[7590-01-P]

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

RIN 3150-AI01

[NRC-2007-0008]

Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule.

SUMMARY: The Nuclear Regulatory Commission (NRC) is amending its regulations to provide alternate fracture toughness requirements for protection against pressurized thermal shock (PTS) events for pressurized water reactor (PWR) pressure vessels. This final rule provides alternate PTS requirements based on updated analysis methods. This action is desirable because the existing requirements are based on unnecessarily conservative probabilistic fracture mechanics analyses. This action reduces regulatory burden for those PWR licensees who expect to exceed the existing requirements before the expiration of their licenses, while maintaining adequate safety, and may choose to comply with the final rule as an alternative to complying with the existing requirements.

EFFECTIVE DATE: **[INSERT DATE 30 DAYS AFTER THE DATE OF PUBLICATION IN THE FEDERAL REGISTER]**.

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SUPPLEMENTARY INFORMATION:

I. Background.

II. Discussion.

I. Background

PTS events are system transients in a PWR in which there is a rapid operating temperature cooldown that results in cold vessel temperatures with or without repressurization of the vessel. The rapid cooling of the inside surface of the reactor vessel causes thermal stresses. The thermal stresses can combine with stresses caused by high pressure. The aggregate effect of these stresses is an increase in the potential for fracture if a pre-existing flaw is present in a material susceptible to brittle failure. The ferritic, low alloy steel of the reactor vessel beltline adjacent to the core, where neutron radiation gradually embrittles the material over the lifetime of the plant, can be susceptible to brittle fracture.

The current PTS rule, described in § 50.61, “Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events,” adopted on July 23, 1985 (50 FR 29937), establishes screening criteria below which the potential for a reactor vessel to fail due to a PTS event is deemed to be acceptably low. These screening criteria effectively

define a limiting level of embrittlement beyond which operation cannot continue without further plant-specific evaluation.

A licensee may not continue to use a reactor vessel with materials predicted to exceed the screening criteria in § 50.61 without implementing compensatory actions or additional plant-specific analyses unless the licensee receives an exemption from the requirements of the rule. Acceptable compensatory actions are neutron flux reduction, plant modifications to reduce the PTS event probability or severity, and reactor vessel annealing, which are addressed in §§ 50.61(b)(3), (b)(4), and (b)(7); and 50.66, "Requirements for Thermal Annealing of the Reactor Pressure Vessel."

Currently, no operating PWR vessel is projected to exceed the § 50.61 screening criteria before the expiration of its 40 year operating license. However, several PWR vessels are approaching the screening criteria, while others are likely to exceed the screening criteria during the extended period of operation of their first license renewal.

The NRC's Office of Nuclear Regulatory Research (RES) developed a technical basis that supports updating the PTS regulations. This technical basis concluded that the risk of through-wall cracking due to a PTS event is much lower than previously estimated. This finding indicated that the screening criteria in § 50.61 are unnecessarily conservative and may impose an unnecessary burden on some licensees. Therefore, the NRC developed a proposed new rule, § 50.61a, "Alternate Fracture Requirements for Protection against Pressurized Thermal Shock Events," providing alternate screening criteria and corresponding embrittlement correlations based on the updated technical basis. The NRC decided that providing a new section containing the updated screening criteria and updated embrittlement correlations would be appropriate. The NRC could have revised § 50.61 to include the new requirements, which could be implemented as an alternative to the current requirements. However, providing two

sets of requirements within the same regulatory section was considered confusing and/or ambiguous as to which requirements apply to which licensees.

The NRC published the proposed rule for public comment in the *Federal Register* on October 3, 2007 (72 FR 56275). Following the closure of the comment period on the proposed rule and during the development of the PTS final rule, the NRC determined that several changes to the October 3, 2007 proposed rule language were desirable to adequately address issues raised in stakeholder's comments. Because these modifications may not have represented a logical outgrowth from the October 2007 proposed rule's provisions, the NRC requested stakeholder feedback on the modified provisions in a supplemental proposed rule published in August 11, 2008 (73 FR 46557). In the supplemental proposed rule, the NRC proposed modifications to the provisions related to the applicability of the rule and the evaluation of reactor vessel surveillance data. In addition, the NRC requested comments on the adjustments of volumetric examination data to demonstrate compliance with the rule. After consideration of the October 2007 proposed rule, the August 2008 supplemental proposed rule and the stakeholder comments received on both, the NRC has decided to adopt the PTS final rule as described further in this document.

II. Discussion

The NRC completed a research program that concluded that the risk of through-wall cracking due to a PTS event is much lower than previously estimated. This finding indicates that the screening criteria in § 50.61 are unnecessarily conservative and may impose an unnecessary burden on some licensees. Therefore, the NRC developed a final rule, § 50.61a, that can be implemented by PWR licensees.

The § 50.61a alternate screening criteria and corresponding embrittlement correlations are based on a technical basis as documented in the following reports: (1) NUREG-1806,

"Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limits in the PTS Rule (10 CFR 50.61): Summary Report," (ADAMS Accession No. ML061580318); (2) NUREG-1874, "Recommended Screening Limits for Pressurized Thermal Shock (PTS)" (ADAMS Accession No. ML070860156); (3) Memorandum from Elliot to Mitchell, dated April 3, 2007, "Development of Flaw Size Distribution Tables for Draft Proposed Title 10 of the *Code of Federal Regulations* (10 CFR) 50.61a," (ADAMS Accession No. ML070950392); (4) "Statistical Procedures for Assessing Surveillance Data for 10 CFR Part 50.61a," (ADAMS Accession No. ML081290654); and (5) "A Physically Based Correlation of Irradiation Induced Transition Temperature Shifts for RPV Steel," (ADAMS Accession No. ML081000630).

Applicability of the Final Rule.

The final rule is based on, in part, analyses of information from three currently operating PWRs. Because the severity of the risk-significant transient classes (e.g., primary side pipe breaks, stuck open valves on the primary side that may later re-close) is controlled by factors that are common to PWRs in general, the NRC concluded that the results and screening criteria developed from the analysis of these three plants can be applied with confidence to the entire fleet of operating PWRs. This conclusion is based on an understanding of characteristics of the dominant transients that drive their risk significance and on an evaluation of a larger population of high embrittlement PWRs. This evaluation revealed no design, operational, training, or procedural factors that could credibly increase either the severity of these transients or the frequency of their occurrence in the general PWR population above the severity and frequency characteristic of the three plants that were modeled in detail. The NRC also concluded that insignificant PTS events are not expected to become dominant.

The final rule is applicable to licensees whose construction permits were issued before **[INSERT EFFECTIVE DATE OF FINAL RULE]** and whose reactor vessels were designed and

fabricated to the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), 1998 Edition or earlier. This would include applicants for plants such as Watts Bar Unit 2 who have not yet received an operating license. However, it cannot be demonstrated, *a priori*, that reactor vessels that were not designed and fabricated to the specified ASME Code editions will have material properties, operating characteristics, PTS event sequences and thermal-hydraulic responses consistent with those evaluated as part of the technical basis for this rule. Therefore, the NRC determined that it would not be prudent at this time to extend the use of the rule to future PWR plants and plant designs such as the Advanced Passive (AP) 1000, Evolutionary Power Reactor (EPR) and U.S. Advanced Pressurized Water Reactor (US-APWR). These designs have different reactor vessels than those in the currently operating plants, and the fabrication of the vessels based on these designs may differ from the vessels evaluated in the analyses that form the bases for the final rule. Licensees of reactors who commence commercial power operation after the effective date of this rule or licensees with reactor vessels that were not designed and fabricated to the 1998 Edition or earlier of the ASME Code may, under the provisions of § 50.12, seek an exemption from § 50.61a(b) to apply this rule if a plant-specific basis analyzing their plant operating characteristics, materials of fabrication, and welding methods is provided.

Updated Embrittlement Correlation.

The technical basis for § 50.61a uses many different models and parameters to estimate the yearly probability that a PWR will develop a through-wall crack as a consequence of PTS loading. One of these models is a revised embrittlement correlation that uses information on the chemical composition and neutron exposure of low alloy steels in the reactor vessel's beltline region to estimate the resistance to fracture of these materials. Although the general trends of the embrittlement models in §§ 50.61 and 50.61a are similar, the form of the revised

embrittlement correlation in § 50.61a differs substantially from the correlation in § 50.61. The correlation in the § 50.61a final rule has been updated to more accurately represent the substantial amount of reactor vessel surveillance data that has accumulated since the embrittlement correlation was last revised during the 1980s.

Inservice Inspection Volumetric Examination and Flaw Assessments.

The § 50.61a final rule differs from § 50.61 in that it contains a requirement for licensees who choose to follow its requirements to analyze the results from the ASME Code, Section XI inservice inspection volumetric examinations. The examinations and analyses will determine if the flaw density and size distribution in the licensee's reactor vessel beltline are bounded by the flaw density and size distribution used in the technical basis. The technical basis was developed using a flaw density, spatial distribution, and size distribution determined from experimental data, as well as from physical models and expert elicitation. The experimental data were obtained from samples removed from reactor vessel materials from cancelled plants (i.e., Shoreham and the Pressure Vessel Research Users Facility (PVRUF) vessel). The NRC considers that the analysis of the ASME Code inservice inspection volumetric examination is needed to confirm that the flaw density and size distributions in the reactor vessel, to which the final rule may be applied, are consistent with those in the technical basis.

Paragraph (g)(6)(ii)(C) of 10 CFR 50.55a requires licensees to implement the ASME Code, Section XI, Appendix VIII, Supplements 4 and 6. Supplement 4 contains qualification requirements for the reactor vessel inservice inspection volume from the clad-to-base metal interface to the inner 1.0 inch or 10 percent of the vessel thickness, whichever is larger. Supplement 6 contains qualification requirements for reactor vessel weld volumes other than those near the clad-to-base metal interface. Analysis of the performance by qualified inspectors indicates that there is an 80 percent or greater probability of detecting a flaw that contributes to

crack initiation from PTS events when they are inspected using the ASME Code, Section XI, Appendix VIII, Supplement 4 requirements[1].

The true flaw density for flaws with a through-wall extent of between 0.1 and 0.3 inch can be inferred from the ASME Code examination results and the probability of detection. The technical basis for the final rule concludes that flaws as small as 0.1 inch in through-wall extent contribute to the through-wall crack frequency (TWCF), and nearly all of the contributions come from flaws buried less than 1 inch below the inner diameter surface of the reactor vessel. For weld flaws that exceed the sizes prescribed in the final rule, the risk analysis indicates that a single flaw can be expected to contribute a significant fraction of the 1×10^{-6} per reactor year limit on TWCF. Therefore, if a flaw that exceeds the sizes prescribed in the final rule is found in a reactor vessel, it is important to assess it individually.

The technical basis for the final rule also indicates that flaws buried deeper than 1 inch from the clad-to-base interface are not as susceptible to brittle fracture as similar size flaws located closer to the inner surface. Therefore, the final rule does not require the comparison of the density of these flaws, but still requires large flaws, if discovered, to be evaluated for contributions to TWCF if they are within the inner three-eights of the vessel thickness. The limitation for flaw acceptance, specified in ASME Code, Section XI, Table IWB-3510-1, approximately corresponds to the threshold for flaw sizes that can make a significant contribution to TWCF if present in reactor vessel material at this depth. Therefore, the final rule requires that flaws exceeding the size limits in ASME Code, Section XI, Table IWB-3510-1 be evaluated for contribution to TWCF in addition to the other evaluations for such flaws that are prescribed in the ASME Code.

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[1] Becker, L., "Reactor Pressure Vessel Inspection Reliability," Proceeding of the Joint EC-IAEA Technical Meeting on the Improvement in In-Service Inspection Effectiveness, Petten, the Netherlands, November 2002.

The numerical values in Tables 2 and 3 of the final rule represent the number of flaws in each size range that were derived from the technical basis. Verifying that a plant that intends to implement this rule has weld, plate and/or forging flaw distributions which are consistent with those assumed in the technical basis is necessary to ensure the applicability of the rule to that plant. If one or more larger flaws are found in a reactor vessel, they must be evaluated to ensure that they are not causing the TWCF to exceed the regulatory limit.

The final rule also clarifies that, to be consistent with ASME Code, Section XI, Appendix VIII, the smallest flaws that must be sized are 0.075 inches in through-wall extent. For each flaw detected that has a through-wall extent equal to or greater than 0.075 inches, the licensee shall document the dimensions of the flaw, its orientation and its location within the reactor vessel, and its depth from the clad-to-base metal interface. Those planar flaws for which the major axis of the flaw is identified by an ultrasonic transducer oriented in the circumferential direction must be documented as “axial.” All other planar flaws may be categorized as “circumferential.” The NRC may also use this information to evaluate whether plant-specific information gathered suggests that the NRC staff should generically re-examine the technical basis for the rule.

Surface cracks that penetrate through the stainless steel clad and more than 0.070 inch into the welds or the adjacent base metal were not included in the technical basis because these types of flaws have not been observed in the beltline of any operating PWR vessel. However, flaws of this type were observed in the Quad Cities Unit 2 reactor vessel head in 1990 (NUREG-1796, “Safety Evaluation Report Related to the License Renewal of the Dresden Nuclear Power Station, Units 2 and 3 and Quad Cities Nuclear Power Station, Units 1 and 2,” dated October 31, 2004). The observed cracks had a maximum depth into the base metal of approximately 0.24 inch and penetrated through the stainless steel clad. Quad Cities Units 2 and 3 are boiling water reactors which are not susceptible to PTS events and hence are not

subject to the requirements of 10 CFR 50.61. The cracking at Quad Cities Unit 2 was attributed to intergranular stress corrosion cracking of the stainless steel cladding, which has not been observed in PWR vessels, and hot cracking of the low alloy steel base metal. If these cracks were in the beltline region of a PWR, they would be a significant contributor to TWCF because of their size and location. The final rule requires licensees to determine if cracks of this type exist in the beltline weld region at each ASME Code, Section XI, ultrasonic examination.

Nondestructive Examination (NDE)-Related Uncertainties.

The flaw sizes in Tables 2 and 3 represent actual flaw dimensions while the results from the ASME Code examinations are estimated dimensions. The available information indicates that, for most flaw sizes in Tables 2 and 3, qualified inspectors will oversize flaws. Comparing oversized flaws to the size and density distributions in Tables 2 and 3 is conservative and acceptable, but not necessary.

As a result of stakeholder feedback received on the NRC solicitation for comments published in the August 2008 supplemental proposed rule, the final rule will permit licenses to adjust the flaw sizes estimated by inspectors qualified under the ASME Code, Section XI, Appendix VIII, Supplement 4 and Supplement 6.

The NRC determined that, in addition to the NDE sizing uncertainties, licensees should be allowed to consider other NDE uncertainties, such as probability of detection and flaw density and location, because these uncertainties may affect the ability of a licensee to demonstrate compliance with the rule. As a result, the language in § 50.61a(e) will allow licensees to account for the effects of NDE-related uncertainties in meeting the flaw size and density requirements of Tables 2 and 3. The methodology to account for the effects of NDE-related uncertainties must be based on statistical data collected from ASME Code inspector qualification tests or any other tests that measure the difference between the actual

flaw size and the size determined from the ultrasonic examination. Verification that a licensee's flaw size and density distribution are upper-bounded by the distribution of Tables 2 and 3 is required to confirm that the risk associated with PTS is acceptable. Collecting, evaluating, and using data from ASME Code inspector qualification tests will require extensive engineering judgment. Therefore, the methodology used to adjust flaw sizes to account for the effects of NDE-related uncertainties must be reviewed and approved by the Director of the Office of Nuclear Reactor Regulation (NRR).

Surveillance Data.

Paragraph (f) of the final rule defines the process for calculating the values for the reference temperature properties (i.e., defined as RT_{MAX-X}) for a particular reactor vessel. These values must be based on the vessel material's copper, manganese, phosphorus, and nickel weight percentages, reactor cold leg temperature, and fast neutron flux and fluence values, as well as the unirradiated nil-ductility transition reference temperature (i.e., RT_{NDT}).

The rule includes a procedure by which the RT_{MAX-X} values, which are predicted for plant-specific materials using a generic temperature shift (i.e., ΔT_{30}) embrittlement trend curve, are compared with heat-specific surveillance data that are collected as part of 10 CFR Part 50, Appendix H surveillance programs. The purpose of this comparison is to assess how well the surveillance data are represented by the generic embrittlement trend curve. If the surveillance data are close (closeness is assessed statistically) to the generic embrittlement trend curve, then the predictions of this embrittlement trend curve are used. This is expected to be the case most often. However, if the heat-specific surveillance data deviate significantly, and non-conservatively, from the predictions of the generic embrittlement trend curve, this indicates that alternative methods (i.e., other than, or in addition to, the generic embrittlement trend curve)

may be needed to reliably predict the temperature shift trend, and to estimate RT_{MAX-X}, for the conditions being assessed.

The NRC is modifying the final rule to include three statistical tests to determine the significance of the differences between heat-specific surveillance data and the embrittlement trend curve. The NRC determined that a single test is not sufficient to ensure that the temperature shift predicted by the embrittlement trend curve represents well the heat-specific surveillance data. Specifically, this single statistical test cannot determine if the temperature shift from the surveillance data show a more rapid increase after significant radiation exposure than the progression predicted by the generic embrittlement trend curve. This potential deficiency could be particularly important during a plant's period of extended operation. The deviations from the generic embrittlement trend curve are best assessed by licensees on a case-by-case basis, which would be submitted for the review of the Director of NRR.

The results of the first statistical test will determine if, on average, the temperature shifts from the surveillance data are significantly higher than the temperature shifts from the generic embrittlement trend curve. The results of the second and third tests will determine if the temperature shift from the surveillance data show a more rapid increase after significant radiation exposure than the progression predicted by the generic embrittlement trend curve.

III. Responses to Comments on the Proposed Rule and Supplemental Proposed Rule

The NRC received 5 comment letters for a total of 54 comments on the proposed rule published on October 3, 2007, and 3 comment letters for a total of 5 comments on the supplemental proposed rule published on August 11, 2008. All the comments on the proposed rule and supplemental proposed rule were submitted by industry stakeholders. A detailed discussion of the public comments and the NRC's responses are contained in a separate document (see Section V, "Availability of Documents," of this document). This section only

discusses the more significant comments received on the proposed rule and supplemental proposed rule provisions and the substantive changes made to develop the final rule requirements. The NRC also requested stakeholder feedback on one question in the supplemental proposed rule. This section discusses the comments received from the NRC inquiry and the changes made to the final rule language as a result of these comments. Comments are discussed by subject.

Comments on the Applicability of the Proposed Rule:

Comment: The commenters stated that the rule, as written, is only applicable to the existing fleet of PWRs. The characteristics of advanced PWR designs were not considered in the analysis. The commenters suggested adding a statement that this rule is applicable to the current PWR fleet and not the new plant designs.

Response: The NRC agrees with the comment that this rule is only applicable to the existing fleet of PWRs. The NRC cannot be assured that plants whose construction permit was issued after **[INSERT EFFECTIVE DATE OF FINAL RULE]**, and whose reactor vessel was designed and fabricated to ASME Code Editions later than the 1998 Edition will have material properties, operating characteristics, PTS event sequences and thermal-hydraulic responses consistent with the reactors that were evaluated as part of the technical basis for § 50.61a. Other factors, including materials of fabrication and welding methods, would also be consistent with the underlying technical basis of 10 CFR 50.61a. As a result of this comment, the NRC modified § 50.61a(b) and the statement of considerations of the rule to reflect this position to allow the use of the rule only to plants whose construction permit was issued before **[INSERT EFFECTIVE DATE OF FINAL RULE]** and whose reactor vessel was designed and fabricated to the 1998 Edition or earlier of the ASME Code.

Comments on Surveillance Data:

Comment: The commenters stated that there is little added value in the requirement to assess the surveillance data as a part of this rule because variability in data has already been accounted for in the derivation of the embrittlement correlation.

The commenters also stated that there is no viable methodology for adjusting the projected ΔT_{30} for the vessel based on the surveillance data. Any effort to make this adjustment is likely to introduce additional error into the prediction. Note that the embrittlement correlation described in the basis for the revised PTS rule (i.e., NUREG-1874) was derived using all of the then available industry-wide surveillance data.

In the event that the surveillance data does not match the ΔT_{30} value predicted by the embrittlement correlation, the best estimate value for the pressure vessel material is derived using the embrittlement correlation. The likely source of the discrepancy is an error in the characterization of the surveillance material or of the irradiation environment. Therefore, unless the discrepancy can be resolved, obtaining the ΔT_{30} prediction based on the best estimate chemical composition for the heat of the material is more reliable than a prediction based on a single set of surveillance measurements.

The commenters suggested removing the requirement to assess surveillance data, including Table 5, of this rule.

Response: The NRC does not agree with the proposed change. The NRC believes that there is added value in the requirement to assess reactor vessel surveillance data. Although variability has been accounted for in the derivation of the embrittlement correlation, it is the NRC's view that the surveillance data assessment required in § 50.61a(f)(6) is needed to determine if the embrittlement for a specific heat of material in a reactor vessel is consistent with the embrittlement predicted by the embrittlement correlation.

The commenters also assert that there is no viable methodology for adjusting the projected ΔT_{30} for the vessel based on the surveillance data, and that any adjustment is likely to introduce additional error into the prediction. The NRC believes that although there is no single methodology for adjusting the projected ΔT_{30} for the vessel based on the surveillance data, it is possible, on a case-specific basis, to justify adjustments to the generic ΔT_{30} prediction. For this reason the rule does not specify a method for adjusting the ΔT_{30} value based on surveillance data, but rather requires the licensee to propose a case-specific ΔT_{30} adjustment procedure for review and approval of the Director of NRR. Although the commenters assert that it is possible that error could be introduced, it is the NRC view that appropriate plant-specific adjustments based upon available surveillance data may be necessary to project reactor pressure vessel embrittlement for the purpose of this rule.

As the result of these public comments, the NRC has continued to work on statistical procedures to identify deviations from generic embrittlement trends, such as those described in § 50.61a(f)(6) of the proposed rule. Based on this work, the NRC enhanced the procedure described in § 50.61a(f)(6) to, among other things, detect trends from plant- and heat-specific surveillance data that may emerge at high fluences that are not reflected by Equations 5, 6, and 7. The empirical basis for the NRC's concern regarding the potential for un-modeled high fluence effects is described in documents located at ADAMS Accession Nos. ML081120253, ML081120289, ML081120365, ML081120380, and ML081120600. The technical basis for the enhanced surveillance data assessment procedure is described in the document located at ADAMS Accession No. ML081290654.

Comment: The second surveillance data check described in the supplemental proposed rule should be eliminated from the rule because the slope change evaluation appears to be of limited value.

The second required surveillance data check is to address a slope change. The intent of this section appears to identify potential increases in the embrittlement rate at high fluence. The industry intends to move forward with an initiative to populate the power reactor vessel surveillance program database with higher neutron fluence surveillance data (i.e., extending to fluence values equivalent to 60-80 effective full power year (EFPY)) that will adequately cover materials variables for the entire PWR fleet. This database should provide a more effective means of evaluating the potential for enhanced embrittlement rates at high fluence values rather than using an individual surveillance data set to modify the trend with fluence. Data from this initiative will be available in the next few years to assess the likelihood of enhanced embrittlement rates for the PWR fleet.

Response: The NRC does not agree with the commenters' statement that the slope test (i.e., § 50.61a(f)(6)(iii)) has limited value and that it should be eliminated from the rule. The NRC believes that the slope test provides a method for determining whether high neutron fluence surveillance data is consistent with the ΔT_{30} model in the rule. Because there are currently only a few surveillance data points from commercial power reactors at high neutron fluences and the slope test will provide meaningful information, the NRC determines that the slope test should not be eliminated from the rule.

The NRC agrees with the industry initiative to obtain additional power reactor data at higher fluences. The NRC will review this data and the information available to evaluate the effects of high neutron fluence exposure when it becomes available. At that point, the NRC will determine if modifications to the embrittlement model and/or the surveillance data checks in § 50.61a should be made.

No changes were made to the rule language as a result of this comment.

Comments Related to the NRC Inquiry Related to the Adjustment of Volumetric Examination Data:

Comment: § 50.61a(e) should be modified to allow licensees to account for the effects of flaw sizing uncertainties and other uncertainties in meeting the requirements of Tables 2 and 3. The rule language should allow the use of applicable data from ASME qualification tests, vendor-specific performance demonstration tests, and other current and future data that may be applicable for assessing these uncertainties. The rule language should permit flaw sizes to be adjusted to account for the sizing uncertainties and other uncertainties before comparing the estimated size and density distribution to the acceptable size and density distributions in Tables 2 and 3.

The industry will provide guidance to enable licensees to account for the effects of sizing uncertainties and other uncertainties in meeting the requirements of Tables 2 and 3 of the rule. Guidance to ensure that the risk associated with PTS is acceptable will be provided to the Director of NRR for review and approval when completed.

Response: The NRC agrees that, in addition to the NDE sizing uncertainties, licensees should be allowed to consider other NDE uncertainties (e.g., probability of detection, flaw density and location) in meeting the requirements of the rule as these uncertainties may affect the ability of a licensee to demonstrate compliance with the rule. As a result, the language in § 50.61a(e) was modified to allow licensees to account for the effects of NDE-related uncertainties in meeting the flaw size and density requirements of Tables 2 and 3. This requirement would be accomplished by requiring licensees to base their methodology to account for the NDE uncertainties on statistical data collected from ASME Code inspector qualification tests and any other tests that measure the difference between the actual flaw size and the size determined from the ultrasonic examination. Collecting, evaluating, and using data

from these tests will require extensive engineering judgment. Therefore, the methodology would have to be reviewed and approved by the Director of NRR.

Lastly, the commenters proposed to provide industry guidance to enable licensees to account for the effects of NDE uncertainties. The NRC determined that the rule language clearly states the information that must specifically be provided for NRC review and approval if licensees choose to account for NDE uncertainties. However, if industry guidance documents are developed, the NRC will consider them when submitted for review and approval.

IV. Section-by-Section Analysis

The following section-by-section analysis discusses the sections that are being modified as a result of this final rulemaking.

§50.8(b) - Information collection requirements: OMB approval

This paragraph is modified to include the amended information collection requirements as a result of this final rule.

§ 50.61 - Fracture toughness requirements for protection against pressurized thermal shock events

Section 50.61 contains the current requirements for PTS screening limits and embrittlement correlations. Paragraph (b) of this section is modified to reference § 50.61a as a voluntary alternative to compliance with the requirements of § 50.61. No changes are made to the current PTS screening criteria, embrittlement correlations, or any other related requirements in this section.

§ *50.61a – Alternate fracture toughness requirements for protection against pressurized thermal shock events*

A new § 50.61a is added. Section 50.61a contains PTS screening limits based on updated probabilistic fracture mechanics analyses. This section provides requirements on PTS analogous to that of § 50.61, fracture toughness requirements for protection against PTS events for PWRs. However, § 50.61a differs extensively in how the licensee determines the resistance to fractures initiating from different flaws at different locations in the vessel beltline, as well as in the fracture toughness screening criteria. The final rule requires quantifying PTS reference temperatures (RT_{MAX-X}) for flaws along axial weld fusion lines, plates, forgings, and circumferential weld fusion lines, and comparing the quantified value against the RT_{MAX-X} screening criteria. Although comparing quantified values to the screening criteria is also required by the current § 50.61, the new § 50.61a provides screening criteria that vary depending on material product form and vessel wall thickness. Further, the embrittlement correlation and the method of calculation of RT_{MAX-X} values in § 50.61a differ significantly from that in § 50.61 as described in the technical basis for this rule. The new embrittlement correlation was developed using multivariable surface-fitting techniques based on pattern recognition, understanding of the underlying physics, and engineering judgment. The embrittlement database used for this analysis was derived primarily from reactor vessel material surveillance data from operating reactors that are contained in the Power Reactor Embrittlement Data Base (PR-EDB) developed at Oak Ridge National Laboratory. The updated RT_{MAX-X} estimation procedures provide a better (compared to the existing regulation) method for estimating the fracture toughness of reactor vessel materials over the lifetime of the plant. However, if extensive mixed oxide (MOX) fuels with a high plutonium component are to be used, the neutron irradiation of the vessel material will contain more neutrons per unit energy produced and those neutrons will have higher energies. Extensive use of MOX fuel would result

in a change in the Reactor Core Fuel Assembly (RCFA) design. Thus, in accordance to § 50.90, licensees are required to submit a license amendment before changing the RCFA design. The § 50.61a final rule requires that licensees verify an appropriate RT_{MAX-X} value has been calculated for each reactor vessel beltline material considering plant-specific information that could affect the use of the model. A licensee using MOX fuel would use its surveillance data to meet the requirements of § 50.61a and must justify the applicability of the model expressed by Equations 5, 6, and 7 listed in the final rule.

§ 50.61a(a)

This paragraph contains definitions for terms used in § 50.61a. It explains that terms defined in § 50.61 have the same meaning in § 50.61a, unless otherwise noted.

§ 50.61a(b)

This paragraph sets forth the applicability of the final rule and specifies that its provisions apply only to those holders of operating licenses whose construction permits were issued before **[INSERT EFFECTIVE DATE OF FINAL RULE],** and whose reactor vessels were designed and fabricated to the 1998 Edition or earlier of the ASME Code. Both elements must be satisfied in order for a licensee to take advantage of § 50.61a. The rule does not apply to any combined license issued under Part 52 for two reasons: (1) the combined license would be issued after **[INSERT EFFECTIVE DATE OF FINAL RULE],** and (2) none of the reactor vessels for the nuclear power reactors covered by these combined licenses would have been designed and fabricated to the 1998 Edition or earlier of the ASME Code. The same logic also explains why § 50.61a would not apply to any design certification or manufacturing license issued under Part 52.

§ 50.61a(c)

This paragraph establishes the requirements governing NRC approval of a licensee's use of § 50.61a. The licensee has to make a formal request to the NRC via a license amendment, and would only be allowed to implement § 50.61a upon NRC approval. The license amendment request must provide information that includes: (1) calculations of the values of RT_{MAX-X} values as required by § 50.61a(c)(1); (2) examination and assessment of flaws discovered by ASME Code inspections as required by § 50.61a(c)(2); and (3) comparison of the RT_{MAX-X} values against the applicable screening criteria as required by § 50.61a(c)(3). In doing so, the licensee also would be required to use §§ 50.61a(e), (f) and (g) to perform the necessary calculations, comparisons, examinations, assessments, and analyses.

§ 50.61a(d)

This paragraph defines the requirements for subsequent examinations and flaw assessments after initial approval to use § 50.61a has been obtained under the requirements of § 50.61a(c). It also defines the required compensatory measures or analyses to be taken if a licensee determines that the screening criteria will be exceeded. Paragraph (d)(1) defines the requirements for subsequent RT_{MAX-X} assessments consistent with the requirements of §§ 50.61a(c)(1) and (c)(3). Paragraph (d)(2) defines the requirements for subsequent examination and flaw assessments using the requirements of § 50.61a(e). Paragraphs (d)(3) through (d)(7) define the requirements for implementing compensatory measures or plant-specific analyses should the value of RT_{MAX-X} be projected to exceed the PTS screening criteria in Table 1 of this section.

§ 50.61a(e)

This paragraph defines the requirements for verifying that the PTS screening criteria in § 50.61a are applicable to a particular reactor vessel. The final rule requires that the verification be based on an analysis of test results from ultrasonic examination of the reactor vessel beltline materials required by ASME Code, Section XI.

§ 50.61a(e)(1)

This paragraph establishes limits on flaw density and size distributions within the volume described in ASME Code, Section XI, Figures IWB-2500-1 and IWB-2500-2, and limited to a depth of approximately 1 inch from the clad-to-base metal interface or 10 percent of the vessel thickness, whichever is greater. Flaws in this inspection volume contribute approximately 97 to 99 percent to the TWCF at the screening limit.

The verification shall be performed line-by-line for Tables 2 and 3. For example, for the second line in Table 2, the licensee would tabulate all of the flaws detected in the relevant inspection volume in welds and would tally the number that have through-wall extents between the minimum (TWE_{MIN}) and maximum (TWE_{MAX}) values for line 2 (0.075 inches and 0.475 inches), would divide that total number by the number of thousands of inches of weld length examined to get a density, and would compare the resulting density to the limit in line 2, column 3 (which is 166.70 flaws per 1000 inches of weld metal). The licensee would then perform a similar analysis for line 3 in Table 2 by tallying the number of the flaws that have through-wall extents between the TWE_{MIN} and TWE_{MAX} values for line 3 (0.125 inches and 0.475 inches), would divide the total number by the number of thousands of inches of weld length examined to get a density, and would compare the resulting density to the limit in line 3, column 3 (which is 90.80 flaws per 1000 inches of weld metal). This process would be repeated for each line in the tables.

This paragraph allows licensees to adjust test results from the volumetric examination to account for the effects of NDE-related uncertainties. If test data is adjusted to account for NDE-related uncertainties, the methodology and statistical data used to account for these uncertainties must be submitted for review and approval by the Director of NRR.

This paragraph also states that if the licensee's flaw density and size distribution exceeds the values in Tables 2 and 3, a neutron fluence map would have to be submitted in accordance with § 50.61a(e)(6).

§§ 50.61a(e)(1)(i) and (e)(1)(ii)

These paragraphs describe the flaw density limits for welds and for plates and forgings, respectively.

§ 50.61a(e)(1)(iii)

This paragraph describes the specific ultrasonic examination information to be submitted to the NRC. This paragraph establishes the documenting requirement for axial and circumferential flaws with a through-wall extent equal to or greater than 0.075 inches. Licensees must document indications that have been observed through ultrasonic inspections intended to locate axially-oriented flaws as "axial" (i.e., an axial flaw would be one identified by an ultrasonic transducer oriented in the circumferential direction). All other indications may be categorized as "circumferential." The NRC will use this information to evaluate whether plant-specific information gathered in accordance with this rule suggests that the NRC should generically re-examine the technical basis for the rule.

§ 50.61a(e)(2)

This paragraph requires that licensees verify that clad-to-base metal interface flaws do not open to the inside surface of the vessel. These types of flaws could have a substantial effect on the TWCF.

§ 50.61a(e)(3)

This paragraph establishes limits for flaws that are between the clad-to-base metal interface and three-eights of the reactor vessel wall thickness from the interior surface. Flaws exceeding these limits could affect the TWCF. Flaws greater than three-eights of the reactor vessel wall thickness from the interior surface of the reactor vessel thickness do not contribute to the TWCF at the screening limit.

§ 50.61a(e)(4)

This paragraph establishes requirements to be met if flaws exceed the limits in §§ 50.61a(e)(1) and (e)(3), or open to the inside surface of the reactor vessel. This section requires an analysis to demonstrate that the reactor vessel would have a TWCF of less than $1x10^{-6}$ per reactor year. The analysis could be a complete, plant-specific, probabilistic fracture mechanics analysis or could be a simplified analysis of flaw size, orientation, location and embrittlement to demonstrate that the actual flaws in the reactor vessel are not in locations, and/or do not have orientations, that would cause the TWCF to be greater than $1x10^{-6}$ per reactor year. With specific regard to circumferentially-oriented flaws that exceed the limits of §§ 50.61a(e)(1) and (e)(3), it may be noted that even if a reactor pressure vessel has a circumferential weld at the RT_{MAX-CW} limits of Table 1, this weld only contributes $1x10^{-8}$ per reactor year to the TWCF predicted for the vessel. Licensees must comply with this if the requirements of §§ 50.61a(e)(1), (e)(2), and (e)(3) are not satisfied.

§ 50.61a(e)(5)

This paragraph describes the critical parameters to be addressed if flaws exceed the limits in §§ 50.61a(e)(1) and (e)(3) or if the flaws would open to the inside surface of the reactor vessel. This paragraph will be required to be implemented if the requirements of §§ 50.61a(e)(1), (e)(2), and (e)(3) are not satisfied.

§ 50.61a(e)(6)

This paragraph establishes the requirements for submitting a neutron fluence map if the flaw density and sizes are greater than those specified in Tables 2 and 3. Regulatory Guide 1.190 provides an acceptable methodology for determining the reactor vessel neutron fluence.

§ 50.61a(f)(1) through (f)(5)

These paragraphs define the process for calculating the values for the material properties (i.e., $RT_{MAX\text{-}X}$) for a particular reactor vessel. These values are based on the vessel's copper, manganese, phosphorus, and nickel weight percentages, reactor cold leg temperature, and neutron flux and fluence values, as well as the unirradiated RT_{NDT} of the product form in question.

§ 50.61a(f)(6)

This paragraph requires licensees to consider the plant-specific information that could affect the use of the embrittlement model established in the final rule.

§ 50.61a(f)(6)(i)

This paragraph establishes the requirements to perform data checks to determine if the surveillance data show a significantly different trend than what the embrittlement model in this rule predicts. Licensees are required to evaluate the surveillance for consistency with the embrittlement model by following the procedures specified by §§ 50.61a(f)(6)(ii), (f)(6)(iii), and (f)(6)(iv).

§ 50.61a(f)(6)(ii)

This paragraph establishes the requirements to perform an estimate of the mean deviation of the surveillance data set from the embrittlement model. The mean deviation for the surveillance data set must be compared to values given in Table 5 or Equation 10. The surveillance data analysis must follow the criteria in §§ 50.61a(f)(6)(v) and (f)(6)(vi).

§ 50.61a(f)(6)(iii)

This paragraph establishes the requirements to estimate the slope of the embrittlement model residuals (i.e., the difference between the measured and predicted value for a specific data point). The licensee must estimate the slope using Equation 11 and compare this value to the maximum permissible value in Table 6. This surveillance data analysis must follow the criteria in §§ 50.61a(f)(6)(v) and (f)(6)(vi).

§ 50.61a(f)(6)(iv)

This paragraph establishes the requirements to estimate an outlier deviation from the embrittlement model for the specific data set using Equations 8 and 12. The licensee must

compare the normalized residuals to the allowable values in Table 7. This surveillance data analysis must follow the criteria in §§ 50.61a(f)(6)(v) and (f)(6)(vi).

§ 50.61a(f)(6)(v)

This paragraph establishes the criteria to be satisfied in order to calculate the ΔT_{30} shift values.

§ 50.61a(f)(6)(vi)

This paragraph establishes the actions to be taken by a licensee if the criteria in § 50.61a(f)(6)(v) are not met. The licensee must submit an evaluation of the surveillance data and propose values for ΔT_{30}, considering their plant-specific surveillance data, for review and approval by the Director of NRR. The licensee must submit an evaluation of each surveillance capsule removed from the vessel after the submittal of the initial application for review and approval by the Director of NRR no later than 2 years after the capsule is withdrawn from the vessel.

§ 50.61a(g)

This paragraph provides the necessary equations and variables required by § 50.61a(f). These equations were calibrated to the surveillance database collected in accordance with the requirements of 10 CFR part 50, appendix H. This database contained data occupying the range of variables detailed in the table below.

Variable	Symbol	Units	Values Characterizing the Surveillance Database			
			Average	Standard Deviation	Minimum	Maximum
Neutron Fluence (E>1MeV)	φt	n/cm^2	1.24E+19	1.19E+19	9.26E+15	1.07E+20
Neutron Flux (E>1MeV)	φ	$n/cm^2/sec$	8.69E+10	9.96E+10	2.62E+08	1.63E+12
Irradiation Temperature	T	°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

Tables 1 through 7

Table 1 provides the PTS screening criteria for comparison with the licensee's calculated RT_{MAX-X} values. Tables 2 and 3 provide values to be used in § 50.61a(e). Tables 4 through 7 provide values to be used in § 50.61a(f).

V. Availability of Documents

The documents identified below are available to interested persons through one or more of the following methods, as indicated.

Public Document Room (PDR). The NRC PDR is located at 11555 Rockville Pike, Rockville, Maryland 20852.

Regulations.gov (Web). These documents may be viewed and downloaded electronically through the Federal eRulemaking Portal http://www.regulations.gov, Docket number NRC-2007-0008.

NRC's Electronic Reading Room (ERR). The NRC's public electronic reading room is located at www.nrc.gov/reading-rm.html.

Document	PDR	Web	**ERR** (ADAMS)
Federal Register Notice - Proposed Rule: Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events (RIN 3150-AI01), 72 FR 56275, October 3, 2007	x	NRC-2007-0008	ML072750659
Regulatory History for RIN 3150-AI01 Proposed Rulemaking Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events	x		ML072880444
Letter from Thomas P. Harrall, Jr., dated December 17, 2007, “Comments on Proposed Rule 10 CFR 50, Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events, RIN 3150-AI01” [Identified as Duke]	x	NRC-2007-0008	ML073521542
Letter from Jack Spanner, dated December 17, 2007, “10 CFR 50.55a Proposed Rulemaking Comments RIN 3150-AI01” [Identified as EPRI]	x	NRC-2007-0008	ML073521545
Letter from James H. Riley, dated December 17, 2007, “Proposed Rulemaking - Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events (RIN 3150-AI01), 72 FR 56275, October 3, 2007” [Identified as NEI]	x	NRC-2007-0008	ML073521543
Letter from Melvin L. Arey, dated December 17, 2007, “Transmittal of PWROG Comments on the NRC Proposed Rule on Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events, RIN 3150-AI01, PA-MSC-0232” [Identified as PWROG]	x	NRC-2007-0008	ML073521547

Document	PDR	Web	ERR (ADAMS)
Letter from T. Moser, dated December 17, 2007, “Strategic Teaming and Resource Sharing (STARS) Comments on RIN 3150-AI01, Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events, 72 FR 56275 (October 3,2007)” [Identified as STARS]	x	NRC-2007-0008	ML073610558
Federal Register Notice – Supplemental Proposed Rule: Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events (RIN 3150-AI01), 73 FR 46557 August 11, 2008	x	NRC-2007-0008	ML081440656
Supplemental Regulatory Analysis	x	NRC-2007-0008	ML081440673
Supplemental OMB Supporting Statement	x	NRC-2007-0008	ML081440736
Regulatory History Related to Supplemental Proposed Rule: Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events, 10 CFR 50.61a (RIN 3150-AI01)	x	NRC-2007-008	ML082740222
Email from Todd A. Henderson, FENOC, dated September 15, 2008, “RIN 3150-AI01: Comments on Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events” [Identified as FENOC]	x	NRC-2007-0008	ML082600288
Letter from Dennis E. Buschbaum, dated September 9, 2008, “Transmittal of PWROG Additional Comments on the NRC "Proposed Rule on Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events", RIN 3150-AI0l, PA-MSC0421” [Identified as PWROG2]	x	NRC-2007-0008	ML082550705
Letter from Jack Spanner, dated September 10, 2008, “Proposed Rulemaking Comments RIN 3150-AI01” [Identified as EPRI2]	x	NRC-2007-0008	ML082550710
“Statistical Procedures for Assessing Surveillance Data for 10 CFR Part 50.61a”	x		ML081290654

Document	PDR	Web	ERR (ADAMS)
"A Physically Based Correlation of Irradiation Induced Transition Temperature Shifts for RPV Steel"	x		ML081000630
NUREG-1806, "Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limits in the PTS Rule (10 CFR 50.61): Summary Report,"	x		ML061580318
NUREG-1874, "Recommended Screening Limits for Pressurized Thermal Shock (PTS)"	x		ML070860156
Memorandum from Elliot to Mitchell, dated April 3, 2007, "Development of Flaw Size Distribution Tables for Draft Proposed Title 10 of the Code of Federal Regulations (10 CFR) 50.61a,"	x		ML070950392
Memo from J. Uhle, dated May 15, 2008, "Embrittlement Trend Curve Development for Reactor Pressure Vessel Materials"	x		ML081120253
Draft "Technical Basis for Revision of Regulatory Guide 1.99: NRC Guidance on Methods to Estimate the Effects of Radiation Embrittlement on the Charpy V-Notch Impact Toughness of Reactor Vessel Materials"	x		ML081120289
"Comparison of the Predictions of RM-9 to the IVAR and RADAMO Databases"	x		ML081120365
Memo from M. Erickson Kirk, dated December 12, 2007, "New Data from Boiling Water Reactor Vessel Integrity Program (BWRVIP) Integrated Surveillance Project (ISP)"	x		ML081120380
"Further Evaluation of High Fluence Data"	x		ML081120600
Regulatory Guide (RG) 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Analysis Reports for Pressurized Water Reactors"	x		ML003740028
Final OMB Supporting Statement Related to Final Rule: Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events, 10 CFR 50.61a (RIN 3150-AI01)	x	NRC-2007-0008	ML083500231

Document	PDR	Web	**ERR** (ADAMS)
Regulatory Analysis Related to Final Rule: Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events, 10 CFR 50.61a (RIN 3150-AI01)	x	NRC-2007-0008	ML083500225
Summary and Analysis of Public Comments related to the Alternate Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events	x	NRC-2007-0008	ML083500218

VI. Agreement State Compatibility

Under the "Policy Statement on Adequacy and Compatibility of Agreement States Programs," approved by the Commission on June 20, 1997, and published in the *Federal Register* (62 FR 46517) on September 3, 1997, this rule is classified as compatibility category "NRC." Agreement State Compatibility is not required for Category "NRC" regulations. The NRC program elements in this category are those that relate directly to areas of regulation reserved to the NRC by the Atomic Energy Act or the provisions of Title 10 of the *Code of Federal Regulations*. Although an Agreement State may not adopt program elements reserved to NRC, it may wish to inform its licensees of certain requirements via a mechanism that is consistent with the particular State's administrative procedure laws. Category "NRC" regulations do not confer regulatory authority on the State.

VII. Voluntary Consensus Standards

The National Technology Transfer and Advancement Act of 1995, Pub. L. 104-113, requires that Federal agencies use technical standards that are developed or adopted by voluntary consensus standards bodies unless using such a standard is inconsistent with applicable law or is otherwise impractical.

The NRC determined that there is only one technical standard developed that could be used for characterizing the embrittlement correlations. That standard is the American Society for Testing and Materials (ASTM) standard E-900, “Standard Guide for Predicting Radiation-Induced Temperature Transition Shift in Reactor Vessel Materials.” This standard contains a different embrittlement correlation than that of this final rule. However, the correlation developed by the NRC has been more recently calibrated to available data. As a result, ASTM standard E-900 is not a practical candidate for application in the technical basis for the final rule because it does not represent the broad range of conditions necessary to justify a revision to the regulations.

The ASME Code requirements are used as part of the volumetric examination analysis requirements of the final rule. ASTM Standard Practice E 185, “Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels,” is incorporated by reference in 10 CFR part 50, appendix H and used to determine 30-foot-pound transition temperatures. These standards were selected for use in the final rule based on their use in other regulations within 10 CFR part 50 and their applicability to the subject of the desired requirements.

VIII. Finding of No Significant Environmental Impact: Availability

The Commission has determined under the National Environmental Policy Act of 1969, as amended, and the Commission's regulations in 10 CFR part 51, Subpart A, that this rule is not a major Federal action significantly affecting the quality of the human environment and, therefore, an environmental impact statement is not required.

The determination of this environmental assessment is that there will be no significant offsite impact to the public from this action. Section 50.61a would maintain the same functional requirements for the facility as the existing PTS rule in § 50.61. This final rule establishes

screening criteria, limiting levels of embrittlement beyond which plant operation cannot continue without further plant-specific evaluation or modifications. This provides reasonable assurance that licensees operating below the screening criteria could endure a PTS event without fracture of vessel materials, thus assuring integrity of the reactor pressure vessel. In addition, the final rule is risk-informed and sufficient safety margins are maintained to ensure that any potential increases in core damage frequency and large early release frequency resulting from implementation of § 50.61a are negligible. The final rule will not significantly increase the probability or consequences of accidents, result in changes being made in the types of any effluents that may be released off site, or result in a significant increase in occupational or public radiation exposure. Therefore, there are no significant radiological environmental impacts associated with this final rule. Nonradiological plant effluents are not affected as a result of this final rule.

The NRC requested the views of the States on the environmental assessment for this rule. No comments were received. Therefore, the environmental assessment determination published on October 3, 2007 (72 FR 56275) remains unchanged.

IX. Paperwork Reduction Act Statement

This final rule contains new or amended information collection requirements contained in 10 CFR part 50, that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501, *et seq*). These requirements were approved by the Office of Management and Budget (OMB), approval number 3150-0011.

The burden to PWR licensees using the requirements of 10 CFR 50.61a in lieu of the requirements of 10 CFR 50.61 for these information collections is estimated to average 363 hours per response. This includes the time for reviewing instructions, searching existing data

sources, gathering and maintaining the data needed, and completing and reviewing the information collection.

Send comments on any aspect of these information collections, including suggestions for reducing the burden, to the Records and FOIA/Privacy Services Branch (T-5 F53), 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 20503, or by e-mail to Nathan_J._Frey@omb.eop.gov.

Public Protection Notification

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.

X. Regulatory Analysis

The NRC has prepared a regulatory analysis of this regulation. The analysis examines the costs and benefits of the alternatives considered by the NRC. The NRC concluded that implementing the final rule would provide savings to licensees projected to exceed the PTS screening criteria established in § 50.61 in their plant lifetimes. Availability of the regulatory analysis is provided in Section V, “Availability of Documents” of this document. No public comments were received on the proposed or supplemental regulatory analyses.

XI. Regulatory Flexibility Act Certification

In accordance with the Regulatory Flexibility Act (5 U.S.C. 605(b)), the NRC certifies that this rule would not have a significant economic impact on a substantial number of small entities.

This final rule would affect only the licensing and operation of currently operating nuclear power plants. The companies that own these plants do not fall within the scope of the definition of "small entities" set forth in the Regulatory Flexibility Act or the size standards established by the NRC (10 CFR 2.810).

XII. Backfit Analysis

The NRC has determined that the requirements in this final rule would not constitute backfitting as defined in 10 CFR 50.109(a)(1). Therefore, a backfit analysis has not been prepared for this rule.

The requirements of the current PTS rule, 10 CFR 50.61, would continue to apply to all PWR licensees and would not change as a result of this final rule. The requirements of the alternate PTS rule would not be required, but could be used by current PWR licensees at their option. Current PWR licensees choosing to implement the alternate PTS rule are required to comply with its requirements as an alternative to complying with the requirements of the current PTS rule. Because the alternate PTS rule would not be mandatory for any PWR licensee, but rather could be voluntarily implemented, the NRC has determined that this rulemaking would not constitute backfitting.

XIII. Congressional Review Act

Under the Congressional Review Act of 1996, the NRC has determined that this action is not a major rule and has verified this determination with the Office of Information and Regulatory Affairs of the OMB.

List of Subjects for 10 CFR Part 50

Antitrust, Classified information, Criminal penalties, Fire protection, Intergovernmental relations, Nuclear power plants and reactors, Radiation protection, Reactor siting criteria, Reporting and recordkeeping requirements.

For the reasons set out in the preamble and under the authority of the Atomic Energy Act of 1954, as amended; the Energy Reorganization Act of 1974, as amended; and 5 U.S.C. 552 and 553; the NRC is adopting the following amendments to 10 CFR part 50.

PART 50 - DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

1. The authority citation for Part 50 continues to read as follows:

Authority: Secs. 102, 103, 104, 105, 161, 182, 183, 186, 189, 68 Stat. 936, 937, 938, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 444, as amended (42 U.S.C. 2132, 2133, 2134, 2135, 2201, 2232, 2233, 2236, 2239, 2282); secs. 201, as amended, 202, 206, 88 Stat. 1242, as amended, 1244, 1246 (42 U.S.C. 5841, 5842, 5846); sec. 1704, 112 Stat. 2750 (44 U.S.C. 3504 note); Energy policy Act of 2005, Pub. L. No. 109-58, 119 Stat. 194 (2005). Section 50.7 also issued under Pub. L. 95-601, sec. 10, 92 Stat. 2951 as amended by Pub. L. 102-486, sec. 2902, 106 Stat. 3123 (42 U.S.C. 5841). Section 50.10 also issued under secs. 101, 185, 68 Stat. 955, as amended (42 U.S.C. 2131, 2235); sec. 102, Pub. L. 91-190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.13, 50.54(dd), and 50.103 also issued under sec. 108, 68 Stat. 939, as amended (42 U.S.C. 2138).

Sections 50.23, 50.35, 50.55, and 50.56 also issued under sec. 185, 68 Stat. 955 (42 U.S.C. 2235). Sections 50.33a, 50.55a and Appendix Q also issued under sec. 102, Pub. L. 91-190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.34 and 50.54 also issued under sec. 204, 88 Stat. 1245 (42 U.S.C. 5844). Sections 50.58, 50.91, and 50.92 also issued under Pub. L. 97-415, 96 Stat. 2073 (42 U.S.C. 2239). Section 50.78 also issued under sec. 122, 68 Stat. 939

(42 U.S.C. 2152). Sections 50.80 - 50.81 also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C. 2234). Appendix F also issued under sec. 187, 68 Stat. 955 (42 U.S.C. 2237).

2. Section 50.8(b) is revised to read as follows:

§ 50.8 Information collection requirements: OMB approval.

* * * * *

(b) The approved information collection requirements contained in this part appear in §§ 50.30, 50.33, 50.34, 50.34a, 50.35, 50.36, 50.36a, 50.36b, 50.44, 50.46, 50.47, 50.48, 50.49, 50.54, 50.55, 50.55a, 50.59, 50.60, 50.61, 50.61a, 50.62, 50.63, 50.64, 50.65, 50.66, 50.68, 50.69, 50.70, 50.71, 50.72, 50.74, 50.75, 50.80, 50.82, 50.90, 50.91, 50.120, and appendices A, B, E, G, H, I, J, K, M, N,O, Q, R, and S to this part.

* * * * *

3. In § 50.61, paragraph (b)(1) is revised to read as follows:

§ 50.61 Fracture toughness requirements for protection against pressurized thermal shock events.

* * * * *

(b) *Requirements*. (1) For each pressurized water nuclear power reactor for which an operating license has been issued under this part or a combined license issued under Part 52 of this chapter, other than a nuclear power reactor facility for which the certification required under § 50.82(a)(1) has been submitted, the licensee shall have projected values of RT_{PTS} or $RT_{MAX\text{-}X}$, accepted by the NRC, for each reactor vessel beltline material. For pressurized water nuclear power reactors for which a construction permit was issued under this part before **[INSERT EFFECTIVE DATE OF FINAL RULE]** and whose reactor vessel was designed and fabricated to the 1998 Edition or earlier of the ASME Code, the projected values must be in accordance with

this section or § 50.61a. For pressurized water nuclear power reactors for which a construction permit is issued under this part after **[INSERT EFFECTIVE DATE OF FINAL RULE]** and whose reactor vessel is designed and fabricated to an ASME Code after the 1998 Edition, or for which a combined license is issued under Part 52, the projected values must be in accordance with this section. When determining compliance with this section, the assessment of RT_{PTS} must use the calculation procedures described in paragraph (c)(1) and perform the evaluations described in paragraphs (c)(2) and (c)(3) of this section. The assessment must specify the bases for the projected value of RT_{PTS} for each vessel beltline material, including the assumptions regarding core loading patterns, and must specify the copper and nickel contents and the fluence value used in the calculation for each beltline material. This assessment must be updated whenever there is a significant[2] change in projected values of RT_{PTS}, or upon request for a change in the expiration date for operation of the facility.

* * * * *

4. Section 50.61a is added to read as follows:

§ 50.61a Alternate fracture toughness requirements for protection against pressurized thermal shock events.

(a) *Definitions*. Terms in this section have the same meaning as those presented in 10 CFR 50.61(a), with the exception of the term "ASME Code."

(1) *ASME Code* means the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Division I, "Rules for the Construction of Nuclear Power Plant Components," and Section XI, Division I, "Rules for Inservice Inspection of Nuclear Power Plant

[2] Changes to RT_{PTS} values are considered significant if either the previous value or the current value, or both values, exceed the screening criterion before the expiration of the operating license or the combined license under Part 52 of this chapter, including any renewed term, if applicable for the plant.

Components," edition and addenda and any limitations and modifications thereof as specified in § 50.55a.

(2) $RT_{MAX\text{-}AW}$ means the material property which characterizes the reactor vessel's resistance to fracture initiating from flaws found along axial weld fusion lines. $RT_{MAX\text{-}AW}$ is determined under the provisions of paragraph (f) of this section and has units of °F.

(3) $RT_{MAX\text{-}PL}$ means the material property which characterizes the reactor vessel's resistance to fracture initiating from flaws found in plates in regions that are not associated with welds found in plates. $RT_{MAX\text{-}PL}$ is determined under the provisions of paragraph (f) of this section and has units of °F.

(4) $RT_{MAX\text{-}FO}$ means the material property which characterizes the reactor vessel's resistance to fracture initiating from flaws in forgings that are not associated with welds found in forgings. $RT_{MAX\text{-}FO}$ is determined under the provisions of paragraph (f) of this section and has units of °F.

(5) $RT_{MAX\text{-}CW}$ means the material property which characterizes the reactor vessel's resistance to fracture initiating from flaws found along the circumferential weld fusion lines. $RT_{MAX\text{-}CW}$ is determined under the provisions of paragraph (f) of this section and has units of °F.

(6) $RT_{MAX\text{-}X}$ means any or all of the material properties $RT_{MAX\text{-}AW}$, $RT_{MAX\text{-}PL}$, $RT_{MAX\text{-}FO}$, $RT_{MAX\text{-}CW}$, or sum of $RT_{MAX\text{-}AW}$ and $RT_{MAX\text{-}PL,}$ for a particular reactor vessel.

(7) φt means fast neutron fluence for neutrons with energies greater than 1.0 MeV. φt is utilized under the provisions of paragraph (g) of this section and has units of n/cm^2.

(8) φ means average neutron flux for neutrons with energies greater than 1.0 MeV. φ is utilized under the provisions of paragraph (g) of this section and has units of $n/cm^2/sec$.

(9) ΔT_{30} means the shift in the Charpy V-notch transition temperature at the 30 ft-lb energy level produced by irradiation. The ΔT_{30} value is utilized under the provisions of paragraph (g) of this section and has units of °F.

(10) *Surveillance data* means any data that demonstrates the embrittlement trends for the beltline materials, including, but not limited to, surveillance programs at other plants with or without a surveillance program integrated under 10 CFR part 50, appendix H.

(11) T_C means cold leg temperature under normal full power operating conditions, as a time-weighted average from the start of full power operation through the end of licensed operation. T_C has units of °F.

(12) *CRP* means the copper rich precipitate term in the embrittlement model from this section. The CRP term is defined in paragraph (g) of this section.

(13) *MD* means the matrix damage term in the embrittlement model for this section. The MD term is defined in paragraph (g) of this section.

(b) *Applicability*. The requirements of this section apply to each holder of an operating license for a pressurized water nuclear power reactor whose construction permit was issued before **[INSERT EFFECTIVE DATE OF FINAL RULE]** and whose reactor vessel was designed and fabricated to the ASME Boiler and Pressure Vessel Code, 1998 Edition or earlier. The requirements of this section may be implemented as an alternative to the requirements of 10 CFR 50.61.

(c) *Request for Approval*. Before the implementation of this section, each licensee shall submit a request for approval in the form of an application for a license amendment in accordance with § 50.90 together with the documentation required by paragraphs (c)(1), (c)(2), and (c)(3) of this section for review and approval by the Director of the Office of Nuclear Reactor Regulation (Director). The application must be submitted for review and approval by the Director at least three years before the limiting RT_{PTS} value calculated under 10 CFR 50.61 is projected to exceed the PTS screening criteria in 10 CFR 50.61 for plants licensed under this part.

(1) Each licensee shall have projected values of RT_{MAX-X} for each reactor vessel beltline material for the EOL fluence of the material. The assessment of RT_{MAX-X} values must use the calculation procedures given in paragraphs (f) and (g) of this section. The assessment must specify the bases for the projected value of RT_{MAX-X} for each reactor vessel beltline material, including the assumptions regarding future plant operation (e.g., core loading patterns, projected capacity factors); the copper (Cu), phosphorus (P), manganese (Mn), and nickel (Ni) contents; the reactor cold leg temperature (T_C); and the neutron flux and fluence values used in the calculation for each beltline material. Assessments performed under paragraphs (f)(6) and (f)(7) of this section, shall be submitted by the licensee to the Director in its license amendment application to utilize § 50.61a.

(2) Each licensee shall perform an examination and an assessment of flaws in the reactor vessel beltline as required by paragraph (e) of this section. The licensee shall verify that the requirements of paragraphs (e), (e)(1), (e)(2), and (e)(3) of this section have been met. The licensee must submit to the Director, in its application to use § 50.61a, the adjustments made to the volumetric test data to account for NDE-related uncertainties as described in paragraph (e)(1) of this section, all information required by paragraph (e)(1)(iii) of this section, and, if applicable, analyses performed under paragraphs (e)(4), (e)(5) and (e)(6) of this section.

(3) Each licensee shall compare the projected RT_{MAX-X} values for plates, forgings, axial welds, and circumferential welds to the PTS screening criteria in Table 1 of this section, for the purpose of evaluating a reactor vessel's susceptibility to fracture due to a PTS event. If any of the projected RT_{MAX-X} values are greater than the PTS screening criteria in Table 1 of this section, then the licensee may propose the compensatory actions or plant-specific analyses as required in paragraphs (d)(3) through (d)(7) of this section, as applicable, to justify operation beyond the PTS screening criteria in Table 1 of this section.

(d) *Subsequent Requirements*. Licensees who have been approved to use 10 CFR 50.61a under the requirements of paragraph (c) of this section shall comply with the requirements of this paragraph.

(1) Whenever there is a significant change in projected values of RT_{MAX-X}, so that the previous value, the current value, or both values, exceed the screening criteria before the expiration of the plant operating license; or upon the licensee's request for a change in the expiration date for operation of the facility; a re-assessment of RT_{MAX-X} values documented consistent with the requirements of paragraph (c)(1) and (c)(3) of this section must be submitted in the form of a license amendment for review and approval by the Director. If the surveillance data used to perform the re-assessment of RT_{MAX-X} values meet the requirements of paragraph (f)(6)(v) of this section, the licensee shall submit the data and the results of the analysis of the data to the Director for review and approval within one year after the capsule is withdrawn from the vessel. If the surveillance data meet the requirements of paragraph (f)(6)(vi) of this section, the licensee shall submit the data, the results of the analysis of the data, and proposed ΔT_{30} and RT_{MAX-X} values considering the surveillance data in the form of a license amendment to the Director for review and approval within two years after the capsule is withdrawn from the vessel. If the Director does not approve the assessment of RT_{MAX-X} values, then the licensee shall perform the actions required in paragraphs (d)(3) through (d)(7) of this section, as necessary, before operation beyond the PTS screening criteria in Table 1 of this section.

(2) The licensee shall verify that the requirements of paragraphs (e), (e)(1), (e)(2), and (e)(3) of this section have been met. The licensee must submit, within 120 days after completing a volumetric examination of reactor vessel beltline materials as required by ASME Code, Section XI, the adjustments made to the volumetric test data to account for NDE-related uncertainties as described in paragraph (e)(1) of this section and all information required by paragraph (e)(1)(iii) of this section in the form of a license amendment for review and approval

by the Director. If a licensee is required to implement paragraphs (e)(4), (e)(5), and (e)(6) of this section, the information required in these paragraphs must be submitted in the form of a license amendment for review and approval by the Director within one year after completing a volumetric examination of reactor vessel materials as required by ASME Code, Section XI.

(3) If the value of RT_{MAX-X} is projected to exceed the PTS screening criteria, then the licensee shall implement those flux reduction programs that are reasonably practicable to avoid exceeding the PTS screening criteria. The schedule for implementation of flux reduction measures may take into account the schedule for review and anticipated approval by the Director of detailed plant-specific analyses which demonstrate acceptable risk with RT_{MAX-X} values above the PTS screening criteria due to plant modifications, new information, or new analysis techniques.

(4) If the analysis required by paragraph (d)(3) of this section indicates that no reasonably practicable flux reduction program will prevent the RT_{MAX-X} value for one or more reactor vessel beltline materials from exceeding the PTS screening criteria, then the licensee shall perform a safety analysis to determine what, if any, modifications to equipment, systems, and operation are necessary to prevent the potential for an unacceptably high probability of failure of the reactor vessel as a result of postulated PTS events. In the analysis, the licensee may determine the properties of the reactor vessel materials based on available information, research results and plant surveillance data, and may use probabilistic fracture mechanics techniques. This analysis and the description of the modifications must be submitted to the Director in the form of a license amendment at least three years before RT_{MAX-X} is projected to exceed the PTS screening criteria.

(5) After consideration of the licensee's analyses, including effects of proposed corrective actions, if any, submitted under paragraphs (d)(3) and (d)(4) of this section, the Director may, on a case-by-case basis, approve operation of the facility with RT_{MAX-X} values in

excess of the PTS screening criteria. The Director will consider factors significantly affecting the potential for failure of the reactor vessel in reaching a decision. The Director shall impose the modifications to equipment, systems and operations described to meet paragraph (d)(4) of this section.

(6) If the Director concludes, under paragraph (d)(5) of this section, that operation of the facility with RT_{MAX-X} values in excess of the PTS screening criteria cannot be approved on the basis of the licensee's analyses submitted under paragraphs (d)(3) and (d)(4) of this section, then the licensee shall request a license amendment, and receive approval by the Director, before any operation beyond the PTS screening criteria. The request must be based on modifications to equipment, systems, and operation of the facility in addition to those previously proposed in the submitted analyses that would reduce the potential for failure of the reactor vessel due to PTS events, or on further analyses based on new information or improved methodology. The licensee must show that the proposed alternatives provide reasonable assurance of adequate protection of the public health and safety.

(7) If the limiting RT_{MAX-X} value of the facility is projected to exceed the PTS screening criteria and the requirements of paragraphs (d)(3) through (d)(6) of this section cannot be satisfied, the reactor vessel beltline may be given a thermal annealing treatment under the requirements of § 50.66 to recover the fracture toughness of the material. The reactor vessel may be used only for that service period within which the predicted fracture toughness of the reactor vessel beltline materials satisfy the requirements of paragraphs (d)(1) through (d)(6) of this section, with RT_{MAX-X} values accounting for the effects of annealing and subsequent irradiation.

(e) *Examination and Flaw Assessment Requirements*. The volumetric examination results evaluated under paragraphs (e)(1), (e)(2), and (e)(3) of this section must be acquired using procedures, equipment and personnel that have been qualified under the ASME Code,

Section XI, Appendix VIII, Supplement 4 and Supplement 6, as specified in 10 CFR 50.55a(b)(2)(xv).

(1) The licensee shall verify that the flaw density and size distributions within the volume described in ASME Code, Section XI,[1] Figures IWB-2500-1 and IWB-2500-2 and limited to a depth from the clad-to-base metal interface of 1-inch or 10 percent of the vessel thickness, whichever is greater, do not exceed the limits in Tables 2 and 3 of this section based on the test results from the volumetric examination. The values in Tables 2 and 3 represent actual flaw sizes. Test results from the volumetric examination may be adjusted to account for the effects of NDE-related uncertainties. The methodology to account for NDE-related uncertainties must be based on statistical data from the qualification tests and any other tests that measure the difference between the actual flaw size and the NDE detected flaw size. Licensees who adjust their test data to account for NDE-related uncertainties to verify conformance with the values in Tables 2 and 3 shall prepare and submit the methodology used to estimate the NDE uncertainty, the statistical data used to adjust the test data and an explanation of how the data was analyzed for review and approval by the Director in accordance with paragraphs (c)(2) and (d)(2) of this section. The verification of the flaw density and size distributions shall be performed line-by-line for Tables 2 and 3. If the flaw density and size distribution exceeds the limitations specified in Tables 2 and 3 of this section, the licensee shall perform the analyses required by paragraph (e)(4) of this section. If analyses are required in accordance with paragraph (e)(4) of this section, the licensee must address the effects on through-wall crack frequency (TWCF) in accordance with paragraph (e)(5) of this section and must prepare and submit a neutron fluence map in accordance with the requirements of paragraph (e)(6) of this section.

[1] For forgings susceptible to underclad cracking the determination of the flaw density for that forging from the licensee's inspection shall exclude those indications identified as underclad cracks.

(i) The licensee shall determine the allowable number of weld flaws in the reactor vessel beltline by multiplying the values in Table 2 of this section by the total length of the reactor vessel beltline welds that were volumetrically inspected and dividing by 1000 inches of weld length.

(ii) The licensee shall determine the allowable number of plate or forging flaws in their reactor vessel beltline by multiplying the values in Table 3 of this section by the total surface area of the reactor vessel beltline plates or forgings that were volumetrically inspected and dividing by 1000 square inches.

(iii) For each flaw detected in the inspection volume described in paragraph (e)(1) with a through-wall extent equal to or greater than 0.075 inches, the licensee shall document the dimensions of the flaw, including through-wall extent and length, whether the flaw is axial or circumferential in orientation and its location within the reactor vessel, including its azimuthal and axial positions and its depth embedded from the clad-to-base metal interface.

(2) The licensee shall identify, as part of the examination required by paragraph (c)(2) of this section and any subsequent ASME Code, Section XI ultrasonic examination of the beltline welds, any flaws within the inspection volume described in paragraph (e)(1) of this section that are equal to or greater than 0.075 inches in through-wall depth, axially-oriented, and located at the clad-to-base metal interface. The licensee shall verify that these flaws do not open to the vessel inside surface using surface or visual examination technique capable of detecting and characterizing service induced cracking of the reactor vessel cladding.

(3) The licensee shall verify, as part of the examination required by paragraph (c)(2) of this section and any subsequent ASME Code, Section XI ultrasonic examination of the beltline welds, that all flaws between the clad-to-base metal interface and three-eights of the reactor vessel thickness from the interior surface are within the allowable values in ASME Code, Section XI, Table IWB-3510-1.

(4) The licensee shall perform analyses to demonstrate that the reactor vessel will have a TWCF of less than $1x10^{-6}$ per reactor year if the ASME Code, Section XI volumetric examination required by paragraph (c)(2) or (d)(2) of this section indicates any of the following:

(i) The flaw density and size in the inspection volume described in paragraph (e)(1) exceed the limits in Tables 2 or 3 of this section;

(ii) There are axial flaws that penetrate through the clad into the low alloy steel reactor vessel shell, at a depth equal to or greater than 0.075 inches in through-wall extent from the clad-to-base metal interface; or

(iii) Any flaws between the clad-to-base metal interface and three-eights[2] of the vessel thickness exceed the size allowable in ASME Code, Section XI, Table IWB-3510-1.

(5) The analyses required by paragraph (e)(4) of this section must address the effects on TWCF of the known sizes and locations of all flaws detected by the ASME Code, Section XI, Appendix VIII, Supplement 4 and Supplement 6 ultrasonic examination out to three-eights of the vessel thickness from the inner surface, and may also take into account other reactor vessel-specific information, including fracture toughness information.

(6) For all flaw assessments performed in accordance with paragraph (e)(4) of this section, the licensee shall prepare and submit a neutron fluence map, projected to the date of license expiration, for the reactor vessel beltline clad-to-base metal interface and indexed in a manner that allows the determination of the neutron fluence at the location of the detected flaws.

(f) *Calculation of RT_{MAX-X} values*. Each licensee shall calculate RT_{MAX-X} values for each reactor vessel beltline material using φt. The neutron flux (φ[t]), must be calculated using a

[2] Because flaws greater than three-eights of the vessel wall thickness from the inside surface do not contribute to TWCF, flaws greater than three-eights of the vessel wall thickness from the inside surface need not be analyzed for their contribution to PTS.

methodology that has been benchmarked to experimental measurements and with quantified uncertainties and possible biases[3].

(1) The values of RT_{MAX-AW}, RT_{MAX-PL}, RT_{MAX-FO}, and RT_{MAX-CW} must be determined using Equations 1 through 4 of this section. When calculating RT_{MAX-AW} using Equation 1, RT_{MAX-AW} is the maximum value of ($RT_{NDT(U)}$ + ΔT_{30}) for the weld and for the adjoining plates. When calculating RT_{MAX-CW} using Equation 4, RT_{MAX-CW} is the maximum value of ($RT_{NDT(U)}$ + ΔT_{30}) for the circumferential weld and for the adjoining plates or forgings.

(2) The values of ΔT_{30} must be determined using Equations 5, 6 and 7 of this section, unless the conditions specified in paragraph (f)(6)(v) of this section are not met, for each axial weld, plate, forging, and circumferential weld. The ΔT_{30} value for each axial weld calculated as specified by Equation 1 of this section must be calculated for the maximum fluence ($\varphi t_{AXIAL-WELD}$) occurring along a particular axial weld at the clad-to-base metal interface. The ΔT_{30} value for each plate calculated as specified by Equation 1 of this section must also be calculated using the same value of $\varphi t_{AXIAL-WELD}$ used for the axial weld. The ΔT_{30} values in Equation 1 shall be calculated for the weld itself and each adjoining plate. The ΔT_{30} value for each plate or forging calculated as specified by Equations 2 and 3 of this section must be calculated for the maximum fluence (φt_{MAX}) occurring at the clad-to-base metal interface over the entire area of each plate or forging. In Equation 4, the fluence ($\varphi t_{WELD-CIRC}$) value used for calculating the plate, forging, and circumferential weld ΔT_{30} value is the maximum fluence occurring for each material along the circumferential weld at the clad-to-base metal interface. The ΔT_{30} values in Equation 4 shall be calculated for the circumferential weld and for the adjoining plates or forgings. If the conditions specified in paragraph (f)(6)(v) of this section are not met, licensees must propose ΔT_{30} and RT_{MAX-X} values in accordance with paragraph (f)(6)(vi) of this section.

[3] Regulatory Guide 1.190 dated March 2001, establishes acceptable methods for determining neutron flux.

(3) The values of Cu, Mn, P, and Ni in Equations 6 and 7 of this section must represent the best estimate values for the material. 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[4] as shown in Table 4 of this section for P and Mn, must be used.

(4) The values of $RT_{NDT(U)}$ must be evaluated according to the procedures in the ASME Code, Section III, paragraph NB-2331. If any other method is used for this evaluation, the licensee shall submit the proposed method for review and approval by the Director along with the calculation of RT_{MAX-X} values required in paragraph (c)(1) of this section.

(i) If a measured value of $RT_{NDT(U)}$ is not available, a generic mean value of $RT_{NDT(U)}$ for the class[5] of material must be used if there are sufficient test results to establish a mean.

(ii) The following generic mean values of $RT_{NDT(U)}$ must be used unless justification for different values is provided: 0°F for welds made with Linde 80 weld flux; and -56°F for welds made with Linde 0091, 1092, and 124 and ARCOS B-5 weld fluxes.

(5) The value of T_c in Equation 6 of this section must represent the time-weighted average of the reactor cold leg temperature under normal operating full power conditions from the beginning of full power operation through the end of licensed operation.

(6) The licensee shall verify that an appropriate RT_{MAX-X} value has been calculated for each reactor vessel beltline material by considering plant-specific information that could affect

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[4] Data from reactor vessels fabricated to the same material specification in the same shop as the vessel in question and in the same time is an example of "generic data."

[5] The class of material for estimating $RT_{NDT(U)}$ must be determined by the type of welding flux (Linde 80, or other) for welds or by the material specification for base metal.

the use of the model (i.e., Equations 5, 6 and 7) of this section for the determination of a material's ΔT_{30} value.

(i) The licensee shall evaluate the results from a plant-specific or integrated surveillance program if the surveillance data satisfy the criteria described in paragraphs (f)(6)(i)(A) and (f)(6)(i)(B) of this section:

(A) The surveillance material must be a heat-specific match for one or more of the materials for which RT_{MAX-X} is being calculated. The 30-foot-pound transition temperature must be determined as specified by the requirements of 10 CFR part 50, appendix H.

(B) If three or more surveillance data points measured at three or more different neutron fluences exist for a specific material, the licensee shall determine if the surveillance data show a significantly different trend than the embrittlement model predicts. This must be achieved by evaluating the surveillance data for consistency with the embrittlement model by following the procedures specified by paragraphs (f)(6)(ii), (f)(6)(iii), and (f)(6)(iv) of this section. If fewer than three surveillance data points exist for a specific material, then the embrittlement model must be used without performing the consistency check.

(ii) The licensee shall estimate the mean deviation from the embrittlement model for the specific data set (i.e., a group of surveillance data points representative of a given material). The mean deviation from the embrittlement model for a given data set must be calculated using Equations 8 and 9 of this section. The mean deviation for the data set must be compared to the maximum heat-average residual given in Table 5 or derived using Equation 10 of this section. The maximum heat-average residual is based on the material group into which the surveillance material falls and the number of surveillance data points. For surveillance data sets with greater than 8 data points, the maximum credible heat-average residual must be calculated using Equation 10 of this section. The value of σ used in Equation 10 of this section must be obtained from Table 5 of this section.

(iii) The licensee shall estimate the slope of the embrittlement model residuals (estimated using Equation 8) plotted as a function of the base 10 logarithm of neutron fluence for the specific data set. The licensee shall estimate the T-statistic for this slope (T_{SURV}) using Equation 11 and compare this value to the maximum permissible T-statistic (T_{MAX}) in Table 6. For surveillance data sets with greater than 15 data points, the T_{MAX} value must be calculated using Student's T distribution with a significance level (α) of 1 percent for a one-tailed test.

(iv) The licensee shall estimate the two largest positive deviations (i.e., outliers) from the embrittlement model for the specific data set using Equations 8 and 12. The licensee shall compare the largest normalized residual (r *) to the appropriate allowable value from the third column in Table 7 and the second largest normalized residual to the appropriate allowable value from the second column in Table 7.

(v) The ΔT_{30} value must be determined using Equations 5, 6, and 7 of this section if all three of the following criteria are satisfied:

(A) the mean deviation from the embrittlement model for the data set is equal to or less than the value in Table 5 or the value derived using Equation 10 of this section;

(B) the T-statistic for the slope (T_{SURV}) estimated using Equation 11 is equal to or less than the maximum permissible T-statistic (T_{MAX}) in Table 6; and

(C) the largest normalized residual value is equal to or less than the appropriate allowable value from the third column in Table 7 and the second largest normalized residual value is equal to or less than the appropriate allowable value from the second column in Table 7. If any of these criteria is not satisfied, the licensee must propose ΔT_{30} and RT_{MAX-X} values in accordance with paragraph (f)(6)(vi) of this section.

(vi) If any of the criteria described in paragraph (f)(6)(v) of this section are not satisfied, the licensee shall review the data base for that heat in detail, including all parameters used in Equations 5, 6, and 7 of this section 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 $RT_{MAX\text{-}X}$ 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 in accordance with the requirements of paragraphs (c)(1) and (d)(1) of this section.

(7) The licensee shall report any information that significantly influences the $RT_{MAX\text{-}X}$ value to the Director in accordance with the requirements of paragraphs (c)(1) and (d)(1) of this section.

(g) *Equations and variables used in this section.*

Equation 1: $RT_{MAX\text{-}AW} = MAX\ \{[RT_{NDT(U)\text{ - plate}} + \Delta T_{30\text{ - plate}}],$

$[RT_{NDT(U)\text{ - axial weld}} + \Delta T_{30\text{ - axialweld}}]\}$

Equation 2: $RT_{MAX\text{-}PL} = RT_{NDT(U)\text{ - plate}} + \Delta T_{30\text{ - plate}}$

Equation 3: $RT_{MAX\text{-}FO} = RT_{NDT(U)\text{ - forging}} + \Delta T_{30\text{ - forging}}$

Equation 4: $RT_{MAX\text{-}CW} = MAX\ \{[RT_{NDT(U)\text{ - plate}} + \Delta T_{30\text{ - plate}}],$

$[RT_{NDT(U)\text{ - circweld}} + \Delta T_{30\text{ - circweld}}],$

$[RT_{NDT(U)\text{ - forging}} + \Delta T_{30\text{ - forging}}]\}$

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

Equation 6: $MD = A \times (1 - 0.001718 \times T_C) \times (1 + 6.13 \times P \times Mn^{2.471}) \times \varphi t_e^{0.5}$

Equation 7: $CRP = B \times (1 + 3.77 \times Ni^{1.191}) \times f(Cu_e,P) \times g(Cu_e,Ni,\varphi t_e)$

where:

P [wt-%] = phosphorus content

Mn [wt-%] = manganese content

Ni [wt-%] = nickel content

Cu [wt-%] = copper content

A = 1.140×10^{-7} for forgings

= 1.561×10^{-7} for plates

= 1.417×10^{-7} for welds

B = 102.3 for forgings

= 102.5 for plates in non-Combustion Engineering manufactured vessels

= 135.2 for plates in Combustion Engineering vessels

= 155.0 for welds

$\varphi te = \varphi t$ for $\varphi \geq 4.39 \times 10^{10}$ n/cm^2/sec

$= \varphi t \times (4.39 \times 10^{10} / \varphi)^{0.2595}$ for $\varphi < 4.39 \times 10^{10}$ n/cm^2/sec

where:

φ [n/cm^2/sec] = average neutron flux

t [sec] = time that the reactor has been in full power operation

φt [n/cm^2] = $\varphi \times t$

$f(Cu_e,P) = 0$ for $Cu \leq 0.072$

$= [Cu_e - 0.072]^{0.668}$ for $Cu > 0.072$ and $P \leq 0.008$

$= [Cu_e - 0.072 + 1.359 \times (P - 0.008)]^{0.668}$ for $Cu > 0.072$ and $P > 0.008$

and $Cu_e = 0$ for $Cu \leq 0.072$

= MIN (Cu, maximum Cu_e) for $Cu > 0.072$

and maximum Cu_e = 0.243 for Linde 80 welds

= 0.301 for all other materials

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

Equation 8: Residual (r) = measured ΔT_{30} - predicted ΔT_{30} (by Equations 5, 6 and 7)

Equation 9: Mean deviation for a data set of n data points = $(1/n) \times \sum_{i=1}^{n} r_i$

Equation 10: Maximum credible heat-average residual = $2.33\sigma/n^{0.5}$

where:

n = number of surveillance data points (sample size) in the specific data set

σ = standard deviation of the residuals about the model for a relevant material group given in Table 5.

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

where:

m is the slope of a plot of all of the *r* values (estimated using Equation 8) versus the base 10 logarithm of the neutron fluence for each r value. The slope shall be estimated using the method of least squares.

(*se(m)*) is the least squares estimate of the standard-error associated with the estimated slope value *m*.

Equation 12: $r^* = \frac{r}{\sigma}$

where:

r is defined using Equation 8 and σ is given in Table 5

Table 1 - PTS Screening Criteria

Product Form and RT_{MAX-X} Values	RT_{MAX-X} Limits [°F] for Different Vessel Wall Thicknesses[6] (T_{WALL})		
	$T_{WALL} \leq 9.5$in.	9.5in. < $T_{WALL} \leq 10.5$in.	10.5in. < $T_{WALL} \leq 11.5$in.
Axial Weld RT_{MAX-AW}	269	230	222
Plate RT_{MAX-PL}	356	305	293
Forging without underclad cracks RT_{MAX-FO}[7]	356	305	293
Axial Weld and Plate $RT_{MAX-AW} + RT_{MAX-PL}$	538	476	445
Circumferential Weld RT_{MAX-CW}[8]	312	277	269
Forging with underclad cracks RT_{MAX-FO}[9]	246	241	239

[6] Wall thickness is the beltline wall thickness including the clad thickness.

[7] Forgings without underclad cracks apply to forgings for which no underclad cracks have been detected and that were fabricated in accordance with Regulatory Guide 1.43.

[8] RT_{PTS} limits contribute $1x10^{-8}$ per reactor year to the reactor vessel TWCF.

[9] Forgings with underclad cracks apply to forgings that have detected underclad cracking or were not fabricated in accordance with Regulatory Guide 1.43.

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

Table 4 - Conservative estimates for chemical element weight percentages

Materials	P	Mn
Plates	0.014	1.45
Forgings	0.016	1.11
Welds	0.019	1.63

Table 5 - Maximum heat-average residual [°F] for relevant material groups by number of available data points (Significance Level = 1%)

Material Group	σ [°F]	Number of available data points					
		3	4	5	6	7	8
Welds, for Cu > 0.072	26.4	35.5	30.8	27.5	25.1	23.2	21.7
Plates, for Cu > 0.072	21.2	28.5	24.7	22.1	20.2	18.7	17.5
Forgings, for Cu > 0.072	19.6	26.4	22.8	20.4	18.6	17.3	16.1
Weld, Plate or Forging, for Cu ≤ 0.072	18.6	25.0	21.7	19.4	17.7	16.4	15.3

Table 6 – T_{MAX} Values for the Slope Deviation Test (Significance Level = 1%)

Number of available data points (n)	T_{MAX}
3	31.82
4	6.96
5	4.54
6	3.75
7	3.36
8	3.14
9	3.00
10	2.90
11	2.82
12	2.76
14	2.68
15	2.65

Table 7 – Threshold Values for the Outlier Deviation Test (Significance Level = 1%)

Number of available data points (n)	Second largest allowable normalized residual value (r*)	Largest allowable normalized residual value (r*)
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

Dated at Rockville, Maryland, this **xx** day of **xx** 2009.

For the Nuclear Regulatory Commission.

Annette L. Vietti-Cook,
Secretary of the Commission.

Table 7 – Threshold Values for the Outlier Deviation Test (Significance Level = 1%)

Number of available data points (n)	Second largest allowable normalized residual value (r*)	Largest allowable normalized residual value (r*)
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

Dated at Rockville, Maryland, this **xx** day of **xx** 2009.

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Annette L. Vietti-Cook,
Secretary of the Commission.

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