U.S. NUCLEAR REGULATORY COMMISSION

REGULATORY GUIDE 1.230, REVISION 0

Issue Date: September 2018
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REGULATORY GUIDANCE ON THE ALTERNATE PRESSURIZED THERMAL SHOCK RULE

A. INTRODUCTION

Purpose

This regulatory guide (RG) describes a method that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable to permit use of the alternate fracture toughness requirements for protection against pressurized thermal shock (PTS) events for pressurized-water reactor (PWR) reactor pressure vessels (RPVs) in Title 10 of the *Code of Federal Regulations*, Part 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. 1), Section 50.61a, "Alternate Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events" (10 CFR 50.61a).

Applicability

This RG applies to each holder of an operating license for a pressurized-water nuclear power reactor whose construction permit was issued before February 3, 2010, and whose RPV was designed and fabricated to the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), 1998 edition or earlier (Ref. 2).

Applicable Rules and Regulations

- 10 CFR 50.61a provides alternate fracture toughness requirements for protection against PTS events for PWR RPVs to the requirements in 10 CFR 50.61, "Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events."

Written suggestions regarding this guide or development of new guides may be submitted through the NRC's public Web site in the NRC Library at http://www.nrc.gov/reading-rm/doc-collections/, under Document Collections, in Regulatory Guides, at http://www.nrc.gov/reading-rm/doc-collections/reg-guides/contactus.html,

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Related Guidance

- RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" (Ref. 3), provides calculations and measurement procedures that are acceptable to the NRC staff for determining pressure vessel fluence.

Purpose of Regulatory Guides

The NRC issues RGs to describe to the public methods that the staff considers acceptable for use in implementing specific parts of the agency's regulations, to explain techniques that the staff uses in evaluating specific problems or postulated accidents, and to provide guidance to applicants. Regulatory guides are not substitutes for regulations and compliance with them is not required. Methods and solutions that differ from those set forth in RGs will be deemed acceptable if they provide a basis for the findings required for the issuance or continuance of a permit or license by the Commission.

Paperwork Reduction Act

This RG provides guidance for implementing the mandatory information collections in 10 CFR Part 50 that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et. seq.). These information collections were approved by the Office of Management and Budget (OMB), under control number 3150-0011. Send comments regarding this information collection to the Information Services Branch, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by e-mail to Infocollects.Resource@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0011), Office of Management and Budget, Washington, DC

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The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.

B. DISCUSSION

Reason for Issuance

This RG is being issued to describe a method that the staff of the NRC considers acceptable to meet the alternate fracture toughness requirements for protection against pressurized thermal shock (PTS) events for pressurized-water reactor (PWR) reactor pressure vessels (RPVs) in 10 CFR 50.61a, “Alternate Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events.” The alternate PTS requirements are based on updated analysis methods, and are desirable because the requirements in 10 CFR 50.61, “Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events,” are based on overly conservative probabilistic fracture mechanics (PFM) analyses.

Background

The RPV in a nuclear power plant is exposed to neutron radiation during normal operation. Over time, the RPV steel becomes progressively embrittled in the region adjacent to the core. If a RPV were to have a pre-existing flaw of critical size and certain severe system transients were to occur, this flaw could propagate rapidly through the RPV, resulting in a through-wall crack. The severe transients of concern, known as PTS events, are characterized by rapid cooling (i.e., thermal shock) of the internal RPV surface that may be combined with repressurization. The simultaneous occurrence of critical-size flaws, embrittled steel, and a severe PTS transient is a low probability event.

The NRC established the requirements for fracture toughness in 10 CFR 50.61 and many operating plants were licensed to meet these requirements. However, after additional information became available, it was recognized that the initial requirements established in 10 CFR 50.61 were based on overly conservative assumptions. In the *Federal Register* dated January 4, 2010 (Ref. 4), the NRC amended its regulations to provide alternate fracture toughness requirements for protection against PTS events for PWR RPVs. The alternate requirements contained in 10 CFR 50.61a maintain adequate safety while reducing regulatory burden for a PWR licensee who expects to exceed the requirements contained in 10 CFR 50.61 before the expiration of its license. A PWR licensee may choose to apply the provisions of 10 CFR 50.61a as a voluntary alternative to complying with the requirements of 10 CFR 50.61.

The “Alternate PTS Rule” contained in 10 CFR 50.61a provides revised PTS screening criteria in the form of an embrittlement reference temperature, RT_{MAX-X}, which characterizes the RPV material’s resistance to fracture initiating from flaws based on more comprehensive analysis methods.

This document contains four regulatory positions that provide guidance concerning methods that the NRC staff considers acceptable for meeting the various criteria within the Alternate PTS Rule. These four regulatory positions are described below:

1. *Criteria relating to the date of construction and design requirements*: The Alternate PTS Rule is applicable to PWR licensees whose construction permits were issued before February 3, 2010, and whose RPVs were designed and fabricated in accordance with Section III of the ASME Code, 1998 Edition or earlier. The purpose of this applicability restriction is that the structural and thermal hydraulic analyses that established the basis for the Alternate PTS Rule embrittlement limits only represented plants constructed before this date. Licensees whose construction permits were issued after February 3, 2010, or with reactor vessels that were not designed and fabricated to the 1998 Edition or earlier of the ASME code must apply for and receive a specific exemption from 10 CFR 50.61a(b) via 10 CFR 50.12 in order to utilize the alternate 10 CFR 50.61a criteria.

2. *Criteria relating to the evaluation of plant-specific surveillance data*: The Alternate PTS Rule includes three statistical tests that should be performed on RPV surveillance data to determine whether the surveillance data are sufficiently close to the predictions of the embrittlement trend curve (ETC) used in 10 CFR 50.61a such that the predicted values based on the ETC are valid for use. Position 2 of this document provides guidance by which licensees can assess plant-specific data to the 10 CFR 50.61a ETC using statistical tests.

3. *Criteria relating to ISI data and NDE requirements*: The Alternate PTS Rule describes a number of tests and conditions on the collection and analysis of inservice inspection (ISI) data and requirements for nondestructive examination (NDE) that are intended to provide reasonable assurance that the distribution of flaws assumed to exist in the PFM calculations that provided the basis for the fracture resistance limits defined in 10 CFR 50.61a (defined in terms of RT_{MAX-X} values) provides an acceptable model of the population of flaws in the RPV of interest. Position 3 of this document provides guidance by which licensees can satisfy these criteria.

4. *Criteria relating to alternate limits on embrittlement.* The Alternate PTS Rule provides embrittlement criteria in the form of RT_{MAX-X} limits, as specified in Table 1 of 10 CFR 50.61a. Position 4 of this document describes an alternate procedure by which licensees can assess their plant-specific through-wall cracking frequency (TWCF) for cases where the RT_{MAX-X} limits are not met.

Further details and the technical background associated with the guidance provided in this document may be found in NUREG-2163, "Technical Basis for Regulatory Guidance on the Alternate Pressurized Thermal Shock Rule" (Ref. 5).

Harmonization with International Standards

The NRC staff reviewed guidance from the International Atomic Energy Agency and did not identify any related standards that provided useful guidance to NRC staff, applicants, or licensees.

Documents Discussed in Staff Regulatory Guidance

This regulatory guide endorses, in part, the use of one or more codes or standards developed by external organizations, and other third-party guidance documents. These codes, standards and third-party guidance documents may contain references to other codes, standards or third-party guidance documents ("secondary references"). If a secondary reference has itself been incorporated by reference into NRC regulations as a requirement, then licensees and applicants must comply with that standard as set forth in the regulation. If the secondary reference has been endorsed in a regulatory guide as an acceptable approach for meeting an NRC requirement, then the standard constitutes a method acceptable to the NRC staff for meeting that regulatory requirement as described in the specific regulatory guide. If the secondary reference has neither been incorporated by reference into NRC regulations nor endorsed in a regulatory guide, then the secondary reference is neither a legally-binding requirement nor a "generic" NRC-approved acceptable approach for meeting an NRC requirement. However, licensees and applicants may consider and use the information in the secondary reference, if appropriately justified, consistent with current regulatory practice, and consistent with applicable NRC requirements.

C. STAFF REGULATORY GUIDANCE

1. Criteria Relating to the Date of Construction and Design Requirements

The regulation in 10 CFR 50.61a(b) states that the Alternate PTS Rule applies to holders of an operating license for a PWR whose construction permit was issued before February 3, 2010, and whose reactor vessel was designed and fabricated to the ASME Boiler and Pressure Code, 1998 Edition or earlier. If a licensee does not fit within this category (e.g., a licensee whose construction permit was issued after February 3, 2010), the provisions of 10 CFR 50.61a may not be used as an alternative to the requirements of 10 CFR 50.61 unless the licensee applies for and obtains a specific exemption under 10 CFR 50.12 from the 10 CFR 50.61a(b) prohibition.

2. Criteria Relating to the Evaluation of Plant-Specific Surveillance Data

This regulatory position describes a procedure by which licensees can assess their plant-specific material surveillance data using the three statistical tests required by Paragraph (f)(6) of 10 CFR 50.61a. If the criteria for all three statistical tests are satisfied for all beltline materials (Ref. 6), then the $RT_{MAX\text{-}X}$ screening criteria in Table 1 of 10 CFR 50.61a can be used without modification. Conversely, if any of the criteria from the three statistical tests required by 10 CFR 50.61a are not satisfied for any beltline material, then additional action is required to justify the use of the $RT_{MAX\text{-}X}$ screening criteria in Table 1 of 10 CFR 50.61a as required by Paragraph (f)(6)(vi) of 10 CFR 50.61a.

Use the following equations when performing the procedure described in this regulatory position. Equation numbers identical to those in Paragraph (g) of 10 CFR 50.61a are used for consistency. Equation (5) is the embrittlement trend curve (ETC).

$$\Delta T_{30} = MD + CRP \tag{5}$$

$$MD = \mathrm{A} \cdot (1 - 0.001718\, T_c)(1 + 6.13\, \mathrm{P}\, \mathrm{Mn}^{2.471})\varphi t_e{}^{0.5} \tag{6}$$

$$\text{where: } A = \begin{cases} 1.14 \times 10^{-7} \text{ for forgings} \\ 1.561 \times 10^{-7} \text{ for plates} \\ 1.417 \times 10^{-7}, \text{ for welds} \end{cases}$$

$$CRP = \mathrm{B} \cdot \left[1 + 3.77\, \mathrm{Ni}^{1.191}\right] \mathrm{f}(\mathrm{Cu_e}, \mathrm{P}) \cdot \mathrm{g}\left(\mathrm{Cu_e}, \mathrm{Ni}, \varphi t_e\right) \tag{7}$$

$$\text{where: } B = \begin{cases} 102.3 \text{ for forgings} \\ 102.5 \text{ for plates in non-Combustion Engineering vessels} \\ 135.2 \text{ for plates in Combustion Engineering vessels} \\ 155.0 \text{ for welds} \end{cases}$$

$$\mathrm{Cu_e} = \begin{cases} 0 \text{ if Cu} \le 0.072 \\ min\left[\mathrm{Cu}, \mathrm{MAX}(\mathrm{Cu_e})\right] \text{ if Cu} > 0.072 \end{cases}$$

$$\mathrm{MAX}(\mathrm{Cu_e}) = \begin{cases} 0.243 \text{ for Linde 80 welds} \\ 0.301 \text{ for all other materials} \end{cases}$$

$$f(Cu_e, P) = \begin{cases} 0 \text{ for } Cu \le 0.072 \\ (Cu_e - 0.072)^{0.668} \text{ for } Cu > 0.072 \text{ and } P \le 0.008 \\ [(Cu_e - 0.072) + 1.359(P - 0.008)\,]^{0.668} \text{ for } Cu > 0.072 \text{ and } P > 0.008 \end{cases}$$

$$g(Cu_e, Ni, \varphi t_e) = 0.5 + 0.5 \times tanh\left\{\frac{log_{10}(\varphi t_e) + 1.1390 \times Cu_e - 0.448 \times Ni - 18.120}{0.629}\right\}$$

$$\varphi t_e = \begin{cases} \varphi t, \text{ for } \varphi \ge 4.39\ 10^{10} \\ \varphi t \left(\frac{4.39\ 10^{10}}{\varphi}\right)^{0.259} \text{ for } \varphi < 4.39\ 10^{10} \end{cases}$$

The variables in these equations and their units are shown in Table 1.

Table 1. Variables, Symbols, and Units used in Eqs. (5) – (7).

Variable	Type	Symbol	Units
Transition temperature shift	Result of eq. (5)	ΔT_{30}	°F
Neutron Fluence (E > 1 MeV)	Input	φt	n/cm^2
Effective Neutron Fluence (E > 1 MeV)[1]	Calculated	φt_e	n/cm^2
Neutron Flux (E > 1 MeV)	Input	φ	n/cm^2/sec
Irradiation Temperature	Input	T_C	°F
Copper content	Input	Cu	weight %
Effective Copper content[2]	Calculated	Cu_e	weight %
Nickel content	Input	Ni	weight %
Manganese content	Input	Mn	weight %
Phosphorus content	Input	P	weight %

Notes:
1. Effective neutron fluence is a value of fluence modified by flux, see eq. (7).
2. The copper available to influence embrittlement is limited to this effective value, see eq. (7).

As with any equation calibrated to empirical data, inaccuracies have a greater tendency to occur at the extremes, or beyond the limits, of the calibration dataset. Users of Eqn. (5) should therefore exercise caution when applying it to conditions near to or beyond the extremes of its calibration dataset, which appear in Table 2.

Table 2. Independent Variables in the Eqn. (5) ETC and the Ranges and Mean Values of the Calibration Dataset

Variable	Symbol	Units	Values of Surveillance Database			
			Average	Standard Deviation	Minimum	Maximum
Neutron Fluence (E > 1 MeV)	ϕt	n/cm^2	1.24E+19	1.19E+19	9.26E+15	1.07E+20
Neutron Flux (E > 1 MeV)	ϕ	n/cm^2/sec	8.69E+10	9.96E+10	2.62E+08	1.63E+12
Irradiation Temperature	T_C	°F	545	11	522	570
Copper Content	Cu	weight %	0.140	0.084	0.010	0.410
Nickel Content	Ni	weight %	0.56	0.23	0.04	1.26
Manganese Content	Mn	weight %	1.31	0.26	0.58	1.96
Phosphorus Content	P	weight %	0.012	0.004	0.003	0.031

Step 1: Assess the Availability of Surveillance Data and Collect Information to Support the Statistical Analysis

Paragraph (f)(6)(i) of 10 CFR 50.61a requires that the licensee assess the suitability of its surveillance data. Licensees who utilize this guidance should assess its surveillance data as follows:

(a) For each shell material in the RPV beltline region, identify all surveillance data of the same heat of material from the plant being assessed, and from any other reactor that is operating, or that has previously operated, under a license issued by the NRC.

(b) For each beltline material identified in Step 1(a), count the number of values of shift produced by irradiation in the Charpy V-notch (CVN) transition temperature at the 30 ft-lb energy level (ΔT_{30}). When counting data for individual plates and forgings, ΔT_{30} obtained for different notch orientations should be treated as part of the same data set.

i. If there are fewer than three ΔT_{30} values, each measured at a unique fluence value, for a material then surveillance tests are not required for that material. The remaining steps of this procedure may be ignored and the ETC in 10 CFR 50.61a (Equation (5)) may be used.

ii. If there are three or more ΔT_{30} values, each measured at a unique fluence value, for a material then statistical surveillance tests are required for that material. The remaining steps of this procedure should be followed.

(c) For all materials remaining after Step (1)(b), assemble the following information:

- heat identification
- plant identification
- capsule identification
- product form
- notch orientation
- the unirradiated reference temperature, $RT_{NDT(U)}$
- ΔT_{30}
- Charpy-V notch energy data used to estimate ΔT_{30}
- fluence

- flux
- operating time
- cold leg temperature under normal full-power operating conditions (T_c)
    - Note: T_c (°F) is determined as the time-weighted average coolant temperature of the reactor coolant system cold leg covering the time period when the capsule was in the reactor.
- copper (Cu) content
- nickel (Ni) content
- phosphorus (P) content
- manganese (Mn) content
- citation (that is, the reference, or references, for all of the above-stated information)

The values of, Cu, Ni, P, and Mn must represent the best estimate values for the material (10 CFR 50.61a(f)(3)). For a plate or forging, the best estimate value is normally the mean of the measured values for that plate or forging. For a weld, the best estimate value is normally the mean of the measured values for a weld deposit made using the same weld wire heat number as the critical vessel weld. If these values are not available, either the upper limiting values given in the material specifications to which the vessel material was fabricated, or conservative estimates (i.e., mean plus one standard deviation) based on generic data should be used. Such conservative estimates for phosphorus and manganese appear in Table 4 of 10 CFR 50.61a. Similarly, upper bound estimates for Cu and Ni are provided in 10 CFR 50.61.

Step 2: Perform Statistical Assessments of the Surveillance Data

For each material remaining after Step (1)(b)(ii), determine if each of the following three statistical tests are met:

(a) Mean Test

Paragraph (f)(6)(ii) of 10 CFR 50.61a requires that the licensee perform a statistical mean test. Licensees utilizing this guidance should perform the statistical mean test as follows:

i. Determine the mean deviation from the data from the ETC using the following equation to calculate the residual (*r*) for each surveillance datum identified in Step 1:

$$r = \Delta T_{30(Measured)} - \Delta T_{30(Predicted)} \qquad (8)$$

where the measured ΔT_{30} represents the shift in CVN transition temperature at the 30 ft-lb energy level produced by irradiation for each datum identified in Step 1, and the predicted ΔT_{30} is estimated by Equation (5) using the best-estimate composition for the surveillance material and the best-estimate exposure values for the plant from which the companion measured ΔT_{30} value was obtained.

ii. Estimate the mean residual (r_{mean}):

$$r_{mean} = \frac{1}{n}\sum_{i=1}^{n}\{r_i\} \qquad (9)$$

where *n* is the number of data points in the specific data set,

iii. Estimate the maximum credible heat-average residual (r_{max}):

$$r_{max} = \frac{2.33\sigma}{\sqrt{n}} \tag{10}$$

where σ is from Table 3.

iv. If r_{mean} exceeds r_{max}, then the mean test is not satisfied. If r_{mean} is less than or equal to r_{max}, then the mean test is satisfied; in this case proceed to Step 2(b).

Table 3. Standard Deviation of Residuals about Eq. (5).

Product Form	**Standard Deviation (°F)**	
	Cu ≤ 0.072 wt %	**Cu > 0.072 wt %**
Weld	18.6	26.4
Plate		21.2
Forging		19.6

(b) Slope Test

Paragraph (f)(6)(iii) of 10 CFR 50.61a requires that the licensee perform a statistical slope test. Licensees who utilize this guidance should perform the statistical slope test as follows:

i. Using the method of least squares, estimate the slope (*m*) of the ETC model residuals (i.e., the r values, from Eq. (8)) plotted as a function of the base 10 logarithm of neutron fluence for the specific data set. Also estimate the standard-error of the estimated value of slope, se(m).

ii. Estimate the T-statistic for *m* as follows:

$$T_{SURV} = \frac{m}{se(m)} \tag{11}$$

Determine the critical value of T (T_{CRIT}) from the rightmost column in Table 4.

iii. For surveillance data sets with greater than 15 data points, the T_{MAX} value should be calculated using Student's T distribution with a significance level (α) of 1 percent for a one-tailed test.

iv. If T_{SURV} exceeds $T_{CRIT,}$ then the slope test is not satisfied. If T_{SURV} is less than or equal to T_{CRIT}, then the slope test is satisfied; in this case proceed to Step 2(c).

Table 4. α = 1% Student's-T Values.

Number of ΔT_{30} Values, n	n-2	One-Tailed T_{CRIT} (1%, n-2)
3	1	31.82
4	2	6.96
5	3	4.54
6	4	3.75
7	5	3.36
8	6	3.14
9	7	3.00
10	8	2.90
11	9	2.82
12	10	2.76
13	11	2.72
14	12	2.68
15	13	2.65

(c) Outlier Test

Paragraph (f)(6)(iv) of 10 CFR 50.61a requires that the licensee perform a statistical outlier test. Licensees who utilize this guidance should perform the statistical outlier test as follows:

i. Estimate the normalized residual (r^*), for each of the n observations in the ΔT_{30} dataset:

$$r^* = \frac{r}{\sigma} \tag{12}$$

where r is defined using Equation (8) and σ is from Table 3.

ii. Find the largest and second largest r* values; designate these r^*_1 and r^*_2, respectively.

iii. Find the limit values of $r_{LIMIT(1)}$ and $r_{LIMIT(2)}$ corresponding to the dataset size n in Table 5.

iv. If $r^*_1 \leq r_{LIMIT(1)}$ and $r^*_2 \leq r_{LIMIT(2)}$ then the dataset satisfies the outlier test; otherwise it does not. In either case proceed to Step 2(d).

Table 5. $\alpha = 1\%$ Threshold Value for the Outlier Test.

n	$r_{LIMIT(2)}$	$r_{LIMIT(1)}$
3	1.55	2.71
4	1.73	2.81
5	1.84	2.88
6	1.93	2.93
7	2.00	2.98
8	2.05	3.02
9	2.11	3.06
10	2.16	3.09
11	2.19	3.12
12	2.23	3.14
13	2.26	3.17
14	2.29	3.19
15	2.32	3.21
17	2.37	3.24
26	2.53	3.36
64	2.83	3.62

(d) Outcome of Step 2

i. Assessment: If all surveillance materials meeting the data quantity requirements of Step (1)(b)(ii) satisfy all three statistical surveillance tests of Steps (2)(a), (2)(b), and (2)(c), then the RT_{MAX-X} screening criteria in Table 1 of 10 CFR 50.61a can be used without modification. The values of ΔT_{30} used in estimating the RT_{MAX-X} values should be based on the ETC defined by Equation (5) using best-estimate input values for the plant and plant materials being assessed, and should not be modified based on surveillance data. In the event that any of the statistical tests in Steps (2)(a), (2)(b), and (2)(c) are not satisfied, 10 CFR 50.61a(f)(6)(vi) stipulates that:

> … the licensee shall review the data base for that heat in detail, including all parameters used in [the ETC] and the data used to determine the baseline Charpy V-notch curve for the material in an unirradiated condition. The licensee shall submit an evaluation of the surveillance data to the NRC and shall propose ΔT_{30} and RT_{MAX-X} values, considering their plant-specific surveillance data, to be used for evaluation relative to the acceptance criteria of this rule. These evaluations must be submitted for review and approval by the Director in the form of a license amendment…

The following guidance provides information for these additional evaluations.

ii. Factors to Consider When the Step 2 Statistical Tests are not Satisfied: When any of the statistical tests are not satisfied, values of ΔT_{30} predicted using Equation (5) may under-estimate the embrittlement magnitude. Therefore, review of the data for that heat, including all parameters used in Equation (5) and the data used to determine the CVN curve for the material in the unirradiated condition, should be performed. The most appropriate approach may not be a heat-specific adjustment of the ETC predictions in all cases. For example, statistically significant differences may indicate situations where the available data (i.e., the measured ΔT_{30} values and/or the composition and exposure values associated with the measured ΔT_{30} values) may not be accurate, thereby making adjustment of the ETC predictions to match these data unnecessary. Assessment of the data should consider, but not be limited to, the following factors:

- $RT_{NDT(U)}$ value: A records investigation of the $RT_{NDT(U)}$ value, and/or the performance of additional testing of archival material, may provide a more accurate estimate of $RT_{NDT(U)}$, which may explain the reason for not satisfying the mean and/or outlier tests.

- Irradiated T_{30} values: While most CVN energy *vs*. temperature curves (from which T_{30} values are estimated) are based on ≈8 to 12 individual measurements, some data sets are more limited, which can lead to increased uncertainty in the values of T_{30}. In the event that any of the statistical tests are not satisfied, a review of the individual CVN energy *vs*. temperature curves may help reveal the cause.

- Composition and exposure variables: The input variables to Equation (5) are subject to variability and are often based on limited data. However, the predictions of Equation (5) are very sensitive to the value of the input variables, particularly Cu content, fluence, temperature, and Ni content. If a sensitivity analysis reveals that small variations of the values input to Equation (5) explain the cause of not satisfying the statistical tests, this might indicate that more refined information concerning input values (e.g., additional measurements) are necessary, and may form the basis for proposing ΔT_{30} and RT_{MAX-X} values considering the plant-specific surveillance data. Specific limits are not provided; these should be justified on a case-specific basis.

- Notch orientation: The T_{30} values for plate and forging materials are sensitive to the orientation of the notch in the CVN specimens relative to the primary working directions of the plate or forging materials. Differences in notch orientation between the unirradiated T_{30} values and the T_{30} values for all of the irradiated specimens could help to explain why the mean test is not satisfied. Similarly, differences in notch orientations between the unirradiated T_{30} values and the T_{30} values for the irradiated specimens in a single capsule could help to explain why the outlier is not satisfied. In these situations, the outcome of a records search or metallurgical investigation of the tested specimens may provide part of the basis for proposing ΔT_{30} and RT_{MAX-X} values considering the plant-specific surveillance data.

- Comparative trends analysis: In addition to CVN specimens, surveillance capsules also contain tensile specimens. Like ΔT_{30}, the increase in yield strength with irradiation (ΔYS) also follows predictable trends. If ΔYS data for a particular material that failed the statistical tests follows the trends exhibited by ΔYS data for a similar composition, this information may form part of the basis for proposing ΔT_{30} and RT_{MAX-X} values considering the plant-specific surveillance data.

iii. Specific Procedures: In the event that the evaluation of factors described in Step 2(d)(ii) do not explain or rationalize the cause of the statistical tests not being satisfied, adjustment of the ETC predictions based on plant specific data should be considered. Three situations exist for which a specific procedure may be used, as follows:

1. Mean Test Failure: One procedure for adjusting ETC predictions to account for a failure of the mean test is illustrated on the left side of Figure 1. This procedure is as follows:

a. Calculate the value *ADJ* as follows:

$$ADJ = r_{mean} - r_{max}$$

b. Adjust the prediction of Equation (5) as follows:

$$\Delta T_{30(ADJ)} = MD + CRP + ADJ$$

c. Use the value $\Delta T_{30(ADJ)}$ in place of the predicted ΔT_{30} in all calculations required by the Alternate PTS Rule for the materials that do not satisfy the mean statistical test.

2. Slope Test Failure: One procedure for adjusting ETC predictions to account for a failure of the slope test is to adjust the ETC predictions (Eq. (5)) from the Alternate PTS Rule based on the greater increase of embrittlement with fluence suggested by the plant-specific data. The specific procedure used should be technically justified and documented.

3. Outlier Test Failure (Not Satisfied at Low Fluence): The right side of Figure 1 illustrates a situation where a ΔT_{30} value measured at low fluence is responsible for not satisfying the outlier test. Such a situation is not considered relevant to a PTS evaluation, and may therefore be ignored, provided that both of the following conditions are satisfied:

   a. The fluence of the datum that caused the outlier test failure (ϕt_{LOW}) is less than 10 percent of the fluence at which the PTS evaluation is being performed (ϕt_{EVAL}), and

   b. After elimination of the datum measured at (ϕt_{LOW}), the entry conditions for the surveillance tests are still met (i.e., at least three datum measured at three different fluence values remain) and all three statistical tests are satisfied with the reduced data set.

Other approaches to assessment of surveillance data where all surveillance measurements are bounded are subject to review and approval by the NRC.

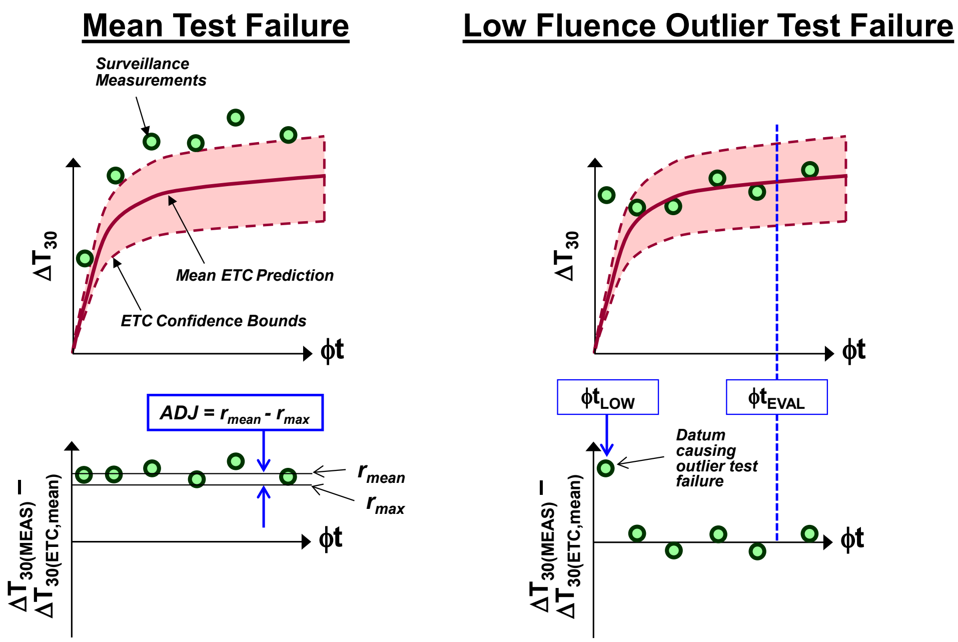

Figure 1. Specific Procedures to Address Unsatisfactory Mean Statistical Test (left) or Low Fluence Outlier Statistical Test (right)

3. Criteria Relating to ISI Data and NDE Requirements

Figure 2 illustrates the procedure described in 10 CFR 50.61a used to quantify and assess the flaws present in the RPV. Compliance with Tables 2 and 3 of 10 CFR 50.61a (repeated here as Figure 3 for convenience) demonstrates that the flaw distribution in the RPV is adequately represented by the flaw distribution assumed in the PFM calculations that established the technical basis for the RT_{MAX-X} limits in Table 1 of 10 CFR 50.61a. The steps in the flowchart of Figure 2 are as follows:

<u>Step A</u>: All recordable flaw data (see Figure 4) should be collected for the inner three-eighths of the wall thickness (3/8t) for the base material and weld metal examination volumes within the RPV beltline region using procedures, equipment and personnel, as required in ASME Code, Section XI (Ref. 7), Mandatory Appendix VIII, Supplements 4 and 6, using UT volumetric examinations. (Note: Any flaws that are detected within the ultrasonic transducer scan paths, but are located outside of the required ASME Code, Section XI, examination volume, should also be included in the evaluation required by the flaw tables in 10 CFR 50.61a; see Step D of this procedure).

<u>Step B</u>: The plant-specific flaw data from Step A should be evaluated for axial flaw surface connection. Any flaws with a through-wall extent greater than or equal to 0.075 inch, axially oriented and located at the clad-to-base metal interface, should be assessed to determine whether or not they connect to the RPV inner surface using examination techniques capable of detecting and characterizing service-induced cracking of the RPV cladding. Eddy current and visual examinations methods are acceptable to the staff for detection of cladding cracks. An appropriate quality standard shall be implemented to ensure these examinations are effective at identification of surface cracking as required by 10 CFR Part 50, Appendix B, Criterion IX “Control of Special Processes,” which requires in part, that measures shall be established to assure that special

processes, including nondestructive testing, are controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements. Appropriate quality standards for implementation of surface examinations are identified in the ASME Code Section XI "Rules for Inservice Inspection of Nuclear Power Plant Components" and/or Section V "Nondestructive Examination."

- If surface connected flaws ***do not exist*** then proceed to Step C.
- Conversely, if surface connected flaws ***do exist*** then proceed to Step G.
    - If the outcome of Step G is a pass then proceed to Step C.
    - If the outcome of Step G is a failure then proceed to Step H.

Step C: The plant-specific flaw data from Step A should be evaluated for acceptability in accordance with ASME Code, Section XI, Table IWB-3510-1 flaw acceptance standards.

- If all flaws are acceptable per the flaw acceptance standards then proceed to Step D.
- Conversely, if some flaws are not acceptable per the ASME flaw acceptance standards then proceed to Step G.
    - If the outcome of Step G is a pass then proceed to Step D.
    - If the outcome of Step G is a failure then proceed to Step H. Additionally, satisfaction of the requirements of Section XI of the ASME Code would need to be demonstrated before the vessel could be approved for a return to service.

Step D: The plant-specific flaw data should be compared to Tables 2 and 3 of 10 CFR 50.61a. A specific example of how this step may be performed, including how the plant-specific flaw data is categorized as weld flaws or plate flaws, is shown in Section 6.3 of NUREG-2163.

- If all flaws are acceptable per the flaw tables of 10 CFR 50.61a then proceed to Step I.
- Conversely, if some flaws are not acceptable per the flaw tables then proceed to Step E.

Step E: NDE uncertainties such as flaw sizing errors, a flaw detection threshold, or probability of detection (POD) may be accounted for in the evaluation. Appendix C of NUREG-2163 describes the development and application of one methodology acceptable to the NRC that accounts for uncertainties in NDE data. This method may be used for the purpose of developing more realistic vessel-specific flaw depth and density distributions for comparison to Tables 2 and 3 of 10 CFR 50.61a, as well as for use in a plant-specific PFM analysis. The methodology considers flaw sizing errors, a flaw detection threshold, POD, and a prior flaw distribution assumption. It uses a Bayesian updating methodology to combine the observed NDE data with the available flaw data and models used as part of the PTS re-evaluation effort. The licensee must submit the adjustments made to the volumetric test data to account for NDE-related uncertainties as described in (c)(2) of 10 CFR 50.61a.

- At the conclusion of this step proceed to Step F.

Step F: The revised flaw distribution results of Step E should be compared to Tables 2 and 3 of 10 CFR 50.61a.

- If all flaws are acceptable per the flaw tables of 10 CFR 50.61a then proceed to Step I.
- Conversely, if some flaws are not acceptable per the flaw tables then proceed to Step G.

Step G: A demonstration that the TWCF is less than $1x10^{-6}$ events per reactor year is necessary to satisfy paragraph (e)(4) of the Alternate PTS Rule. The staff considers the two approaches described here to be acceptable for providing assurance that the TWCF is less than $1x10^{-6}$ events per reactor year. Therefore, all flaws should be evaluated for acceptability using one of the following approaches:

1. Preclusion of brittle fracture. Satisfactory demonstration of upper shelf behavior, which precludes brittle fracture, can be based on maintaining temperature above RT_{NDT} + 60 °F using the following steps:

    i. Compute the irradiated RT_{NDT} for all flaws as follows:

    - Determine the unirradiated value of RT_{NDT}, $RT_{NDT(U)}$, for the material at each flaw location.
    - Determine the fluence at each flaw location.
    - Compute ΔT_{30} for each flaw using Eq. (5) and the fluence at each flaw location.
    - Compute the flaw-specific value of RT_{NDT} as $RT_{NDT(U)} + \Delta T_{30}$ for each flaw.

    ii. Assuming a lower bound PTS transient temperature of 75°F, upper shelf behavior is assured if $RT_{NDT} + 60 \leq 75$ °F. Therefore, the flaw-specific value of RT_{NDT} should be less than or equal to 15 °F. This evaluation is considered acceptable if the flaw-specific values of RT_{NDT} are less than or equal to 15°F for all flaws.

2. Calculate the plant-specific TWCF using a plant-specific PFM analysis. A plant-specific PFM analysis to calculate TWCF is complex, and there are many variations of inputs possible for such an analysis. Therefore, specific guidance for plant-specific PFM analysis to calculate TWCF is not included in this regulatory guide. General considerations to include in a plant-specific PFM analysis are provided in Section 6.2.2 of NUREG-2163. A discussion of the methodology that was used in performing TWCF calculations for PTS may be found in NUREG-1806, "Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61)" (Ref. 8), NUREG-1807 (Ref. 9), and NUREG/CR-6854 (Ref. 10). The steps associated with conducting a plant-specific PFM calculation are as follows:

    i. Perform a Bayesian update of the flaw distribution:
    - The procedures of Appendix C of NUREG-2163 provide an example of how to obtain revised flaw depth and flaw density parameters (similar to those shown in Table 11 of NUREG-2163).

    ii. Calculate the TWCF using a PFM computer code (e.g., ORNL/TM-2012/566, Fracture Analysis of Vessels - Oak Ridge (FAVOR) (Ref. 11)). :
    - Run the generalized procedure for generating flaw-related inputs (see Ref. 12) using the revised flaw depth and flaw density parameters.
    - Develop necessary plant-specific inputs. The guidance in NUREG-1806, NUREG-1807, and NUREG/CR-6854 can be used to provide examples.
    - Run a plant-specific PFM analysis.
    - Calculate the TWCF.

    iii. Compare the plant-specific TWCF to the TWCF limit specified in 10 CFR 50.61a:
    - The evaluation associated with Step I is acceptable if the calculated TWCF is less than or equal to the $1x10^{-6}$ events per reactor year limit specified in 10 CFR 50.61a.

- If the outcome of this step is a pass then proceed to Step I.
- If the outcome of this step is a failure then proceed to Step H.

__Step H__: The licensee should perform a plant-specific assessment for PTS and submit the assessment to the Director of the Office of Nuclear Reactor Regulation (NRR) for review and approval as required by 10 CFR 50.61a(d)(4).

__Step I__: The screening criteria contained in Table 1 of 10 CFR 50.61a may be applied to the plant in question. As required by 10 CFR 50.61a(c) , the plant-specific assessment, including explicit details and results, must be submitted to the Director of NRR for review and approval in the form of a license amendment at least 3 years before RT_{MAX-X} is projected to exceed the Alternate PTS Rule screening criteria.

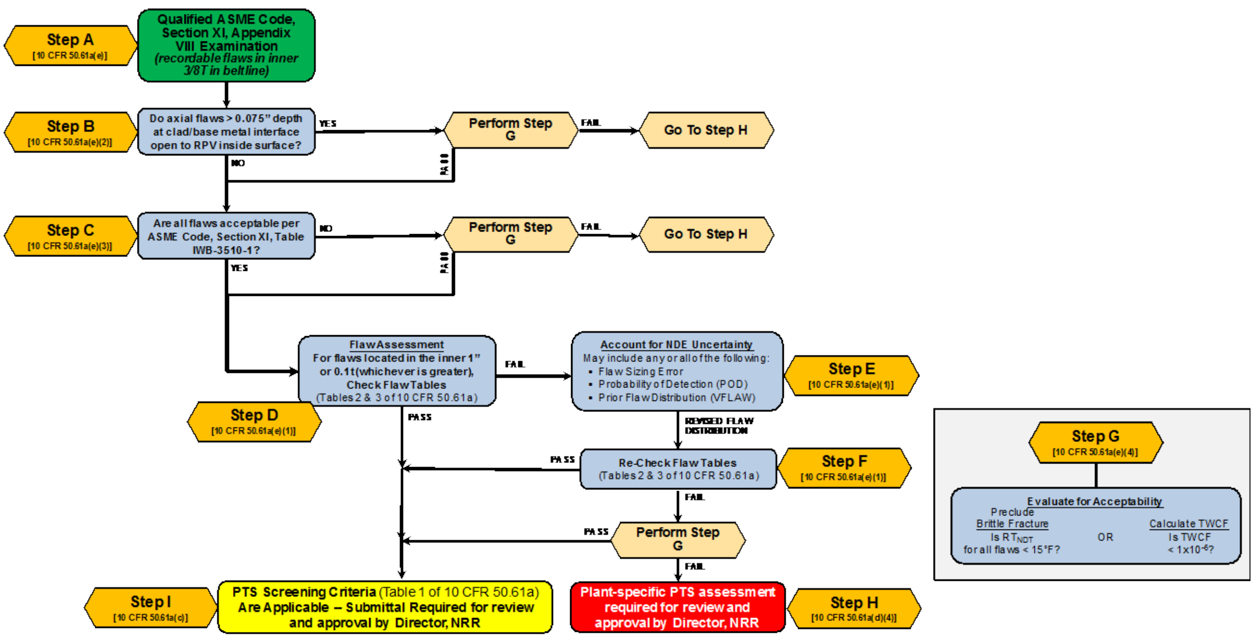

Figure 2. Flow Diagram with Guidance for Meeting the Requirements of the Alternate PTS Rule.

TABLE 2—ALLOWABLE NUMBER OF FLAWS IN WELDS

Through–wall extent, TWE [in.]		Maximum number of flaws per 1000-inches of weld length in the inspection volume that are greater than or equal to TWE_{MIN} and less than TWE_{MAX}
TWE_{MIN}	TWE_{MAX}	
0	0.075	No Limit
0.075	0.475	166.70
0.125	0.475	90.80
0.175	0.475	22.82
0.225	0.475	8.66
0.275	0.475	4.01
0.325	0.475	3.01
0.375	0.475	1.49
0.425	0.475	1.00
0.475	Infinite	0.00

TABLE 3—ALLOWABLE NUMBER OF FLAWS IN PLATES AND FORGINGS

Through-wall extent, TWE [in.]		Maximum number of flaws per 1000 square-inches of inside surface area in the inspection volume that are greater than or equal to TWE_{MIN} and less than TWE_{MAX}. This flaw density does not include underclad cracks in forgings.
TWE_{MIN}	TWE_{MAX}	
0	0.075	No Limit
0.075	0.375	8.05
0.125	0.375	3.15
0.175	0.375	0.85
0.225	0.375	0.29
0.275	0.375	0.08
0.325	0.375	0.01
0.375	Infinite	0.00

Figure 2. Flaw tables (that is, Tables 2 and 3) from 10 CFR 50.61a.

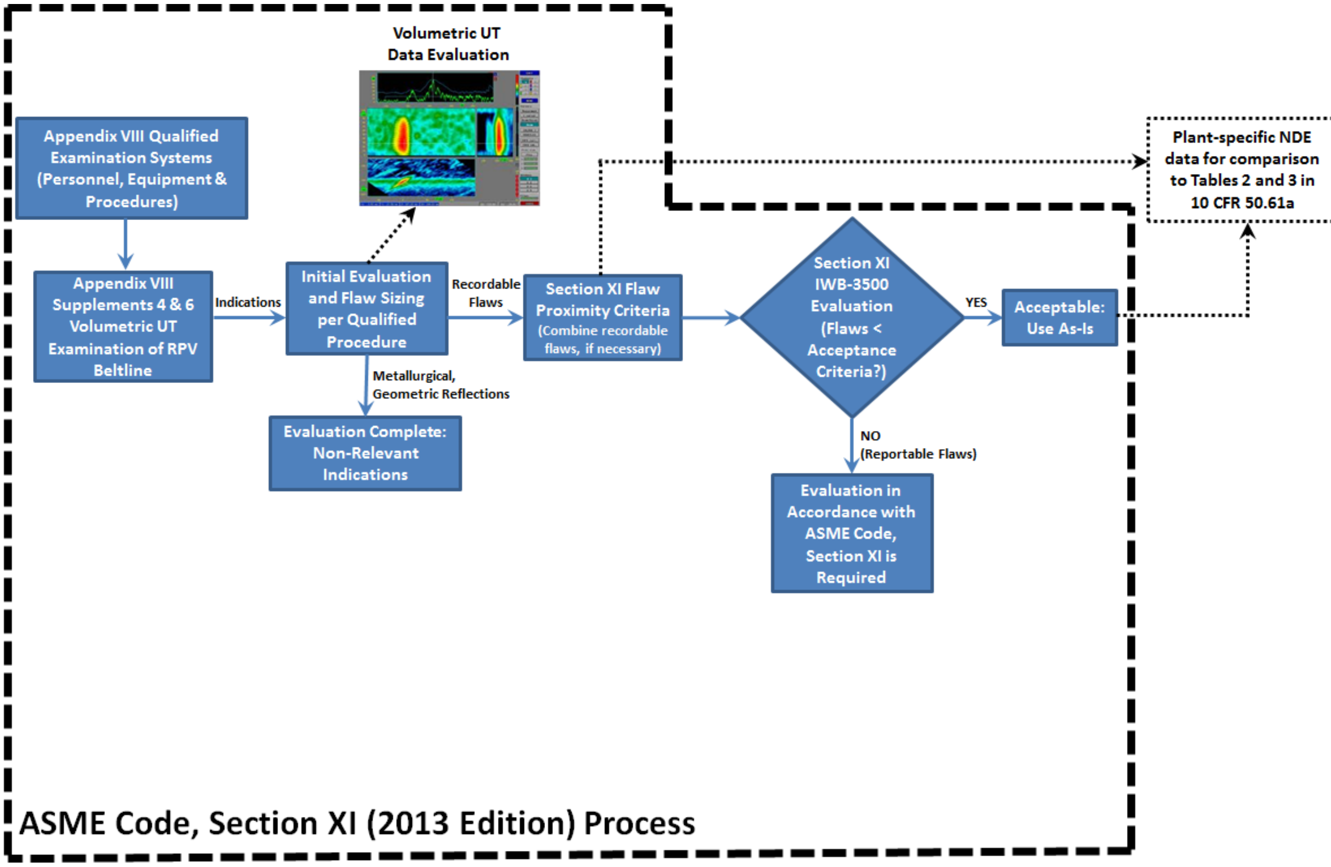

Figure 4. ASME Code, Section XI Examination and Flaw Evaluation Process and Identification of Flaws for Comparison to Alternate PTS Rule

4. Criteria Relating to Alternate Limits on Embrittlement

This regulatory position describes one alternate procedure by which licensees could assess their plant-specific TWCF for cases where embrittlement criteria are not met, as allowed by Paragraph (c)(3) of 10 CFR 50.61a (i.e., the RT_{MAX-X} limits of Table 1 of 10 CFR 50.61a are not satisfied). Performing calculations following this position would fulfill the requirements in Paragraphs (d)(3) through (d)(6) of 10 CFR 50.61a provided that all other requirements of 10 CFR 50.61a concerning flaw evaluations, flaw assessment, statistical evaluation of surveillance data, and plant corrective actions are also satisfied.

One method to calculate a plant-specific TWCF value is using the methods and formulae provided in Section 3.5.1, Step 4, of NUREG-1874 (Ref. 13). Satisfactory demonstration for this position includes the following steps:

- Step 1. Establish the plant characterization parameters (e.g., copper, fluence).

- Step 2. Estimate values of RT_{MAX-X} using the values of the characterization parameters from Step 1. The formulae given in Section 3.5.1 of NUREG-1874 can be used as to estimate RT_{MAX-X}.

- Step 3. Estimate the 95th percentile TWCF value, $TWCF_{95-XX}$, for each of the axial weld flaw, plate flaw, circumferential weld flaw, and forging flaw populations (as applicable) using the

RT_{MAX-X} values from Step 2. The formulae given in Section 3.5.1 of NUREG-1874 can be used to estimate $TWCF_{95-XX}$.

- Step 4. Estimate the total 95th percentile TWCF, $TWCF_{95-TOTAL}$, for the vessel. The formulae given in Section 3.5.1 of NUREG-1874 can be used to estimate $TWCF_{95-TOTAL}$.

The results of this approach are acceptable if the calculated plant-specific value of $TWCF_{95-TOTAL}$ is less than or equal to $1x10^{-6}$ events per reactor year.

D. IMPLEMENTATION

The purpose of this section is to provide information on how licensees[1] may use this guide and information regarding the NRC's plans for using this regulatory guide. In addition, it describes how the NRC staff complies with the Backfit Rule (10 CFR 50.109).

Use by Licensees

Licensees may voluntarily[2] use the guidance in this document to demonstrate compliance with the underlying NRC regulations. Methods or solutions that differ from those described in this regulatory guide may be deemed acceptable if they provide sufficient basis and information for the NRC staff to verify that the proposed alternative demonstrates compliance with the appropriate NRC regulations.

Licensees may use the information in this regulatory guide for actions that do not require NRC review and approval such as changes to a facility design under 10 CFR 50.59, "Changes, Tests, and Experiments," that do not require prior NRC review and approval. Licensees may use the information in this regulatory guide or applicable parts to resolve regulatory or inspection issues.

Use by NRC Staff

The NRC staff does not intend or approve any imposition or backfitting of the guidance in this regulatory guide. The NRC staff does not expect any existing licensee to use or commit to using the guidance in this regulatory guide, unless the licensee makes a change to its licensing basis. The NRC staff does not expect or plan to request licensees to voluntarily adopt this regulatory guide to resolve a generic regulatory issue. The NRC staff does not expect or plan to initiate NRC regulatory action that would require the use of this regulatory guide without further backfitting consideration. Examples of such unplanned NRC regulatory actions include issuance of an order requiring the use of the regulatory guide, requests for information under 10 CFR 50.54(f) as to whether a licensee intends to commit to use of this regulatory guide, generic communications, or a rule requiring the use of this regulatory guide.

During regulatory discussions on plant-specific operational issues, the staff may discuss with licensees various actions consistent with staff positions in this regulatory guide, as one acceptable means of meeting the underlying NRC regulatory requirement. Such discussions would not ordinarily be considered backfitting. However, unless this regulatory guide is part of the licensing basis for a facility, the staff may not represent to the licensee that the licensee's failure to comply with the positions in this regulatory guide constitutes a violation.

If an existing licensee voluntarily seeks a license amendment or change and (1) the NRC staff's consideration of the request involves a regulatory issue directly relevant to this new or revised regulatory guide and (2) the specific subject matter of this regulatory guide is an essential consideration in the staff's determination of the acceptability of the licensee's request, then the staff may request that the licensee either follow the guidance in this regulatory guide or provide an equivalent alternative process that demonstrates compliance with the underlying NRC regulatory requirements. This is not considered backfitting as defined in 10 CFR 50.109(a)(1).

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1 In this section, "licensees" refers to licensees of nuclear power plants under 10 CFR Part 50 whose construction permit was issued before February 3, 2010, and whose RPV was designed and fabricated to the requirements of the ASME Boiler and Pressure Vessel Code, 1998 edition or earlier.

2 In this section, "voluntary" and "voluntarily" means that the licensee is seeking the action of its own accord, without the force of a legally binding requirement or an NRC representation of further licensing or enforcement action.

If a licensee believes that the NRC is either using this regulatory guide or requesting or requiring the licensee to implement the methods or processes in this regulatory guide in a manner inconsistent with the discussion in this Implementation section, then the licensee may file a backfit appeal with the NRC in accordance with the guidance in NRC Management Directive 8.4, "Management of Facility-Specific Backfitting and Information Collection" (Ref. 14), and in NUREG-1409, "Backfitting Guidelines" (Ref. 15).

REFERENCES[3]

1. *U.S. Code of Federal Regulations*, Title 10 "Energy," Part 50, "Domestic Licensing of Production and Utilization Facilities," U.S. Nuclear Regulatory Commission, Washington, DC.

2. American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Facility Components," New York, NY, 1998 Edition or earlier.[4]

3. Nuclear Regulatory Commission (NRC), Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," Washington, DC, March 2001.

4. NRC, "Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," *Federal Register,* Vol. 75 p.13 (75 FR 13), Washington, DC, January 4, 2010.

5. NRC, NUREG-2163, "Technical Basis for Regulatory Guidance on the Alternate Pressurized Thermal Shock Rule," Washington, DC, December 2015 (ML15058A677).

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9. NRC, NUREG-1807, "Probabilistic Fracture Mechanics – Models, Parameters, Uncertainty Treatment Used in FAVOR Version 04.1," Washington, DC, June 2007.

10. NRC, NUREG/CR-6854, ORNL/TM-2004/244, "Fracture Analysis of Vessels – Oak Ridge FAVOR, v04.1, Computer Code: Theory and Implementation of Algorithms, Methods, and Correlations," Washington, DC, August 2007.

11. Oak Ridge National Lab (ORNL)/TM-2012/566, "Fracture Analysis of Vessels – Oak Ridge FAVOR, v12.1, Computer Code: User's Guide," Oak Ridge, TN, November 2012 (ML13008A016).

12. NRC, NUREG/CR-6817, PNNL-14268, "A Generalized Procedure for Generating Flaw-Related Inputs for the FAVOR Code," Washington, DC, March 2004.

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[3] Publicly available NRC published documents are available electronically through the NRC Library on the NRC's public Web site at: http://www.nrc.gov/reading-rm/doc-collections/. The documents can also be viewed on-line or printed for a fee in the NRC's Public Document Room (PDR) at 11555 Rockville Pike, Rockville, MD; the mailing address is USNRC PDR, Washington, DC 20555; telephone 301-415-4737 or (800) 397-4209; fax (301) 415-3548; and e-mail pdr.resource@nrc.gov.

[4] Copies may be purchased from the American Society of Mechanical Engineers, Three Park Avenue, New York, NY 10016-5990; phone (212) 591-8500; fax (212) 591-8501; www.asme.org.

13. NRC, NUREG-1874, "Recommended Screening Limits for Pressurized Thermal Shock (PTS)," Washington, DC, March 2010.

14. NRC, Management Directive 8.4, "Management of Facility-Specific Backfitting and Information Collection," Washington, DC.

15. NRC, NUREG-1409, "Backfitting Guidelines," Washington, DC.