately represent the critical RPV materials as embrittlement increases during continued plant operation out to 80 years. The following sections summarize recent and ongoing activities to make ETC models more accurate and improve surveillance programs for this timeframe.

A.2.1. ASTM E900

The RG 1.99 ETC, which is woven throughout NRC regulations and plant licensing bases, was published in 1988. It was developed using 177 data points, which comprised all the relevant data available at that time. Since then, as both laboratory testing and surveillance capsule testing have provided new data, more complex ETC models have been developed that better account for plant-specific causal factors.

ASTM International has led a long history of ETC development. A consensus ETC model appears in ASTM E900, "Standard Guide for Predicting Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials," which was first published in 1983 and subsequently reviewed or revised in 1987, 1994, 2001, 2002, and 2015 as the ETC model matured [Ref. 28].

For the 2015 update, ASTM compiled and verified data on the transition temperature shift (i.e., increase in the 41-joule Charpy V-notch energy, or ΔT_{41J}) and yield strength increase, taken from the technical literature and from surveillance reports of operating and decommissioned light-water reactors worldwide. Attention was restricted to steels of the type already assessed by ASTM E900 (i.e., steels used in light-water reactors of Western design). The effort produced 4,438 data records on ΔT_{41J} or yield strength: 36 percent from PWR surveillance programs, 8 percent from boiling-water reactor (BWR) surveillance programs, and 56 percent from material test reactor research programs. From these data, ASTM defined the BASELINE data subset, which it would use to assess and then later recalibrate the □T41J trend curve equation. The BASELINE subset was restricted to commercial-grade steels for which the values of all necessary descriptive variables (copper, nickel, manganese, phosphorus, neutron fluence, neutron flux, temperature, and product form) were known, which had been exposed to neutron irradiation in a power reactor (i.e., thereby excluding data from material test reactor research programs), and whose embrittlement had been quantified through ΔT_{41J} measurements using full-size Charpy V-notch specimens. The BASELINE subset included 1,878 ΔT_{41J} surveillance data points from 13 countries: Brazil, Belgium, France, Germany, Italy, Japan, Mexico, The Netherlands, South Korea, Sweden, Switzerland, Taiwan, and the United States. This is an order of magnitude more data than were used to develop the RG 1.99 ETC.

The responsible ASTM subcommittee made every effort to ensure that the data used in its evaluation were accurate and its fidelity with respect to the source documents, and that information on chemistry and neutron fluence was the most up-to-date available. National experts checked the data from the largest national data collections (the United States, Japan, France, Germany, and Belgium). Additionally, data from Brazil, Italy, Mexico, South Korea, Sweden, Switzerland, and Taiwan were entered from the surveillance reports into a spreadsheet by one subcommittee member and checked by another.

The ETC developed by ASTM [Ref. 29] evolved over 4 years and relies on 32 empirically fitted parameters. It predicts ΔT_{41J} using seven variables:

- two exposure variables: neutron fluence, temperature
- four compositional variables: copper, nickel, manganese, phosphorus
- one categorical variable: product form

A.2.2. ASME Embrittlement Trend Curve Code Case

The fracture toughness models adopted by Nonmandatory Appendices A and G of the ASME BPVC and by recent ASME BPVC Section XI Code Cases all quantify the variation of toughness with temperature by positioning the allowable toughness curve using an index temperature (i.e., RT_{NDT}, RT_{To}, or T_0). For RPVs, the value of index temperature used must account for neutron irradiation embrittlement. The ASME BPVC is not prescriptive on how to adjust the index temperature for embrittlement, but provides the following guidance throughout various sections:

- The embrittlement shift is to be determined from surveillance specimens of the actual material and product form, irradiated according to the surveillance techniques of ASTM E185.
- The effects of neutron irradiation should be considered by shifting RT_{NDT} as a function of irradiation, based on data and methods acceptable to the regulatory authority having jurisdiction at the plant site.
- ASME BPVC, Section XI, Appendix G, allows three options for forecasting embrittlement trends: (1) from plant-specific surveillance data, (2) from an equation given in Appendix G, or (3) using "irradiation degradation models acceptable to the regulatory authority having jurisdiction at the plant site."

Because ASME has an international membership, it is progressing toward removing the phrase "acceptable to the regulatory authority having jurisdiction at the plant site," since the requirements for approval of codes and standards in regulations differ across countries.

To update the guidance on neutron irradiation embrittlement while accommodating the international community, ASME has begun developing a Section XI Code Case to define consistent requirements for evaluating embrittlement predictions. The effort aims to improve upon existing ASME guidance by making it comprehensive, consistent, clear, and current. While there are no plans for the Code Case to recommend a particular ETC model, it will provide appropriate acceptance criteria for demonstrating the adequacy of an ETC model. As of the writing of this report, ASME is developing the basis for the Code Case to address the following aspects of an ETC model:

- source of embrittlement data
- forecasting of embrittlement trends
- accounting for embrittlement in the interrelationships between various toughness properties
- accounting for uncertainties associated with embrittlement

The current schedule is for publication before 2023.

A.2.3. EPRI Pressurized-Water Reactor Supplemental Surveillance Program

The ASTM E900 BASELINE embrittlement database, described earlier, has limited U.S. power reactor surveillance data at neutron fluences beyond 4×10^{19} neutrons per square centimeter (n/cm^2) (E > 1 megaelectron volt[2]) for validating ETC model predictions. Extending plant

[2] This is assumed for all listed fluences, unless otherwise noted.

operation to 80 years under SLR or longer is projected to result in peak neutron fluences approaching $1x10^{20}$ n/cm^2 for some operating U.S. reactors.

To rectify this data deficiency, the Electric Power Research Institute (EPRI) has developed the PWR Supplemental Surveillance Program (PSSP), which will collect high-fluence data for benchmarking ETC models up to $1x10^{20}$ n/cm^2. The PSSP will irradiate two supplemental RPV surveillance capsules in two host PWR plants [Ref. 30]. These capsules contain previously irradiated PWR surveillance materials, so that neutron fluence objectives applicable to the current PWR fleet for at least 80 years of operation can be achieved within a reasonable 10-year period of additional irradiation.

Each capsule holds 144 Charpy-size specimens, for a total of 288 specimens. The capsules include 27 unique materials. The Charpy-size specimens were generally reconstituted from previously irradiated and tested specimens taken from plant-specific surveillance programs. The two plants hosting the capsules are Westinghouse-designed three-loop PWRs, which have a relatively high neutron flux of about $1.2x10^{11}$ n/cm^2/s in the capsule irradiation locations; over 10 years, this amounts to an additional fluence of about $3.5x10^{19}$ n/cm^2 on these specimens. The two PSSP capsules were placed in service in 2016 and 2018. Ten years was selected as a reasonable time frame that will produce sufficiently high neutron fluence, which is applicable to operation of the current PWR fleet to 80 years. The testing of these capsules will provide high-fluence transition temperature shift data to validate current ETCs or inform the development of new ones applicable to PWR operation in the high neutron fluence regime.

A.2.4. BWR Vessel and Internals Project Subsequent License Renewal Integrated Surveillance Program

The U.S. BWR power plants were designed and built with a surveillance capsule program to measure plant-specific embrittlement of the RPV. Until 2002, each plant in the fleet individually demonstrated compliance with Appendix H to 10 CFR Part 50. Since 2002, however, in lieu of plant-specific programs, the U.S. BWR fleet has relied on an integrated surveillance program (ISP) to provide fracture toughness data for RPV materials, and satisfy Appendix H requirements, in lieu of plant-specific programs. BWRVIP-86, Revision 1-A contains the details and basis for such a program [Ref. 31]. The current ISP was designed to support the surveillance needs of the BWR fleet through 60 years of operation.

Anticipating that some BWR licensees would request SLR to 80 years, EPRI began the development of an extension to the current ISP for SLR, with consideration of the following constraints:

- It is currently uncertain which plants, or how many, will pursue SLR.
- Some current ISP host plants may not pursue SLR.
- Plants pursuing SLR may have surveillance materials not representative of other plants and therefore are not suitable as host plants.

- Current host plants will likely not have additional capsules available for testing after the completion of the current ISP.
- Some representative surveillance materials were only in the supplemental surveillance program capsules, and no further capsules containing those materials are available for testing.
- Many BWRs, as well as ISP host plants have lag factors rather than lead factors.

BWRVIP-321-A [Ref. 32] details the industry's plan to extend the current ISP for the BWR fleet through the subsequent period of extended operation (80 years). The basis of this plan is that the original ISP test matrix, as approved through BWRVIP-86, Revision 1-A, provides adequate and appropriate surveillance data for all U.S. BWRs. Although some plants (including some host plants) may not pursue SLR, the approach is to ensure that all ISP representative materials have specimens that are irradiated to a neutron fluence that bounds the SLR neutron fluences of all target materials represented by that surveillance material. This plan will utilize existing data as much as possible. For some materials, specimens from capsules that were exposed to a wide range of neutron fluence levels have been tested, and some tested specimens have attained neutron fluences exceeding projected 80-year RPV fluences. Where there are gaps in data (e.g., where 80-year surveillance data do not exist), previously tested specimens will be further irradiated and reconstituted, as necessary, to generate additional surveillance data to support ISP participants that chose to pursue SLR. BWRVIP-321-A contains the details and basis for such a program [Ref. 32].

A.3. Probabilistic Fracture Mechanics Scoping Study on Effects of ETC Underprediction

As indicated in Section 3.5 of this report, the NRC staff performed a probabilistic fracture mechanics scoping study to evaluate the risk associated with potential RG 1.99 ETC underpredictions of radiation embrittlement. This study analyzed surveillance capsule data against specific underprediction levels, or embrittlement shift deltas (ESDs). More details on the scoping study and associated uncertainties follow.

A.3.1. Details on Probabilistic Fracture Mechanics Scoping Study

As indicated in Section 3.5 of this report, the staff used Version 16.1 of the Fracture Analysis of Vessels, Oak Ridge (FAVOR), code [Ref. 33, 34] to quantify the risks associated with a set of normal operating events, given the use of the RG 1.99 ETC to determine the normal-operation PWR and BWR P-T limits and leak test curves. For the analysis, the staff selected a model PWR plant (Palisades Nuclear Plant) and a model BWR plant (Hatch Nuclear Plant), which provide relatively conservative maximum embrittlement levels after 80 years of operation. The staff obtained the RPV geometry and embrittlement maps for both plants from the Reactor Vessel Integrity Database [Ref. 35]. The embrittlement maps provide the material chemistry, radiation flux value (which is used to determine the fluence at each location), and the initial RT_{NDT} for each base and weld RPV material.

The staff then used the RG 1.99 ETC to determine the maximum adjusted reference temperature for each plant at 72 EFPYs, at a depth of 1/4 of the RPV thickness (1/4T). The 72-EFPY fluence level was chosen as a conservative representation of an 80-year plant life, assuming an average capacity factor of 0.9. The maximum adjusted reference temperatures calculated were 234 degrees Fahrenheit (F) for the Palisades PWR and 93 degrees F for the Hatch BWR. These values were then used to develop the P-T limit curves for a presumed flaw with a crack depth of 1/4T and a surface crack length-to-depth ratio of 6 to 1, as required by ASME BPVC [Ref. 5]. Reference 36 gives more details on the FAVOR inputs, analysis assumptions, and the approach adopted to develop the model plants.

After establishing these baseline conditions, the staff assessed the effect of potential underpredictions by the RG 1.99 ETC in terms of the ESD, which is the difference between the embrittlements predicted by the ASTM E900-15 and RG 1.99 ETCs. The ASTM E900-15 ETC is assumed to represent the "true" RPV embrittlement after 80 years of operation. The staff introduced the ESD into the FAVOR analysis by simply adjusting the initial RT_{NDT} value to account for the difference between the two ETCs. The PWR and BWR ESD values were chosen separately by extrapolating individual surveillance capsule data to 80 years of operation using the RG 1.99 and ASTM E900-15 ETCs. The staff chose ESD values of -40 degrees F (a conservative ESD) and 0 degrees F, as well as temperatures representing the 50th-, 75th-, 95th-, and 99th-percentile ESD values. Note that much higher ESD values arise from these percentiles than either from the limiting materials in the PWR and BWR targeted sample results, or from capsule surveillance data collectively fitted to the ETC model. For example, the ESD was 193 degrees F at the 99th percentile of all capsule data.

The staff then assessed the probability of RPV failure for a 1/4T flaw with a surface crack length-to-depth ratio of 6 to 1, and for a SSBF (i.e., 0.03T for the PWR and 0.04T for the BWR) with various surface crack length-to-depth ratios. The 1/4T flaw was chosen because the ASME BPVC uses this flaw to determine P-T limit curves; also, it is meant to bound the largest credible flaw that could exist in service. This SSBF geometry was also evaluated because it represents a more credible flaw type that often leads to the highest probability of RPV failure due to thermal stresses at the interface between the stainless-steel cladding and the ferritic RPV material [Ref. 37].

For each combination of reactor type, flaw type, and ESD, the following cooldown and leak test transients were studied:

- BWR and PWR cooldown following the operational P-T limit curve (using a uniform cooldown rate of either 100 degrees F/hour (hr) or 50 degrees F/hr)
- BWR plant cooldown following the saturation curve
- BWR plant performing leak test following P-T limit curves (using a uniform cooldown rate of either 40 degrees F/hr or 100 degrees F/hr at the end of the leak test)
- PWR plant following cooldown curves for 42 actual plant cooldowns and leak tests

It is worth noting that during normal operation, a BWR plant does not cooldown following the P-T limit curve, but rather the saturation curve. The BWR P-T limit runs were therefore used primarily for comparison. The 42 actual PWR transients were normal-operation plant cooldown histories obtained from 17 Westinghouse PWRs. To accurately assess the cooldown risk, it would be necessary to know how representative these transients are, as well as the embrittlement level of the plant at the time of each cooldown; this information is unknown.

As noted in Section 3.5, the staff used the conditional probability of through-wall crack failure (CPF) as a conservative screening metric. In conventional probabilistic risk assessment, it is more common to use metrics such as core damage frequency (CDF) and large early-release frequency. Prior formal studies of RPV failure risk have used the through-wall crack frequency (TWCF) to conservatively represent CDF,[3] with a TWCF change greater than 1×10^{-6}/year (yr) used to determine if the change is regarded as significant [Ref. 13]. To convert CPF to TWCF for a given ESD, it is necessary to assess the probabilities of the assumed 1/4T flaw ($P|_{1/4T}$) and SSBF ($P|_{0.03T}$), along with the frequencies of a transient following the P-T limit curve ($F|_{P\text{-}T}$) and a normal-operation transient ($F|_{norm}$). The TWCF is then computed using the following equation:

$$\text{TWCF} = (F|_{norm})\,(P|_{1/4T})\,\text{CPF}|_{1/4T,norm} + (F|_{norm})\,(P|_{0.03T})\,\text{CPF}|_{0.03T,norm} + (F|_{P\text{-}T})\,(P|_{1/4T})\,\text{CPF}|_{1/4T,P\text{-}T} + (F|_{P\text{-}T})\,(P|_{0.03T})\,\text{CPF}|_{0.03T,P\text{-}T},$$

where the CPF subscripts indicate the combination of flaw type and transient.

Usually in an analysis, one of these four terms dominates and requires consideration. Here, the staff used CPF as a conservative risk surrogate for TWCF to avoid the complications and uncertainty of evaluating the various frequency functions during the scoping study. It is reasonable to use a CPF threshold of 1×10^{-6}, the value historically used for TWCF significance, as long as the product of the flaw probability and transient frequency functions is approximately 1/yr. In this study, the latter product is conservatively expected to range between 0.5/yr and 1×10^{-5}/yr depending on the specific combination of flaw depth and transient type, which makes the CPF metric acceptable for a generic safety evaluation.

A.3.2. Uncertainties Associated with Staff's Probabilistic Fracture Mechanics Scoping Study

While it was appropriate to use CPF as a conservative screening criterion in the scoping study to evaluate generic risk in terms of the ESD (i.e., ETC underpredictions), this metric is not appropriate for plant-specific evaluation, since there are large differences across plants' fabrication and operational practices that ultimately affect the TWCF. Furthermore, it is challenging to assess plant-specific TWCF values, because there are unquantified uncertainties in the frequency of challenging cooldown transients, the probability of occurrence of a critical flaw, and the CPF estimates themselves.

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[3] Using TWCF to estimate CDF is conservative, since a PRA may include other actions and mitigations that would decrease the true CDF.

Uncertainties in the CPF are inherent in the FAVOR analysis. As only one model BWR and one model PWR were simulated, the study considered only a single vessel geometry, embrittlement map, and set of fabrication characteristics (which determine vessel cladding stresses) for each plant type. The variables chosen for the model plants were representative and, in some cases, conservative. This is appropriate for a generic analysis, but it does not capture all the possible combinations of these variables, which determine the plant-specific risk.

Uncertainties in the cooldown transient stem from the allowable variability in cooldown procedures, which are affected by plant-specific design and operational constraints. The scoping study modeled the CPF for heatups and cooldowns following the ASME P-T limit curve. This is a conservative assumption because this curve almost always leads to the highest CPF. The P-T limits, by definition, are not to be exceeded during operation, and operational and administrative constraints provide additional controls to prevent these limits from being exceeded [Ref. 38]. Most importantly, plants are required to have a low-temperature overpressure protection system [Ref. 39, 40] to prevent P-T limit curve violations in the operating region most likely to cause failure of small inner-surface-breaking flaws.

While these systems and constraints are considered effective, there is still a theoretical frequency with which plants are expected to approach or exceed the P-T limits. It is challenging to calculate this frequency generically, because protection systems and other constraints are arrayed and utilized differently at different plants; a meaningful assessment of how frequently a particular plant may reach the P-T limits requires an indepth evaluation of the plant's configuration and operational history. As previously stated, the staff's scoping study analyzed 42 cooldown transients from 17 Westinghouse PWRs, using data that Westinghouse provided to the NRC. For these transients, the staff calculated CPF values less than 1×10^{-6}/yr. However, this sample represents less than 1 percent of the entire PWR cooldown transient population, and there is no information on how well the sample models the entire population of transients.

Finally, there are uncertainties in the frequency with which critical flaws occur. These arise from uncertainties in the RPV fabrication process and in preservice and inservice inspection. The fabrication process affects the frequency of pre-existing defects: RPV ingot production, the fabrication of plates or forgings from the ingot, the welding processes, and the cladding processes can all induce cracks and other defects that may challenge RPV integrity. Preservice inspection is required for all components, with radiography used for the plates, forgings, and welds, and dye penetrant testing used for the welds and cladding. Radiology is generally effective at finding volumetric defects such as porosity or lack of fusion, but less effective in identifying cracks. Dye penetrant, if performed correctly, is tailored to identify surface-breaking cracks unless they are tightly closed.

Inservice inspection is conducted using ultrasonic techniques. It is limited to welds and the immediately surrounding base material (e.g., 1.5T on either side of circumferential welds and 0.5T on either side of axial welds); these are volumetrically inspected every 10 years (see ASME BPVC, Section XI, Table IWB-2500-1 (B-A) [Ref. 5]), except for BWR circumferential welds, which have not been inspected since the late 1990s. PWR inspections are typically performed from the inner diameter of the RPV, while BWR inspections are performed from either the outer or the inner diameter. It should be noted that in outer-diameter inspections, it is challenging to detect flaws near the inner surface, which are the flaws producing the greatest risk of fracture from cooldown transients. The principal purpose of inservice inspections is to confirm that no cracking has occurred during service that may challenge RPV integrity. Because no such cracking has been identified to date, service-induced cracking of the RPV is not expected to be a significant consideration; only pre-existing fabrication flaws are likely to lead to RPV rupture.

Generic flaw distributions have been developed for use in FAVOR [Ref. 26]. These distributions were based on ultrasonic testing and destructive evaluation of representative areas of four constructed RPVs, analytical simulations of weld fabrication defects, and expert judgment to extend this information to the RPV population. There are separate distributions for the cladding, welds, and baseplates and forgings. This work notes that larger flaws are typically associated with repair welds, which are not always documented. Ideally, a plant-specific analysis would adapt these generic flaw distributions based on plant-specific fabrication and construction records; the existence of many undocumented repair welds could adversely bias such an analysis. Reference 26 also notes that the welding type, the RPV manufacturer, the vintage of the RPV, and the cladding process affect flaw distribution and density. Therefore, while the generic distributions are valuable, plant-specific flaw distributions may deviate from them because of variations in fabrication characteristics and the number of repair welds.

The 1/4T surface-breaking flaw evaluated in the scoping study was chosen for consistency with the flaw size assumed in ASME P-T limit curve evaluations; it bounds the fabrication flaws that may exist. The assumption is that the CPF associated with the bounding flaw is higher than the CPF for smaller, more realistic flaws. There is no evidence, nor any expectation, that such large flaws exist. However, relatively large flaws associated with repair welds near the inner diameter are plausible.

A.4. Recent Staff Evaluations of Reactor Pressure Vessel Structural Integrity Issues

Several issues observed domestically and internationally over the last 10 years have raised questions about RPV integrity. The NRC has assessed the risks associated with each of these issues independently, as summarized below.

A.4.1. Effects of Small Surface-Breaking Flaws

The first issue arose during the NRC's technical evaluation to support an industry-proposed risk-informed revision of Appendix G to 10 CFR Part 50, on determining P-T limit curves. The staff found that small surface-breaking flaws (SSBFs), which just penetrate the cladding and extend into the RPV shell, can potentially lead to high failure probabilities when cooldown follows the P-T limit curve. Such flaws could emanate from underclad cracks that may have developed during fabrication at some plants [Ref. 41]. The staff evaluated the generic safety implications of this situation using the LIC-504 process[4] [Ref. 42] and concluded that, while no immediate generic safety issue exists, the NRC staff should analyze the situation further. The principal basis of the staff's finding was that low-temperature overpressure protection systems and administrative limits made it very unlikely that plants would exceed the P-T limits at low temperatures. However, evaluation of BWR leak test and realistic cooldown transients (many of which were studied during the RG 1.99 ETC analysis) resulted in a few scenarios in BWRs where TWCF was over 1 x10^{-6}/yr. Additional analysis of this issue was completed [Ref. 37] but could not generically demonstrate lower failure probabilities than in the LIC-504 evaluation. More refined analyses are ongoing.

A.4.2. Quasilaminar Flaws Due to Hydrogen Flakes

In 2012, thousands of small (approximately 15-millimeter) quasilaminar indications were found in the Belgian nuclear power plants Doel 3 and Tihange 2. These flaws developed during fabrication of the RPV shell forgings, owing to insufficient hydrogen outgassing from the original steel ingot, which created "hydrogen flakes." All of the flaws were embedded, located principally within the inner half of the vessel (i.e., from near the inner surface to mid-thickness), and oriented either axially or with an axial inclination angle that was typically less than 15°.[5] They were discovered during an inspection for near-cladding defects using nondestructive evaluation techniques more sensitive than those used previously to inspect the same areas. The Belgian authorities conducted rigorous follow-on inspections, mechanical testing, and evaluations and demonstrated that these flaws had not likely grown during operation and did not challenge the structural reliability of the vessel [Ref. 43].

The NRC released the information notice IN 2013-19 [Ref. 44] on this issue, stating that while there was insufficient evidence to rule out the existence of similar flaws in U.S. RPV forgings, preservice inspection requirements should have identified any rejectable indications that could challenge structural integrity. Subsequent evaluations by the U.S. industry concluded that any such large number of quasilaminar flaws would have been detected and recorded with a high level of certainty during construction examinations. The industry also performed a bounding FAVOR computational evaluation of a beltline ring forging at the end of an 80-year license, from which it determined that the presence of thousands of flaws similar in size and type to those found in Doel 3 and Tihange 2 would have negligible impact on structural integrity [Ref. 45]. This issue was therefore not expected to significantly affect U.S. plant safety.

A.4.3. Nonconservatisms in Branch Technical Position 5-3

In early 2014, the NRC received a letter from AREVA stating that at least one position in the NRC's Branch Technical Position (BTP) 5-3 [Ref. 46] might be nonconservative [Ref. 47]. For plants constructed after August 15, 1973, the ASME BPVC requires licensees to conduct certain

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[4] LIC-504, "Integrated Risk-Informed Decision-Making Process for Emergent Issues," dated May 30, 2014, is an NRC internal office instruction that provides staff guidance on how to evaluate emergent issues using the risk-informed decision-making process.

[5] That is, the crack plane was largely parallel, not perpendicular, to the RPV axis. Perpendicular flaws typically provide a greater challenge to RPV structural integrity than parallel flaws.

material property tests to determine the RPV's unirradiated fracture toughness properties. BTP 5-3 provides guidance for plants constructed before August 15, 1973, that do not have all the test results required in later editions of the ASME BPVC. The intent of BTP 5-3 is to enable licensees with older plants to use their existing test results to estimate conservative values for missing test results, and then use this information to estimate the RPV's unirradiated fracture toughness.

The NRC staff performed an extensive deterministic and probabilistic evaluation of the issues raised in the AREVA letter, as well as all other BTP 5-3 regulatory positions, using surveillance information provided under Appendix H to 10 CFR Part 50 [Ref. 48]. Through the deterministic analyses, the staff verified that the position identified by AREVA was indeed nonconservative, and also identified several other nonconservative positions in BTP 5-3. The staff identified ways to add margins to make the existing positions conservative, but work performed by the NRC and EPRI demonstrated that existing margins in the PTS [Ref. 10, 12, 13] and P-T limit curve [Ref. 5] regulations were sufficient to bound the BTP 5-3 nonconservatisms for 60 years of operation.

The staff conducted probabilistic evaluations using the FAVOR code to evaluate the TWCF associated with both heatup and cooldown operational and PTS transients for 72 EFPYs or 80 years of plant operation. The approach was similar to that used in the RG 1.99 ETC study (see Section 3.4 and Reference 36), in that the staff estimated the TWCF associated with the change in risk due to a change in the fracture toughness. While the RG 1.99 ETC study [Ref. 36] considered changes to the nonconservative 72-EFPY toughness values as depicted by the ESD (Section A3.1), the BTP 5-3 effort considered the effects of nonconservatism in the initial fracture toughness values. The staff used a shift in the initial fracture toughness value for bounding PWR plants to demonstrate that the increase in PTS risk was insignificant. It also assessed the risk due to normal operations, using FAVOR and an approach like that of the RG 1.99 ETC study (see Section 3.5 and Reference 36). However, unlike the RG 1.99 ETC study, which characterized changes in the final toughness values and evaluated risk as a function of ESD, the BTP 5-3 effort increased the standard deviation of the initial fracture toughness distribution as a FAVOR input to estimate the change in risk associated with operational cooldown transients. The staff evaluated actual cooldown transients and transients following the limit curve and demonstrated that the BTP 5-3 nonconservatism causes no significant increase in generic risk up to 72 EFPYs. Based on the probabilistic analyses, the staff determined that it was not necessary to modify the existing nonconservative positions within BTP 5-3 [Ref. 48].

A.4.4. Effects of Carbon Macrosegregation in Large Forging Ingots

In 2016, regions of high carbon macrosegregation (CMAC) were discovered in the RPV upper and lower head in the Flamanville Evolutionary Power Reactor being constructed in France. High carbon content, which increases material yield strength, is typically detrimental to fracture toughness in ferritic materials. Subsequent evaluation by the French Nuclear Safety Authority (ASN) identified that large forgings produced by AREVA Creusot Forge and the Japanese Casting and Forging Corporation were potentially susceptible to CMAC [Ref. 49]. The French subsequently identified several other large inservice steam generator lower channel head forgings with these high-carbon regions and conducted a significant amount of inspection, material testing, and analytical evaluation [Ref. 50], to demonstrate that both the Evolutionary Power Reactor RPV and the inservice channel head forgings were acceptable for continued service, albeit with some operational restrictions for the plants whose channel head forgings were most affected [Ref. 51, 52].

The NRC used the LIC-504 process to evaluate the potential impact of this issue on U.S. plants in a final safety assessment [Ref. 53]. The staff's review of plant fabrication information found that no U.S. plants contain forgings made by the Japanese Casting and Forging Corporation, while 17 U.S. plants have pressure boundary components fabricated using forgings from AREVA Creusot Forge. The staff determined that the likelihood of CMAC was low for approximately 70 percent of these components. For the remaining 30 percent, there was insufficient documentation to independently assess the likelihood of CMAC, although it was not expected to be high given the known fabrication history. To assess the likelihood of failure due to CMAC, the staff estimated a maximum carbon content to bound the decrease in fracture toughness and examined the results of the following evaluations: an initial, semiquantitative staff evaluation; the testing and analysis conducted in France that formed the basis of the ASN regulatory decisions; and the results of EPRI-sponsored analysis to address the safety significance. In particular, the EPRI work used the FAVOR code for a generic probabilistic fracture mechanics analysis, which bounded the potentially affected components to verify that the TWCF was less than 1×10^{-6}/yr. The NRC staff concluded that no immediate action was warranted but recommended that the NRC continue to monitor the domestic and international activities on CMAC and evaluate new information as needed.

A.4.5. Uncertainties Associated with Prior Staff Evaluations

The staff used similar approaches to evaluate the RPV integrity issues described in Sections A.4.1–A.4.4. It first used FAVOR to determine the conditional failure probability for the effects of either decreased fracture toughness (for CMAC and BTP 5-3) or potential cracking (for SSBFs and hydrogen cracking). It then coupled the FAVOR results to an analysis demonstrating the relative rarity of the loading events (i.e., PTS or operation on the P-T limit curves) that are typically associated with the highest conditional failure probability and, therefore, pose the greatest challenge to RPV integrity. The staff evaluations all demonstrated that the TWCF for each issue independently was less than the commonly accepted threshold for core damage frequency [Ref. 54]. The evaluations were performed generically for a few representative plants selected because they have relatively high RT_{NDT} or RT_{PTS} values [Ref. 48]; the staff selected these plants to minimize the fracture toughness in the simulations, with the goal of bounding the risk. This approach leads to the highest risk (due to toughness effects) in PTS evaluations, but not necessarily in P-T limit evaluations, since the P-T limit curve is calculated to account for material toughness, so as to promote consistent risk and safety margins regardless of the absolute material toughness.

Because these evaluations were generic and addressed each issue independently, plant-specific uncertainties are inherent in the results. None of the FAVOR analyses realistically considered plant-specific effects; only the BTP 5-3 evaluation did so. (The BTP 5-3 evaluation assessed the potential decrease in material toughness in affected plants to demonstrate that sufficient margin remained in P-T limit and PTS evaluations, because either the BTP 5-3 positions did not affect the limiting material, there was additional uncredited toughness margin greater than the BTP 5-3 nonconservatism, or the material remained below accepted PTS screening limits.) Furthermore, since the evaluations addressed the four issues independently, they did not consider their possible combined or synergistic effects. Independent assessment is appropriate for a generic analysis, but combinations of issues may lead to increased risk at specific plants.

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