RULEMAKING ISSUE

AFFIRMATION

April 9, 2009 SECY-09-0059

FOR: The Commissioners

FROM: R. W. Borchardt
Executive Director for Operations

SUBJECT: FINAL RULE RELATED TO ALTERNATE FRACTURE TOUGHNESS REQUIREMENTS FOR PROTECTION AGAINST PRESSURIZED THERMAL SHOCK EVENTS (10 CFR 50.61a) (RIN 3150-AI01)

PURPOSE:

To obtain Commission approval to publish a final rule to provide alternate fracture toughness requirements for protection against pressurized thermal shock (PTS) events for pressurized-water reactor (PWR) reactor vessels.

BACKGROUND:

PTS events are system transients in a PWR in which there is a rapid operating temperature cooldown that results in cold reactor vessel temperatures with or without repressurization of the reactor vessel. The rapid cooling of the inside surface of the reactor vessel causes thermal stresses. The thermal stresses combine with stresses caused by high pressure. The aggregate effect of these stresses is an increase in the potential for fracture if a preexisting flaw is present in a material susceptible to brittle failure. The ferritic, low-alloy steel of the reactor vessel beltline, which is 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 Title 10, Section 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," of the *Code of Federal Regulations* (10 CFR 50.61), and adopted on July 23, 1985, (50 FR 29937) establishes screening criteria

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below which the potential for a reactor vessel to fail due to a PTS event is deemed to be acceptably low. The screening criteria effectively define a limiting level of embrittlement beyond which the licensee cannot continue operation without further plant-specific evaluation.

A licensee may not continue to operate reactor vessels with materials predicted to exceed the screening criteria in 10 CFR 50.61 without implementing compensatory actions or additional plant-specific analyses unless it receives an exemption from the requirements of the rule. Acceptable compensatory actions are neutron flux reduction, plant modifications to reduce the PTS event frequency or severity, and reactor vessel annealing. These actions are addressed in 10 CFR 50.61(b)(3), 10 CFR 50.61(b)(4), 10 CFR 50.61(b)(7), and 10 CFR 50.66, "Requirements for Thermal Annealing of the Reactor Pressure Vessel," respectively.

Currently, no operating PWR reactor vessel is projected to exceed the 10 CFR 50.61 screening criteria before the expiration of its 40-year operating license. However, several PWR reactor vessels are approaching the screening criteria, and others are likely to exceed the screening criteria during the extended period of operation of their first license renewal.

The U.S. Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES) developed a technical basis that supports updating the PTS regulations. This technical basis concludes that the risk of through-wall cracking caused by a PTS event is much lower than previously estimated. This finding indicates that the screening criteria in 10 CFR 50.61 are unnecessarily conservative and may impose an unnecessary burden on some licensees. Therefore, the NRC developed a proposed new rule, 10 CFR 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. This proposed new rule is consistent with the staff requirements memorandum "Staff Requirements - SECY-06-0124 - Rulemaking Plan to Amend Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events (10 CFR 50.61)," dated June 30, 2006, in which the Commission asked the staff to prepare a rulemaking that would allow current PWR licensees to implement the new requirements of 10 CFR 50.61a or to continue to comply with the current requirements of 10 CFR 50.61.

The NRC published the proposed rule for public comment in the *Federal Register* on October 3, 2007 (72 FR 56275). The NRC determined that several changes to the October 3, 2007, proposed rule language were desirable to adequately address issues raised in stakeholders' comments. Because these modifications may not have represented a logical outgrowth from the provisions of the October 3, 2007, proposed rule, the NRC requested stakeholder feedback on the modified provisions through the use of a supplemental proposed rule. The supplemental proposed rule specifically solicited stakeholder comments on the provisions related to the applicability of the rule, to the evaluation of reactor vessel surveillance data, and the adjustment of volumetric examination data to demonstrate compliance with the rule. The NRC published the supplemental proposed rule on August 11, 2008 (73 FR 46557). After consideration of the October 3, 2007, proposed rule, the August 11, 2008, supplemental proposed rule, and the stakeholders' comments received on both, the NRC staff developed this final rule.

DISCUSSION:

The NRC received five comment letters containing a total of 54 comments on the October 3, 2007, proposed rule. The NRC received three comment letters containing a total of five comments on the August 11, 2008, supplemental proposed rule. Industry stakeholders submitted all the comments on the proposed rule and on the supplemental proposed rule. These comments are summarized in Enclosure 1, the *Federal Register* notice, and are discussed in detail in Enclosure 2, "Summary and Analysis of Public Comments on Proposed and Supplemental Proposed Rule Language." Comments that resulted in substantive changes to the final rule are discussed below by subject matter.

Applicability of the Rule

Several commenters stated that the rule should apply only to the existing fleet of PWRs. Plants whose construction permits were issued before the effective date of the final rule and whose reactor vessels were designed and fabricated to the 1998 or earlier Edition of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) will have material properties, operating characteristics, PTS event sequences and thermal-hydraulic responses consistent with those of the reactors that were evaluated as part of the technical basis for 10 CFR 50.61a. Other factors, including materials fabrication and welding methods, would also be consistent with the underlying technical basis of 10 CFR 50.61a. The NRC staff agrees with the commenters and determined that allowing the use of 10 CFR 50.61a only to plants whose reactor vessels were designed and fabricated to the 1998 or earlier Edition of the ASME Code would be prudent. As a result of this comment, the NRC staff modified the rule to reflect this position.

Surveillance Data

Several commenters stated that there is little added value in the requirement to assess reactor vessel surveillance data as a part of this rule because the derivation of the embrittlement correlation has already accounted for the variability in the data. The commenters also stated that there is no viable methodology for adjusting the projected shift in transition temperature (i.e., ΔT_{30}) for the reactor vessel based on the surveillance data. Any effort to make this adjustment is likely to introduce additional error into the prediction. Therefore, the commenters believed that obtaining the ΔT_{30} prediction based on the best estimate of chemical composition for the heat of the material is more reliable than a prediction based on a single set of surveillance measurements.

The NRC staff believes that there is added value in the requirement to assess reactor vessel surveillance data. Although the derivation of the embrittlement correlation has already accounted for the variability in the data, the NRC believes that the surveillance data assessment required in the final rule 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. In addition, the staff believes that, although there is no single methodology for adjusting the projected ΔT_{30} for a reactor vessel based on the surveillance data, a licensee could, on a case-specific basis, justify adjustments to the generic ΔT_{30} prediction. For this reason, the rule does not specify a methodology for adjusting the ΔT_{30} value based on reactor vessel surveillance data, but rather it requires the licensee to propose a case-specific ΔT_{30} adjustment procedure for review and approval by the Director of the Office of Nuclear Reactor

Regulation (NRR). Although the commenters assert that error could possibly be introduced, the staff believes that appropriate plant-specific adjustments based on available reactor vessel surveillance data may be necessary to project reactor vessel embrittlement for this rule.

As the result of these comments, the staff continued to work on statistical procedures to identify deviations from generic embrittlement trends. Based on this work, the staff enhanced the procedure described in the rule to, among other things, detect trends from the plant- and heat-specific surveillance data that may emerge at high fluences that the equations described in the proposed rule do not predict. To address this potential deficiency, which could be particularly important during a plant's period of extended operation, the staff added two more statistical tests in the rule. These tests will determine if the embrittlement trend from a particular heat of material show a more rapid increase after significant radiation exposure than the progression predicted by the generic embrittlement trend curve.

Inservice Inspection Volumetric Examination and Flaw Assessments

In the supplemental proposed rule, the NRC staff requested comments on the adjustments of volumetric examination data to demonstrate compliance with the rule and received numerous comments in support of the staff's initiative. The staff decided to permit the adjustment of flaw sizes to account for the effects of uncertainties related to the nondestructive ultrasonic examination (e.g., probability of detection, flaw density, and flaw location) before the estimated size and density distribution are compared to the allowable size and density distribution in the final rule. Licensees are required to base their methodology to account for the nondestructive examination 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 Director of NRR must review and approve the methodology to ensure that the risk associated with PTS is acceptable.

RESOURCES:

The following full-time equivalent (FTE) required to complete this final rulemaking have been allocated in the fiscal year (FY) 2009 budget.

	FY 2009
NRR	0.4 FTE
RES	0.1 FTE
Office of New Reactors (NRO)	0.1 FTE
Office of the General Counsel (OGC)	0.1 FTE
Office of Administration (ADM)	0.1 FTE

No additional resources are necessary to complete this rulemaking.

RECOMMENDATIONS:

The staff recommends that the Commission take the following three actions:

(1) Approve the enclosed final rule (Enclosure 1) for publication in the *Federal Register*.

(2) Certify that this rule, if promulgated, will not have a significant impact on a substantial number of small entities. This certification is included in the enclosed *Federal Register* notice and satisfies the requirement of the Regulatory Flexibility Act (5 U.S.C. 605(b)).

(3) Note the following:

a. The staff has prepared a final regulatory analysis for this rulemaking (Enclosure 3).

b. The final rule contains amended information collection requirements subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501, et seq.) that must be submitted to the Office of Management and Budget (OMB) for its review and approval before the final rule can be published in the *Federal Register* (Enclosure 4).

c. The staff has determined that this action is not a "major rule," as defined in the Congressional Review Act of 1996 (5 U.S.C. 804(2)) and has confirmed this determination with OMB. The staff will inform the appropriate Congressional and U.S. Government Accountability Office contacts.

d. The staff will inform the appropriate Congressional committees.

e. The Office of Public Affairs will issue a press release when the NRC publishes the final rule in the *Federal Register*.

<u>COORDINATION</u>:

The staff briefed the Materials, Metallurgy, and Reactor Fuels Subcommittee of the Advisory Committee on Reactor Safety (ACRS) on the supplemental proposed rule and on the final rule on October 1, 2008 and March 4, 2009, respectively. The staff also briefed the ACRS Full Committee on the final rule on March 5, 2009. The staff received the Committee's recommendation for approval by letter dated March 13, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML090710128). The staff coordinated this paper with NRR, RES, NRO, ADM, the Office of Information Services, and the Office of Enforcement. In accordance with the requirements from the Office of the Chief Financial Officer, the staff did not submit this paper for review by the Chief Financial Officer because the resource needs are less than $100,000 or 1.0 FTE and because the paper has no impact on other planned work. OGC has no legal objection to this paper.

/RA Bruce S. Mallett for/

R. W. Borchardt
Executive Director
for Operations

Enclosures:
1. *Federal Register* Notice
2. Summary and Analysis of Comments
3. Regulatory Analysis
4. OMB Supporting Statement

[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]**.

ADDRESSES: You can access publicly available documents related to this document using the following methods:

Federal e-Rulemaking Portal: Go to http://www.regulations.gov and search for documents filed under Docket ID NRC-2007-0008. Address questions about NRC Dockets to Carol Gallagher at 301-492-3668; e-mail Carol.Gallagher@nrc.gov.

NRC's Public Document Room (PDR): The public may examine publicly available documents at the NRC's PDR, Public File Area O1-F21, One White Flint North, 11555 Rockville Pike, Rockville, Maryland. The PDR reproduction contractor will copy documents for a fee.

NRC's Agencywide Document Access and Management System (ADAMS): Publicly available documents created or received at the NRC are available electronically at the NRC's Electronic Reading Room at http://www.nrc.gov/reading-rm/adams.html. From this page, the public can gain entry into ADAMS, which provides text and image files of NRC's public documents. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC's PDR reference staff at 1-800-397-4209, or (301) 415-4737, or by e-mail to PDR.Resource@nrc.gov.

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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 1×10^{-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 1×10^{-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 $RT_{MAX\text{-}CW}$ limits of Table 1, this weld only contributes 1×10^{-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-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-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 $RT_{MAX\text{-}X}$ 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 $RT_{MAX\text{-}X}$ 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 $RT_{MAX\text{-}X}$ 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 ($\varphi[t]$), 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 $RT_{MAX\text{-}X}$ 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-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-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: $RT_{MAX-AW} = MAX \{[RT_{NDT(U)\text{ - plate}} + \Delta T_{30\text{ - plate}}],$

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

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

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

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

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

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

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

φt_e = φt 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 ≤ 0.072

= $[Cu_e - 0.072]^{0.668}$ for Cu > 0.072 and P ≤ 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 ≤ 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 1×10^{-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.

U.S. Nuclear Regulatory Commission
Alternate Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events (10 CFR 50.61a)

RIN 3150-AI01
[NRC-2007-0008]

Summary and Analysis of Public Comments on Proposed and Supplemental Proposed Rule Language

The Nuclear Regulatory Commission (NRC) is publishing a final rule providing alternate fracture toughness requirements for protection against pressurized thermal shock (PTS) events. This rule provides new PTS requirements based on updated analysis methods, which may be used voluntarily by existing pressurized water reactor (PWR) licensees. This action is desirable because the existing requirements are based on unnecessarily conservative probabilistic fracture mechanics analyses.

The NRC published a proposed rule for public comments in the *Federal Register* on October 3, 2007 (72 FR 56275). Subsequently, a supplemental proposed rule was published in the *Federal Register* on August 11, 2008 (73 FR 46557). The supplemental proposed rule specifically requested stakeholder comments on the provisions related to the applicability of the rule, the evaluation of reactor vessel surveillance data and the adjustments of volumetric examination data to determine compliance with the rule. The NRC considered the comments received on the proposed rule and the supplemental proposed rule in developing the final rule.

The NRC received 5 comment letters for a total of 54 comments on the proposed rule issued on October 3, 2007. The letters were submitted by the PWR Owners Group (PWROG) [Agencywide Documents Access and Management System (ADAMS) Accession No. ML073521547, identified as PWROG], the Electric Power Research Institute (EPRI) [ADAMS Accession No. ML073521545, identified as EPRI], the Nuclear Energy Institute (NEI) [ADAMS Accession No. ML073521543, identified as NEI] and Duke Energy (Duke) [ADAMS Accession No. ML073521542, identified as DUKE]. The NRC also received comments from the Strategic Teaming and Resource Sharing (STARS) endorsing the publication of the Title 10 of the *Code of Federal Regulations* Section 50.61a (10 CFR 50.61a) to the existing regulations [ADAMS Accession No. ML073610558]. The NRC determined that the comment letter from STARS did not propose changes in the proposed rule language.

The NRC received 3 comment letters for a total of 5 comments on the supplemental proposed rule issued on August 11, 2008. The letters were submitted by PWROG [ADAMS Accession No. ML082550705, identified as PWROG2] and EPRI [ADAMS Accession No. ML082550710, identified as EPRI2]. The NRC also received comments from FirstEnergy Nuclear Operating Company (FENOC) supporting the comments submitted by PWROG and EPRI [ADAMS Accession No. ML082600288]. The NRC determined that the comment letter from FENOC did not provide changes in the supplemental proposed rule language other than those already submitted by PWROG and EPRI.

This document places each public comment into one of the following categories:

(1) Embrittlement Trend Curves and Fluence Maps

(2) Surveillence Data

(3) Flaw Limits and Flaw Density Determinations

(4) Adjustments of Volumetric Examination Data

(5) Miscellaneous

Within each category, the NRC has either repeated comments as written by the commenter or summarized the comments for conciseness and clarity. At the end of the comment or comment summary, the NRC references the specific public comments and the letters by which they were provided to the NRC with an identification number. The identification numbers are stated in the form [XXX]-[YY], where:

[XXX] represents the commenter abbreviation provided above (e.g., NEI, PWROG, and PWROG 2), and;

[YY] represents the NRC-assigned sequential comment number.

Note: Where specific comments were grouped together by the commenter but needed to be addressed separately, the NRC added a lower case alpha character to the comment number for uniqueness (e.g., PWROG-32a and PWROG-32b). The identification numbers are also shown in the margin of the annotated copy of the public comments (ADAMS No. ML090260137).

The NRC's responses to the public comments received are discussed below.

COMMENTS RELATED TO THE PROPOSED RULE LANGUAGE

Comments Related to Embrittlement Trend Curves and Fluence Maps

COMMENT: Revise § 50.61a(f) to remove all reference to equations 5, 6, and 7, and require calculation of ΔT_{30} values based on an NRC approved methodology.

Industry bodies should be used to establish a single consensus embrittlement trend curve that is acceptable for use in § 50.61 and other NRC regulations. The consensus embrittlement trend curves should allow evaluation based on reasonably available data and provide accurate predictions of the transition temperature for individual plants. Although the embrittlement trend curves defined in equations 5, 6, and 7, and described in § 50.61a(f) of the proposed rule provides a reasonable description of generic behavior for use in the probabilistic studies, there is no consensus or use of this equation in providing best estimate predictions for transition temperature shifts in individual plants.

Presentations at recent American Society of Testing Materials (ASTM) E10.02 Subcommittee meetings indicate that both industry and the NRC Office of Regulatory Research are currently working on improved trend curves that are expected to eventually become the basis for revisions to ASTM E900 and NRC Regulatory Guide (RG) 1.99. If these revised trend curves are adopted as industry consensus curves, there is a strong possibility that the NRC regulations will include three distinctly different equations for calculating the same parameter (ΔT_{30}). [PWROG-2a, EPRI-2a, NEI-1]

NRC RESPONSE: The NRC does not agree with the comment. The NRC developed a technical basis for the proposed rule, including the embrittlement trend curve, which incorporates the embrittlement correlation that appears in the proposed § 50.61a. The technical basis ensures that all licensees who meet the requirements of the proposed regulation will have an adequately low reactor vessel through-wall crack frequency (TWCF) resulting from PTS events. While it is possible that industry consensus organizations could develop an embrittlement trend curve that the NRC could determine is acceptable to use to demonstrate an adequately low probability of fracture, no industry body has, to date, developed such a curve. No current ASTM standard incorporates embrittlement trend curves that have been correlated to as wide a range of data as that used in the development of the embrittlement trend curves proposed in § 50.61a. Therefore, the NRC has decided to retain the embrittlement trend curves described in the proposed rule which are fit to, and thereby represent, a larger database of U.S. commercial power reactor surveillance data. Furthermore, even if such an embrittlement correlation was developed, NRC would retain its responsibility to assure that use of the trend curve in the technical basis would continue to provide adequate protection from reactor vessel through-wall fracture due to PTS events.

As the state of knowledge regarding embrittlement of reactor pressure vessel steels advances, the NRC will consider the need to modify § 50.61a (and both § 50.61 and RG 1.99, if necessary) to incorporate new, improved embrittlement trend curves when they become available.

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

COMMENT: Section 50.61a will only apply to a handful of plants. This rule should require the use of an NRC approved methodology rather than the specific trend curves. [PWROG-2b, EPRI-2b]

NRC RESPONSE: The NRC agrees that § 50.61a will likely be utilized by only a small number of licensees, impacting a small number of operating PWR plants. However, the NRC does not agree that this will result in any significant confusion for licensees. Inasmuch as the rule is a voluntary rule, any licensee may choose to remain under its (current) licensing basis which includes the existing embrittlement correlation in § 50.61. The NRC reviewed the impact of the existing embrittlement correlation on existing licensees to determine if they would exceed the embrittlement correlation within the current license term, or the term of the first renewed license. The NRC determined that a few plants will exceed the PTS screening criteria in § 50.61 prior to the end of their extended licenses and have identified two additional plants that will require plant-specific action (SECY-07-0104, ADAMS Accession No. ML070570141). Therefore, only a few licensees are likely to utilize the alternate embrittlement correlation in § 50.61a. Thus, the NRC believes that there will be no confusion by licensees with respect to the NRC's decision to offer two alternative approaches for evaluation of the vulnerability of reactor vessels to PTS events.

The NRC emphasizes that each of the PTS rules provides adequate protection to public health and safety. Therefore, utilizing two embrittlement correlations in separate, alternative rules for determining whether reactor vessels have adequate protection against PTS events represents a regulatory policy issue, rather than a safety issue.

No changes were made to the rule as a result of these comments.

COMMENT: Include the limits on application in the proposed rule unless equations 5, 6, and 7 are removed. The original documentation for the embrittlement trend curves had limits of validity for all of the major variables (e.g., fluence, temperature, Cu, Ni). Even with the maximum allowable values for these variables, the shifts and predicted RT_{MAX-X} values will be below the limit of the proposed rule. [PWROG-21, EPRI-21]

NRC RESPONSE: The NRC does not agree that limits of validity were ever established for the embrittlement trend curve. The NRC determines that the rule does not need to include limits on the applicability of the embrittlement trend curves that are described in equations 5, 6, and 7 because the embrittlement trend curves have been correlated to a sufficiently wide range of U.S. commercial PWR surveillance data. The NRC concludes that the embrittlement trend curves are generically applicable to the current operating fleet. The statement of considerations has been revised to provide licensees the calibration range for the embrittlement trend curves. The calibration range is not intended to provide limits on the applicability of the embrittlement trend curves. If current licensees revise their reactor vessel operating conditions, they should use the embrittlement calibration range as guidance for determining whether the RT_{MAX-X} values for the reactor vessel materials need revision.

The NRC disagrees with the commenters' postulate that the use of values based on the limits of the calibrated range for fluence, temperature, copper, nickel, etc. in the calculation of RT_{MAX-X} would result in values of RT_{MAX-X} that are below the screening criteria. The calculations performed by the NRC indicate that the use of values based on the limits of the calibrated range for the input variables can result in a calculated RT_{MAX-X} that exceeds the screening criteria.

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

COMMENT: Reduce the complexity of the rule by removing the calculations (e.g., RT_{MAX-X} values) and putting this detailed information in the form of a RG. RGs are issued to describe and make available such information as methods acceptable to the NRC for implementing specific parts of the NRC regulations. This information would be suitable for this format. Additional changes such as this are recommended to be reviewed by the NRC in the next revision process. Furthermore, citing specific calculation methods makes the use of any other method, even if new improved methods are developed in the future, unacceptable without an exemption or a new rulemaking. [NEI-4]

In addition, there are concerns related to the wording in § 50.61a(f)(2) and equations 1 through 4 of the rule. Specifically:

(1) Focus on fluence at the weld fusion line may add confusion and a degree of difficulty with regard to defining maximum fluence at a location that is not normally singled out. The fusion line is not defined unambiguously for reactor pressure vessel axial or circumferential welds. The text should refer to maximum fluence at the "weld" to avoid confusion. [PWROG-18, EPRI-18]

(2) There are some inconsistencies with the terms φ_{FL} and φt_{FL}. The terms φ_{FL} and φt_{FL} should be changed to φt_{FL}. [PWROG-19, EPRI-19]

(3) In equation 1, explain if the $[RT_{NDT(U)} + \Delta T_{30\text{-plate}}(\varphi t_{FL})]$ should be determined for each of the plates that is adjacent to the axial weld of interest. It seems like the $RT_{MAX\text{-}AW}$ should be the maximum RT_{NDT} for the weld metal and all the plates joined by the weld. Clarify the wording in § 50.61a(f)(2) to state that $RT_{MAX\text{-}AW}$ and $RT_{MAX\text{-}CW}$ is the maximum RT_{NDT} for the weld metal and all the plates joined by the weld. [PWROG-43, EPRI-43]

(4) Clarify the wording in equations 1 through 4 to show ΔT_{30} and fluence are evaluative factors and not algebraic. It appears that the ΔT_{30} shift in these equations is being multiplied by the φt_{FL} (flux x time or fluence term). This cannot be correct because the unit of $RT_{MAX\text{-}X}$ is temperature. The φt_{FL} term should be part of the subscript denoting the ΔT_{30} based on the maximum fluence for the material of interest. [PWROG-44, EPRI-44]

(5) The use of φt_{FL} for welds and φt_{MAX} for other product forms is confusing. Use the term φt_{MAX} and define it as the maximum fluence for either the weld of interest or other material of interest. Fluence can also be defined as φt_{PL}, φt_{AW}, φt_{FO}, or φt_{CW} to clearly indicate which fluence should be used. [PWROG-45, EPRI-45]

NRC RESPONSE: The NRC does not agree that the proposed rule should be reduced in complexity by removing the calculations (e.g., $RT_{MAX\text{-}X}$ values) and putting this detailed information in the form of a RG. The specificity and detail which has been provided by the current PTS rule, described in § 50.61, has been considered a benefit by a majority of NRC stakeholders. Given the significance of the issue of reactor pressure vessel integrity and PTS, providing clear evaluation procedures, including specifying the embrittlement trend curve, and screening criteria within the current § 50.61 has allowed both licensees and the public to apply and understand the rule more clearly. For this reason, the NRC has decided to retain the same general structure within § 50.61a. Since this rule and § 50.61 contain methodology for determining $RT_{MAX\text{-}X}$ and RT_{PTS} values, respectively, and the NRC has decided to maintain consistency between these two rules, the $RT_{MAX\text{-}X}$ calculations will be retained in this rule. An applicant or licensee who wishes to use a different or modified methodology utilized for calculating $RT_{MAX\text{-}X}$ must request an exemption or seek a generic change in the regulation through a rulemaking.

However, the NRC agrees with the proposed changes to §§ 50.61a(f) and (g) because they provide clarification to the rule language. Specifically:

- Section (f) was modified to clarify that the maximum fluence is at the weld not at the weld fusion line

- Sections (f) and (g) were modified to use the term φt_{FL} consistently throughout the rule.

- Section (f) was modified to indicate that $RT_{MAX\text{-}X}$ and $RT_{MAX\text{-}CW}$ is the maximum value of $(RT_{NDT(U)} + \Delta T_{30\text{-plate}})$ for welds and adjoining plates and forgings. The term φt_{FL} was deleted as discussed below.

- Section (g) (i.e., equations 1 through 4) was modified to delete the terms φt_{MAX} and φt_{FL}. The NRC agrees that the ΔT_{30} should not be multiplied by these terms.

- Sections (f) and (g) were modified to clarify the use of φt for plates, forgings, and circumferential and axial welds.

COMMENT: There are concerns regarding the compliance with RG 1.190 and the benefit of using a fluence map. Specifically, explain:

(1) If the method used to determine the flux and/or fluence has to comply with RG 1.190. [PWROG-3, EPRI-3]

(2) If the fluence map has to be generated with a fluence methodology compliant with RG 1.190. [PWROG-13, EPRI-13]

(3) The benefit of recording and submitting a fluence map. A fluence map should only be required if the indications are outside the limits of Tables 2 and 3. [PWROG-15, EPRI-15]

NRC RESPONSE: The neutron fluence and neutron flux used to determine RT_{MAX-X} should be determined using a methodology that complies with the guidance in RG 1.190. Alternative methodologies may be utilized if they are approved by the NRC. The neutron fluence map should also be generated from a neutron fluence methodology that is in compliance with RG 1.190.

The NRC agrees that the neutron fluence information is only necessary for determining the impact of TWCF of flaws that are beyond the limits in Tables 2 and 3 of the rule. As a result of this comment, § 50.61a(e)(6) has been added and § 50.61a(e)(1) of the proposed rule has been modified to require a neutron fluence map only when the flaw assessment results in flaw density or size greater than that specified in Tables 2 and 3. The requirements in §§ 50.61a(c)(2), (d)(2) and (e)(1)(iii) of the proposed rule for submitting a fluence map for the reactor vessel when the examination results meet the requirements of Tables 2 and 3 have been eliminated from the rule.

Comments Related to Surveillance Data

COMMENT: Eliminate the requirement to assess surveillance data, including Table 5, of the proposed rule. 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 currently 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. [PWROG-4, EPRI-4, NEI-2]

NRC RESPONSE: The NRC does not agree with the proposed change. The NRC believes that there is added value in the requirement to assess surveillance data. Although variability has been accounted for in the derivation of the embrittlement correlation, it is the NRC's view that the surveillance 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 from 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 further enhanced the procedure described in § 50.61a(f)(6) to, among other things, detect signs from the plant- and heat-specific surveillance data that may emerge at high fluences of embrittlement trends 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 assessment procedure is described in the document located at ADAMS Accession No. ML081290654. ML081120380, and ML081120600. The technical basis for the enhanced surveillance assessment procedure is described in the document located at ADAMS Accession No. ML081290654.

COMMENT: Eliminate § 50.61a(f)(6) and Table 5 of the proposed rule. The requirements regarding the evaluation of surveillance or other data relative to the embrittlement trend curve predictions of the ΔT_{30} shift with irradiation should only apply to new data that was not already included in the development of the embrittlement trend curve used in § 50.61a(g) of the proposed rule. The proposed statistical evaluation described in equations 8 through 10 are not consistent with how the standard deviations in Table 5 of the proposed rule were calculated. [PWROG-37, EPRI-37]

NRC RESPONSE: The NRC does not agree with the proposed change. Specifically, the standard deviation values in Table 5 of the proposed rule are based on a large surveillance database of nearly 1000 ΔT_{30} shift values. In contrast, the surveillance data available for each heat of material is a very small portion (i.e., no more than 8 shift values) of the larger database. While the standard deviation values in Table 5 should, in principle, be calculated by first excluding surveillance data being evaluated, this change would not alter the standard deviation values significantly because the surveillance database is very large relative to the size of an individual heat-specific dataset. It is also for this reason that the NRC concludes that the provisions of § 50.61a(f)(6) can be applied to existing data with limited error. As discussed in the NRC response other public comments, the provisions of § 50.61a(f)(6) have been extensively revised, and in so doing other concerns regarding the statistical basis for the standard deviations in Table 5 of the rule have been addressed.

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

Comments Related to Flaw Limits and Flaw Density Determinations

COMMENT: Conduct a technical meeting with industry to discuss concerns related to flaw limits. [PWROG-32a, EPRI-32a, NEI-3]

NRC RESPONSE: The NRC noted that the reason for holding a meeting was unclear. Therefore, on January 22, 2008, the NRC held a telephone conference with the commenters to clarify what would be the purpose of the proposed meeting. The commenters clarified that a detailed list with concerns related to this comment was provided in writing to the NRC during the comment period. The commenters stated that the intent of the comment was to inform the NRC that, if needed, the industry will be available to provide clarifications regarding this detailed list during a meeting or a telephone conference.

The NRC informed the commenters that this detailed list was currently under evaluation. The NRC noted that a separate meeting was not needed as the list with comments was clear and self explanatory. The NRC documented this mutual understanding by issuing a summary of telephone conference dated February 14, 2008 (ADAMS Accession No. ML080440173).

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

COMMENT: Provide tables 2 and 3 as guidance, but not a strict requirement. Sections 50.61a(d)(2) and (e)(4) imply that failure to meet the flaw distribution requirements in Tables 2 and 3 would require a probabilistic analysis within one year to allow continued operation. This means that observation of a single large flaw could trigger a major analysis program. The technical basis for these tables is not obvious and the implications could be onerous. [PWROG-10, EPRI-10]

NRC RESPONSE: The NRC does not agree with the comment. The NRC considers the check of plant-specific flaw distributions to be consistent with the NRC's general treatment of risk-informed decision-making and an essential verification of one of the major technical basis input parameters. Given the significance of the flaw distributions used in the technical basis, the NRC concludes that the check of inspection results against the information in Tables 2 and 3 must be a requirement of the rule. Therefore, the NRC disagrees with the commenters' proposal to treat Tables 2 and 3 as guidance.

Although the NRC agrees with the commenters' statement that, "observation of a single large flaw *could* [emphasis added] trigger a major analysis program," it would not require a probabilistic analysis to be completed within one year to allow continued operation. The NRC contends that the rule provides flexibility in § 50.61a(e)(4) for a licensee, as addressed in the statement of considerations for the proposed rule, to justify the use of the voluntary rule without necessarily initiating a "major analysis program." However, in some cases, a thorough, plant-specific analysis may be warranted depending upon the flaw distribution observed in the plant-specific inspection results.

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

<u>COMMENT</u>: Revise the proposed rule to require that only axially-oriented flaws be evaluated per Tables 2 and 3. Section 50.61a(e)(1)(iii) requires the licensee to document the orientation of the indication relative to the axial direction. However, there is no provision for the use of this information relative to Tables 2 and 3 of the rule. [PWROG-14, EPRI-14]

<u>NRC RESPONSE</u>: The NRC agrees with the comment that the proposed rule does not require consideration of the orientation of the flaw relative to the axial direction. However, the NRC does not agree with the proposed change to the rule requirements. The evaluation of plant-specific inspection data against the values provided in Tables 2 and 3 is to be performed by counting all flaws independent of orientation. The construction and use of the tables within the structure of the proposed rule is based on this understanding. Consistent with the assumptions used in the probabilistic fracture mechanics analysis as part of the basis for the proposed rule, the NRC would expect that all, or the majority of, flaws found during the inspection of axial welds will be axially-oriented, while those observed during the inspection of circumferential welds will be circumferentially-oriented.

Although the documentation requirements of § 50.61a(e)(1)(iii) are not relevant to the evaluation of inspection data against Tables 2 and 3, this documentation is relevant to having appropriate information available if further a more detailed evaluation of a specific vessel is required. The NRC has modified § 50.61a(e)(1)(iii) to clarify the reporting requirements of the rule.

For example, licensees may document planar flaws, as defined in American Society of Mechanical Engineers (ASME) Code, Section XI, Subsection IWA-9000, as “axial” those for which the major axis of the flaw is identified by an ultrasonic transducer oriented in the circumferential direction. All other planar flaws may be categorized as “circumferential.” The NRC has also modified the statement of considerations for the rule to clarify the level of detail expected of this documentation.

<u>COMMENT</u>: Section 50.61a(e)(2) of the proposed rule states that licensees shall verify that if indications are detected at the clad-to-base metal interface and that the licensee shall verify that such indications do not open to the vessel inside surface using a qualified surface or visual examination. A number of forging plants have been identified (as noted in NUREG-1874) as having relatively large areas of underclad cracking. These areas have been inspected repeatedly and have shown no evidence of growth. Furthermore, evaluations have been performed, and approved by the NRC staff (e.g., WCAP-15338-A), that have shown that the growth of these underclad cracks is not likely.

The commenters requested that the NRC explain:

(1) If the intention of the proposed rule is that these plants would be required to perform the proposed surface or visual examinations over these areas during each inservice inspection.

(2) If indications are detected at the clad-to-base metal interface and surface or visual examinations confirm that these indications are not connected to the vessel inside surface, is it

necessary to repeat the surface or visual examinations after subsequent volumetric examinations when the same indications are detected at the clad-to-base metal interface.

The commenters suggested that:

(1) Flaws at the clad-to-base metal interface that have been identified in previous inspections should be exempt from the surface or visual examinations of the proposed rule.

(2) The proposed rule should replace the term “indications” with the term “flaws.” [PWROG-16, EPRI-16]

NRC RESPONSE: It is the intention of the NRC to require the proposed surface or visual examination in § 50.61a(e)(2) of the proposed rule to be performed during each inservice inspection. If indications are detected at the clad-to-base metal interface and surface or visual examination confirms that these indications do not connect to the inside surface of the vessel, the surface or visual examination is to be performed after subsequent volumetric examination when the same indications are detected at the clad-to-base metal interface.

The NRC believes it is an appropriate defense-in-depth measure to require that the performance of a surface or visual examination during each inservice inspection to determine whether flaws at the clad-to-base metal interface have grown through the clad. This inspection must be repeated at each inspection to determine whether environmental factors have caused a flaw to grow through the clad and either penetrate into the steel or link up to a flaw at the clad-to-base metal interface.

Surface connected flaws that are axially-oriented would be a significant contributor to the probability of vessel failure caused by postulated PTS events. Since the volumetric examination is not capable of determining whether an indication at the clad-to-base metal interface is connected to the surface, a surface or visual examination is required to ensure that cracks in the clad have not initiated and grown through the clad and connected with flaws at the clad-to-base metal interface. Given the significant effect that an axially-oriented surface breaking flaw would have on the structural integrity of the reactor pressure vessel, and given the inability of volumetric examination required by ASME Code, Section XI to detect surface flaws, the NRC concludes that flaws in the clad-to-base metal interface that have been identified in previous inspections should not be exempt from the surface or visual examination.

ASME Code, Section XI, Article IWA-9000 defines the terms “indication” and “flaw.” An indication is a response or evidence from the application of a nondestructive examination. The NRC interprets this to mean that an indication in an ultrasonic examination is the signal response during the examination. A flaw is an imperfection or unintentional discontinuity that is detectable by nondestructive examination. Therefore, the NRC will replace the term “indications” with the term “flaws,” where appropriate.

COMMENT: State in the statement of considerations that surface breaking flaws were considered in the proposed rule. The background section of the statement of consideration states that surface breaking flaws that penetrate through the cladding were not included in the technical basis. This is not true because the possibility of having flaws of this type were in fact considered in the pilot plant (i.e., Oconee Unit 1) for the B&W plant designs as described in NUREG-1806 and NUREG-1874. They were included because their existence cannot be excluded in single pass cladding. [PWROG-27, EPRI-27]

NRC RESPONSE: The NRC agrees with the comment. The surface cracks the commenters are referring to were indeed evaluated in the technical basis, but those surface cracks only penetrated 0.070 inch into the welds or adjacent base metal. The NRC modified the statement of considerations to clarify that the surface cracks that penetrate through the stainless steel clad and penetrate more than 0.070 inch into the welds or the adjacent base metal were not included in the technical basis of this rule. In addition, §§ 50.61a(e)(2) and (e)(3)(ii) have been modified to require licensees to evaluate whether flaws are equal to or greater than 0.075 inches in through-wall extent from the clad-to-base metal interface.

COMMENT: Remove §§ 50.61a(e)(2) and (e)(4)(iii) of the proposed rule because they do not provide valuable information. Surface breaking flaws that penetrate through the clad were included in the technical basis. It has been shown that even if surface breaking flaws were to occur in single pass clad and grow by fatigue, they would not contribute to TWCF because of their circumferential orientation. [PWROG-34, EPRI-34]

NRC RESPONSE: The NRC agrees that surface breaking flaws that penetrate through the clad were included in the technical basis and that circumferential surface breaking flaws do not contribute to TWCF. Since axially-oriented flaws are significant and circumferentially-oriented flaws are not significant, §§ 50.61a(e)(2) and (e)(4)(iii) of the proposed rule have been revised to indicate that licensees shall identify and evaluate flaws that are axially-oriented and located at the clad-to-base metal interface.

Although the technical basis included surface breaking flaws that penetrate through the clad, the technical basis did not model the impact of through-clad flaws linking with flaws at the clad-to-base metal interface. The intent of §§ 50.61a(e)(2) and (e)(4)(iii) in the proposed rule [Note: § 50.61a(e)(4)(iii) in the proposed rule has been renumbered to § 50.61a(e)(4)(ii)] is to ensure that flaws, or combination of flaws, which exceed the assumptions in the technical basis do not exist in reactor pressure vessels to which 10 CFR 50.61a is applied. The NRC does not agree that §§ 50.61a(e)(2) and (e)(4)(iii) of the proposed rule should be removed because these paragraphs provide information on whether through-clad flaws have linked with flaws at the clad-to-base metal interface.

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

COMMENT: The embedded flaw limits for one vessel inservice inspection volume in Tables 2 and 3 correspond to an upper 3-sigma bound on the 1000 distribution input to the Fracture Analysis of Vessels (FAVOR) code. The mean limits for the 69 vessels in the U.S. PWR plants are consistent with the average values reported in the FAVOR output for thousands of simulated vessels. Therefore, if the accumulated number of vessel inservice inspection volume indications start to become significantly different than the limits would indicate; an evaluation of effects of these differences could be performed by the NRC. [PWROG-31a, EPRI-31a]

NRC RESPONSE: The NRC agrees that if the accumulated number of vessel inservice inspection volume flaws becomes significantly different than the limits would indicate, an evaluation of effects of these differences should be performed by the NRC.

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

COMMENT: The technical basis for Tables 2 and 3 failed to account for (a) the effects of the uncertainties included in the 1000 embedded flaw distribution inputs to the FAVOR code, (b) the flaws that contributed to TWCF, and (c) the flaws that could be detected during inspection of the beltline region. [PWROG-30a, EPRI-30a, PWROG-28a, EPRI-28a]

NRC RESPONSE: The NRC disagrees that Tables 2 and 3 fail to account for the three issues identified in the comment. Tables 2 and 3 were created to address concerns that the flaws in a plant-specific vessel might not be bounded consistent with the inputs and assumptions used in the FAVOR code calculations and that, in turn, might cause the TWCF to exceeded $1x10^{-6}$ per year if the RT_{MAX-X} limits in Table 1 are reached. The number and sizes of flaws in Tables 2 and 3 are to be compared to the number and sizes of flaws from the ASME Code volumetric examination of the reactor vessel welds.

Effects of Uncertainty. The comment regarding uncertainty is correct insofar as the values in Tables 2 and 3 are based on mean values and not a percentile from a distribution or developed using some other mathematical process. However, uncertainty is addressed in the tables, and in the use of the tables, by including several conservative assumptions. The mean number of flaws input into the FAVOR code and upon which the tables are based come primarily from data on the Shoreham vessel. The Shoreham vessel had about three times as many flaws as the only other available data from the PVRUF vessel and therefore already represents an upper bound of the available flaw distribution data. During the simulations, the FAVOR code assigned all flaw depths to be at the upper end of each bin instead of distributing the sizes throughout the bin as would be more realistic. Consistent with this assumption, the screening criteria would be satisfied if all observed flaws within each bin were at the maximum allowable size even though the actual observed flaw sizes will be distributed throughout the bin. Therefore, the NRC concludes that mean values upon which Tables 2 and 3 are based appropriately address the effects of uncertainty and does not believe that including additional quantitative uncertainty in their derivation is necessary.

Flaws that Contribute to TWCF. The commenters are incorrect that Tables 2 and 3 fail to account for the flaws that contribute to the TWCF. The tables include the flaws which are within one-eighth of the vessel thickness from the clad-to-base metal interface because these flaws are responsible for nearly all of the TWCF from PTS events. However, the limits in Tables 2 and 3 are not based on only those flaws that contribute to vessel failure because the intent of the tables is to ensure that the flaw population in the vessel being assessed is consistent with the flaw population assumed in the technical basis calculations that were performed to support this rule.

Flaws Detected During Inservice Inspections. The NRC does not agree that the technical basis document did not consider the ability to detect flaws in the inservice inspection volume. Current volumetric examination technology can adequately detect flaws greater than 0.075 inches in through-wall extent. Since the rule imposes no limit on the number of flaws less than 0.075 inches in through-wall extent it is not necessary to detect flaws smaller than this.

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

COMMENT: Consider options for alternative methods to develop Tables 2 and 3. For example, include an appropriate revision of the input flaw distributions per NUREG/CR-6817 and a sensitivity study with the latest version of the FAVOR code for their effect on TWCF and the PTS screening limits of Table 1. [PWROG-31b, EPRI-31b]

NRC RESPONSE: The NRC considered alternative methods for developing Tables 2 and 3, but has concluded that the method described in NUREG/CR-6817 and ADAMS Accession No. ML070950392 provides reasonable assurance that any reactor vessel with a plant-specific flaw distribution that meets the table limits will have a TWCF less than or equal to $1x10^{-6}$ per year.

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

COMMENT: An individual utility should not be required to perform the evaluation of the effect of potentially new flaw distributions for the PWR fleet. [PWROG-28b, EPRI-28b]

NRC RESPONSE: The NRC agrees that individual licensees should not be required to perform evaluations of the effects of potentially new flaw distributions for the PWR fleet. The rule does not require licensees to perform evaluations of the effects of potentially new flaw distributions for the PWR fleet. The rule only requires that licensee perform plant-specific assessments when flaws exceed the limits in §§ 50.61a(e)(1) and (e)(3), or if the flaws open to the inside surface.

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

COMMENT: Plant-specific concerns should be addressed by considering options for alternative methods that do not require approval of the Office of Nuclear Reactor Regulation (NRR). [PWROG-28c, EPRI-28c]

NRC RESPONSE: The NRC does not agree with the comment. The protection of a PWR's reactor pressure vessel from failure during a PTS scenario is an issue of significant regulatory concern. The failure of a reactor pressure vessel during a PTS event could lead to a beyond design basis failure of the facility's reactor coolant system and endanger the health and safety of the public. As such, the NRC has a responsibility to ensure that PWR licensees have demonstrated, either through compliance with the screening criteria in § 50.61 or § 50.61a, or through alternative means, that their reactor pressure vessels will be adequately protected against failure during a PTS event in accordance with NRC regulations (including 10 CFR Part 50, Appendix A, General Design Criteria 14, 30, and 31). It is the NRC's position that, in particular when alternative methods are being used to demonstrate adequate protection, NRC staff review and approval is a necessary requirement. The new PTS rule is a major modification to the previous rule and required further development and refinement of several complex analyses. Plant-specific application of these analyses requires a substantial amount of engineering judgment and consequently the NRC has concluded that NRC review and approval of the plant-specific analyses is necessary. Furthermore, NRC review and approval of alternative methods to demonstrate that the reactor vessel will remain adequately protected against PTS events is consistent with the requirements in § 50.61(b)(5). This paragraph states that the Director of NRR should approve operation of a facility with RT_{PTS} in excess of the PTS screening criteria.

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

COMMENT: Licensees should not be required to document the flaws that exceed the limits in Tables 2 and 3 because this information is already available (i.e., no new paperwork requirement). [PWROG-30b, EPRI-30b]

NRC RESPONSE: The NRC does not agree with the comment; the NRC needs to know the size and location of all flaws that exceed the limits in Tables 2 and 3 so that the NRC can evaluate the licensee's assessment of the impact of these flaws on the plant-specific TWCF. Since the rule requires the licensee to provide flaw assessments on vessels that exceed the limits in Tables 2 and 3, the additional requirement to identify flaw size and location is a minimal additional burden. In addition, this information is necessary so that the NRC could generically re-examine the technical basis for the rule.

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

COMMENT: Modify Table 2 to start at 0.075 inches and to delete the reference to "ASME flaw size increment". [PWROG-12, EPRI-12, PWROG-42, EPRI-42]

The minimum flaw size is inconsistent with ASME Code inspection requirements and; therefore, cannot be practically implemented. Also, the term "ASME flaw size increment" is not a term defined in ASME Code, Section XI and the smallest flaw depth qualified by the ASME Code, Section XI, Appendix VIII, is 0.075 inches. Therefore, determining flaw densities with recorded flaws as small as 0.05 inch through-wall extent, as implied in Tables 2 and 3, may require smaller flaw sizes to be reported using a procedure that is not qualified to such a shallow depth. [PWROG-32b, EPRI-32b, PWROG-12, EPRI-12, PWROG-42, EPRI-42]

NRC RESPONSE: The NRC agrees that the minimum flaw size in the first row in Table 2 (and Table 3) should be changed, but has changed the minimum size to zero and not 0.075 inches as suggested by the commenters. The tables start with zero as the smallest flaw size, and extend to infinity as the largest size, to prevent confusion that might arise if some flaw sizes were not included in the tables. There is no limit on the number of flaws in the first bin and; therefore, it is not necessary to identify or size flaws smaller than the minimum size in the second row. The smallest flaws that must be sized in the second bin are 0.075 inches deep. Therefore, the sizing requirements in the tables are consistent with the smallest flaw depth according to ASME Code, Section XI, Appendix VIII. The rule includes several requirements to document and evaluate flaws. Changes were made throughout the rule to clarify that those flaws less than 0.075 inches need not be documented or evaluated.

The NRC also agrees that the ASME Code does not have the term "ASME flaw size increment"; therefore, the term was removed from the rule language; to be consistent with the ASME Code.

COMMENT: The flaw size increments in the proposed tables are inconsistent with those used in the representative plant analyses in NUREG-1874. Specifically:

(1) The embedded flaw size increment in Tables 2 and 3 is less than one percent of the vessel wall thickness. However, an increment of one percent was used to generate the 1000 weld and plate flaw distributions that are input into the FAVOR code as described in NUREG/CR-6817, Revision 1, "A Generalized Procedure for Generating Flaw-Related Inputs for the FAVOR Code," Sections 9.4 and 9.5.

(2) For the probabilistic fracture mechanics calculations, the FAVOR code uses only the largest flaw size for the range of sizes in each increment of one percent of the vessel wall thickness. [PWROG-32c, EPRI-32c]

NRC RESPONSE: The NRC agrees that the flaw size increments of Tables 2 and 3 are different than those used for the FAVOR code calculations but disagrees that they are inconsistent. The flaw size increments used in NUREG-1874 and NUREG/CR-6817 were developed to support the probabilistic fracture mechanics calculations using the FAVOR code. The selected flaw size increments in Tables 2 and 3 are developed from these FAVOR code inputs, but were modified to account the characteristics of the ASME Code inservice inspection methods and requirements. The technical basis for the development of the tables (ADAMS Accession No. ML070950392) describes how the FAVOR code input was transformed into the tables' entries. It should be noted that, to some extent, both the FAVOR code and the ASME Code flaw size increments are selected as a matter of convenience. They both provide a discrete representation of the continuous distribution of flaw sizes that appear in reactor pressure vessel plates and welds.

Lastly, the commenters are correct that during a calculation the FAVOR code uses the largest size in a bin as the size of all flaws in that bin. The basis for the development of Tables 2 and 3 also used the largest size in a bin as the size of all flaws in that bin. Therefore, the basis for the FAVOR code and Tables 2 and 3 are consistent.

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

COMMENT: Base the flaw limits only those embedded flaws that contribute to vessel failure. The limits on embedded flaws in Tables 2 and 3 are based upon the flaws simulated by the FAVOR code, not just those flaws that that could fail due to PTS. The following simulated flaws have minimal contribution to failure and TWCF: embedded flaws up to one foot above and below the beltline region adjacent to the reactor core, flaws with a through-wall extent from 12.5 to 37.5 percent of the vessel wall thickness and all embedded flaws that are oriented in a circumferential direction. [PWROG-32d, EPRI-32d]

NRC RESPONSE: The NRC agrees with the commenters' understanding of the basis of the flaw limits of Tables 2 and 3. However, the NRC does not agree that it is practical to base the limits in Tables 2 and 3 on *only* those flaws that contribute to vessel failure. The flaws that contribute to vessel failure are a function of the size of the flaw, its location (i.e., the embrittlement level of the material in which it is located), and the transient to which it is subjected. It would be impractical to attempt to define, *a priori*, the flaws which contribute to vessel failure for any given vessel. Rather, the intent of the tables is to ensure that the flaw population, as a whole, in the vessel being assessed is consistent with the flaw population assumed in the technical basis calculations that were performed to support this rule. It is necessary for a licensee to demonstrate that the flaw distribution in their reactor vessel is bounded by the flaw distributions used in the FAVOR code in order to demonstrate an adequately low TWCF.

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

COMMENT: The flaw limits are applicable to a large number of vessels, not a single vessel, since they are based on average values of the thousands of simulations used in the representative plant probabilistic analyses. The allowable number of flaws in Tables 2 and 3 are based upon the average number of flaws in a given size range for thousands of vessel simulations by the FAVOR code without any consideration of the variability among the 1000 flaw distributions input to the FAVOR code for both welds and plates. It is expected that the number of embedded flaws in 50 percent of the vessels would be greater than this average value. [PWROG-32e, EPRI-32e]

NRC RESPONSE: The NRC does not agree with the comment. The use of the 1000 flaw distribution inputs to the FAVOR code was only one of several measures to account for uncertainties in the estimates of the number and sizes of flaws. The objective was to specify inputs to the FAVOR code that ensure the use of an overall conservative representation of vessel flaws that can be regarded as bounding for most vessels in the PWR fleet. Based on the methodology for generating the flaw distribution input described in NUREG/CR-6817, the NRC expects that the flaw distributions represented by Tables 2 and 3 should bound the actual flaw distributions in the majority of the operating PWR fleet. Based upon the structure of this entire rule, the NRC has concluded that the implementation of the plant-specific flaw distribution check, as defined by Tables 2 and 3, will ensure that adequate protection is maintained for all plants implementing § 50.61a.

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

COMMENT: The maximum flaw size limits are unrealistic because they do not represent the range of values used in the representative plant analyses. The maximum embedded flaw size for welds in Tables 2 and 3 are set so that on average only one flaw would be expected to occur in each vessel simulated by the FAVOR code. It appears there is no consideration of the maximum embedded flaw size in the 1000 distributions input to the FAVOR code, which are based upon the truncation limits in NUREG/CR-6817, Revision 1. [PWROG-32f, EPRI-32f]

NRC RESPONSE: The NRC does not agree with the comment. The maximum flaw size limits of Tables 2 and 3 will require the specific assessment of flaws that were simulated in the FAVOR code. Such flaws are large and were simulated to occur, on average, at a frequency of less than one per vessel. A case-specific evaluation is necessary because a single large flaw could, by itself, cause a vessel's TWCF to exceed the acceptance criteria. Further, the NRC does not consider the case-specific assessment of such large and (relatively) rare flaws to be overly burdensome. Since most flaws are expected to exist in low embrittlement regions; it will be relatively easy to demonstrate their limited impact on TWCF.

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

COMMENT: The plate embedded flaw limits are unrealistic as they are primarily based upon failures in simulated axial weld flaws. [PWROG-32g, EPRI-32g]

NRC RESPONSE: The NRC does not agree with the statement that the plate embedded flaw limits are unrealistic because the limits are primarily based upon failures in simulated axial weld flaws. The plate embedded flaw limits are based on the representation of plate flaws that are not associated with welds used in the FAVOR code.

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

COMMENT: It appears that the embedded flaw limits for plates in Table 3 are based upon FAVOR output for plate failures, not plate flaws. FAVOR results used for NUREG-1874 show that the majority of plate failures are due to simulated axial weld flaws for Beaver Valley Unit 1. [PWROG-32h, EPRI-32h]

NRC RESPONSE: The NRC does not agree with the comment. The limits of Table 3 for embedded flaws in plates and forgings address only flaws that are fully embedded in the base metal. Flaws along weld fusion lines that have the potential to propagate into adjacent low toughness plates or forgings are not included in the Table 3 limits, but rather, in the Table 2 limits.

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

COMMENT: Clarify if the limits in Table 3 apply to all of the plate material or just the beltline material inspected with the welds per the requirements in ASME Code, Section XI. [PWROG-32i, EPRI-32i]

NRC RESPONSE: The limits in Table 3 apply to the beltline material inspected. It does not apply to all the plate material. ASME Code, Section XI examinations inspect only a small fraction of the total volume of base metal of a vessel beltline. The inspected material is adjacent to the beltline welds. Applications of Table 3 should account for the surface area of inspected plate and forging material.

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

COMMENT: The plate limits should have restrictions regarding their application to forgings susceptible to underclad cracking. [PWROG-32j, EPRI-32j]

NRC RESPONSE: The NRC agrees that the plate limits should have restrictions regarding their application to forgings susceptible to underclad cracking. These restrictions were included in footnote 8 of the proposed rule. Footnote 8 has been deleted from the rule to accommodate editorial and regulatory changes made as a result of other comments from the public. However, these restrictions are now included in footnote 1 and in the heading rows of Tables 2 and 3.

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

COMMENT: There is no guidance on whether the plate embedded flaw limits in Table 3 can be applied for forgings. It appears that the limits of Table 3 can be applied to forgings if they are not susceptible to underclad cracking or the susceptible forging material is below the appropriate PTS screening limit in Table 1 of the rule. [PWROG-32k, EPRI-32k]

NRC RESPONSE: The NRC disagrees with the commenters. The rule states that Table 3 applies equally well to the allowable number of flaws in either plates or forgings. In addition, Table 3 states that the values for allowable numbers of flaws do not include underclad cracks in forgings. Therefore, underclad cracks detected by an ASME Code, Section XI, examination should not be counted for purposes of comparison with the allowable flaw limits of Table 3.

The rule can be applied to vessels with forging materials if underclad cracks are present. However, Table 1 imposes more restrictive PTS screening criteria to forgings with underclad cracks. The absence of underclad cracks can be justified on the basis of (1) inservice inspection results, or (2) by considerations of forging material chemistries and the welding procedure used to apply the cladding to the vessel surface. Underclad cracks are also addressed in paragraph (e)(2) of the rule. Application of the alternative PTS rule, § 50.61(a), requires inspections to verify that flaws at the clad-to-base metal interface do not extend through the clad and thereby open to the inside surface of the vessel.

The NRC believes that guidance on whether the plate embedded flaw limits in Table 3 can be applied for forgings has already been provided. However, a footnote in § 50.61a(e)(1) has been added to further clarify that 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.

<u>COMMENT</u>: Provide an acceptable evaluation method to evaluate the effect of exceeding the embedded flaw limits of Tables 2 and 3. Neither of the options suggested in the statement of considerations of the proposed rule can be practically implemented. If necessary, the TWCF needs to be evaluated to determine if it exceeds the limit of 1×10^{-6} per year and submitted to the Director of NRR for review and approval. It appears that a simple evaluation procedure could be developed based upon the fact that probability of vessel failure (i.e., through-wall crack) during a postulated PTS transient depends on the number of embedded axial flaws in the vessel. The adjusted TWCF contribution of the axial welds and/or plates could then be calculated using the correlations with the RT_{MAX-X} per NUREG-1874, equations 3-5 and 3-6, and evaluated relative to the risk limit of 1×10^{-6} per reactor year without the approval of the Director of NRR being required. [PWROG-32l, EPRI-32l, PWROG-33, EPRI-33]

<u>NRC RESPONSE</u>: The NRC does not agree that a pre-approved (i.e., "acceptable") method to evaluate flaws that are found in the reactor pressure vessel, and are beyond the limits of Tables 2 and 3 of § 50.61a, is needed. Furthermore the NRC does not agree that the evaluation procedure proposed by the commenters would be acceptable in all cases. In the rule and in the Section by Section Analysis of the statement of considerations, the NRC discusses the requirements of § 50.61a(e)(4) and identifies two options that may be pursued by licensees if flaws are found in the reactor pressure vessel that exceed the limits in §§ 50.61a(e)(1) and (e)(3), or if the flaws are found that are open to the inside surface of the reactor vessel. The analysis could be a complete, plant-specific, probabilistic fracture mechanics analysis or a simplified analysis of flaw size, location, and embrittlement to demonstrate that the actual flaws in the reactor vessel are not in locations that would cause the TWCF to be greater than 1×10^{-6} per reactor year. Paragraph (e)(2) requires that if analyses performed under § 50.61a(e)(4) are used to justify continued operation of the facility, approval by the Director of NRR is required prior to implementation.

The NRC believes that either of the options discussed in the preceding paragraph could be implemented if flaws are found in the reactor pressure vessel that exceeds the limits in Tables 2 and 3. It is the NRC's view that the best option may depend on the specifics of the situation being assessed, making it impractical to provide detailed and specific guidance as part of § 50.61a.

As a result of this comment the NRC has clarified the discussion of § 50.61a(e)(4) in the statement of considerations in this *Federal Register* notice to provide additional information regarding analysis options which may meet the intent of § 50.61a(e)(4).

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

<u>COMMENT</u>: Remove the sections from the PTS rule that require reporting the embedded flaws that violate the ASME Code requirements since they provide no additional information of any value per the Paperwork Reduction Requirements. The embedded flaws that violate the size requirements of ASME Code, Section XI, Table 3510-1 are reportable and evaluated per the requirements of ASME Code, Section XI, IWB-3610. This information is already contained in the vessel inspection summary reports that are being sent to NRC. For PTS concerns, the limits on the number of embedded flaws by size in Tables 2 and 3 are controlling. [PWROG-35, EPRI-35]

NRC RESPONSE: The NRC agrees with the comment that embedded flaws violating the size requirements of ASME Code, Section XI, Table 3510-1 are reportable and evaluated per the requirements of ASME Code, Section XI, IWB-3610 and that this information is already contained in the vessel inspection summary reports that are being sent to NRC. The NRC also agrees that, for PTS concerns, the limits on number of embedded flaws by size in Tables 2 and 3 are controlling.

However, the NRC does not agree with the commenters' suggestion to remove the reporting requirements specified in § 50.61a(e)(3) of the proposed rule. The NRC understands that some of the information required to be submitted by the rule may be provided in some, but not all, inservice inspection summary reports to the NRC. For example, the inservice inspection summary report does not necessarily include information about flaw sizes and locations when the flaw sizes are less than the reportable sizes. In order to meet the requirements of the rule, all of the flaw information must be provided to the NRC in one package. The commenters' proposal suggests that the information package provided to the NRC would provide a flaw-by-flaw reference to the inservice inspection summary report for those flaws that are described in that report. The NRC has determined that it would require the same level of effort to provide the actual description of each flaw as it would take to provide the reference information for each flaw. The NRC believes that eliminating the time needed for the NRC to search through different summary reports will increase the efficiency of the NRC evaluation process.

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

COMMENT: A footnote to the Supplement 4 inspection volume defines the volume as the weld volume and not the normal examination volume, which is the weld plus ½ of the vessel wall thickness. This paragraph requires the inner 1 inch/10 percent from the clad interface to be examined or analyzed. This conflicts with § 50.61a(e)(1)(ii) of the proposed rule which implies the plates and forgings are inspected. [PWROG-40a, EPRI-40a]

NRC RESPONSE: The NRC agrees that the intended examination volume includes the weld plus ½ of the vessel wall thickness from the outer edges of the weld and from the clad-to-base metal interface to three-eights of the reactor vessel thickness from the interior surface. As a result of this comment, § 50.61a(e) of the rule has been modified to clarify the volume for the inspection required by §§ 50.61a(e)(1) and (e)(2). This inspection volume includes plates and forging material as implied by § 50.61a(e)(1). Therefore, some portions of plate and forging material are required to be inspected.

COMMENT: Clarify if the examination volume is the inner 1 inch/10 percent, or inner three-eights of the wall thickness not including the cladding. This definition means the plate and forging do not have to be inspected. [PWROG-40b, EPRI-40b]

NRC RESPONSE: The NRC agrees that intended examination volume was not clearly described in the rule. Therefore, § 50.61a(e)(1) has been revised to indicate that the volumetric examination volume is the volume described in ASME Code, Section XI, Figures IWB-2500-01 and IWB-2500-02 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.

Further, § 50.61a(e)(3) requires licensees to verify that for any ASME Code, Section XI ultrasonic examination of beltline welds, all flaws between the clad-to-base metal interface and three-eights of the reactor vessel wall thickness from the interior surface are within ASME Code allowable limits.

COMMENT: The volume between the cladding interface and the interior surface of the reactor pressure vessel are not included in the examination volume. ASME Code, Section XI, Appendix VIII, does not qualify ultrasonic testing procedures for this volume. [PWROG-41, EPRI-41]

NRC RESPONSE: The NRC agrees with the commenters' statement that the volume between the cladding interface and the interior surface of the reactor vessel (i.e., the cladding) is not included in the examination volume. The prescribed inspection does not include volumetric examination of the cladding. However, other sections of the proposed rule (i.e., §§ 50.61a(e)(2) and (e)(4)(iii)) address visual or surface examination of the cladding if a flaw at the clad-to-base metal interface is identified.

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

Miscellaneous Comments

COMMENT: There are concerns related to reactor trip events with subsequent main feedwater overfeed in B&W designed reactors. A review indicates that these event sequences have been considered in the PTS probabilistic risk assessment report, but the significance of these events with respect to PTS has been missed in the overall integrated methodology. It is unclear in the reports as to how this situation occurred, but a re-evaluation of the significance of these events should be performed to determine any impact on the underlying technical basis for the proposed rule. Specifically: [DUKE]

(1) Some main feedwater overfeed cases were run, but there is no indication that any RELAP5 overfeed analyses were performed for the B&W design. The B&W design will overcool more rapidly than other PWR designs because of the once-through steam generators. The initial secondary water inventory is low, and the overfeed will immediately influence the rate of heat transfer. The event progresses to a counter flow water-solid heat exchange process, and the temperature of the primary side cold leg water returning from a steam generator will approach the main feedwater temperature. This low cold leg water temperature along with the cold safety injection water has the potential to severely overcool the reactor vessel. Insights based on overfeed analyses for PWR designs with U-tube steam generators are not applicable to the B&W design. [DUKE-1]

(2) The overfeed events that were analyzed are described as only filling to the top of the steam generator. Perhaps this assumption of a limited duration overfeed is supported by the plant design and/or by operator recovery actions credited by the probabilistic risk assessment. A continued overfeed would be more severe relative to PTS. [DUKE-2]

(3) The probabilistic risk assessment report considers a zero power (i.e., low decay heat) initial plant condition. That initial condition is much more severe for main feedwater overfeed events. Thermal-hydraulic analyses of main feedwater overfeed events should consider this initial condition. [DUKE-3]

(4) The following statement in the summary report for the technical basis for the revision of the PTS rule is not correct for a B&W design "... the extent of the cooldown is limited because the ultimate heat sink temperature is the saturation temperature at atmospheric pressure." The extent of the cooldown for a main feedwater overfeed is related to the main feedwater temperature, which will be low at zero power with no preheating, and the primary cooldown will be enhanced by the cold safety injection water. [DUKE-4]

NRC RESPONSE: The commenters described a PTS scenario resulting in a continuous overfeed of the steam generators at Oconee and questioned whether the significance of this scenario with respect to PTS has been missed in the overall integrated methodology. The NRC agrees that the cooldown from a continuous uncontrolled overfeed sequence could be severe and that the PTS documentation did not fully explain how this sequence was evaluated. Therefore, the NRC re-evaluated the sequence as described below.

(1) The commenter is correct that no RELAP 5 analyses were performed for the described sequence. Instead, the postulated sequence was approximated from existing thermal hydraulic information as having a cool down rate equivalent to that of a 16-inch diameter hot leg break initiated from hot-zero power, and a pressure equal to the operating pressure throughout the transient. The NRC recognizes that this cooling rate may underestimate the actual cooling rate of the reactor coolant system inventory. However, as documented in NUREG-1806, more rapid inventory cooling rates cannot be matched by the reactor pressure vessel itself due to the finite thermal conductivity of the steel. The inventory cooling rate associated with the 16-inch diameter hot leg break is rapid enough to generate the maximum thermal stresses in the reactor pressure vessel wall.

(2) The commenter is correct that plant design and operation resulted in removal of the scenario from the detailed analysis in the Oconee probabilistic risk assessment. However, the commenter is not correct that the scenario is more severe relative to PTS than other scenarios included in NUREG-1874. Severity relative to PTS depends on both the frequency of the scenario and the conditional probability of through-wall crack. The frequency of a continuous overfeed scenario can be developed from the Oconee PTS probabilistic risk assessment. The sequence is a reactor trip, failure of the main feedwater runback control system, failure of the high steam generator level trip of the main feedwater pumps, concluding with the operators failing to manually throttle or trip the main feedwater given that it is overfeeding the steam generators. This yields an expected frequency of about 1×10^{-6} per year for this uncontrolled overfeed scenario. This is a relatively infrequent sequence because it requires the failure of two control systems and the failure of the operators to follow procedures and also failure to recognize a very significant event that includes filling the steam lines with water. The FAVOR code, version 06.1, was used to estimate the conditional probability of through-wall cracking of the postulated sequence for the Oconee plant at all four embrittlement levels reported in NUREG-1874. The results indicate that the conditional probability of a through-wall crack given the transient is not expected to be greater than about 1×10^{-5}, at low embrittlement, and about 1×10^{-2}, at the PTS screening limits. These values were multiplied by the event frequency reported above to estimate the TWCF contribution of the postulated sequence. At low embrittlement levels, the conditional probability of a through-wall crack for most sequences is

effectively zero, so any sequence with a non-zero conditional probability of a through-wall crack (about 1×10^{-11} per reactor year) makes a large contribution to the total value. Therefore, the scenario is a dominant contributor to TWCF at low embrittlement levels, but only because the TWCF is very small. However, at embrittlement levels close to the proposed 10 CFR 50.61a screening limit the sequence contribution (of about 1×10^{-8}) to the total TWCF of 1×10^{-6} per year is insignificant because the relatively high conditional probability of a through-wall crack is more than offset by the very low frequency of the sequence.

(3) The commenter is correct that the zero power transient is more severe. The thermal hydraulic evaluation described above corresponded to hot zero power. Rather than also evaluate the thermal hydraulic transient from at-power, the at-power scenario frequency was conservatively applied to the hot zero power thermal hydraulic results.

(4) The statement "... the extent of the cooldown is limited because the ultimate heat sink temperature is the saturation temperature at atmospheric pressure" is correct for the limited overfeed scenario included in the summary report. The statement is not applicable to the continuous overfeed scenario. Based on the analysis of the continuous overfeed scenario, the NRC concludes that the sequence was appropriately determined to be an insignificant contributor and that including the sequence would not change the PTS screening limits established in the proposed rule.

The NRC concludes that the technical basis for the rule is not adversely affected by the consideration of the sequence identified by the commenter. No changes were made to the rule language as a result of this comment.

COMMENT: Remove the citation to RG 1.174, Revision 1, as basis for TWCF acceptance criteria or explain the differences. RG 1.174, Revision 1, Section 2.2.5.5, states that the acceptance of risk results relative to the limits is to be evaluated using mean values. However, the RT_{MAX} limits in the proposed rule and in NUREG-1874 are based upon the 95^{th} percentile values, which are much higher than the mean values of TWCF as shown in NUREG-1874, Table 3.1. If the technical basis calculations need to be redone for any reason, the mean values of TWCF should be used instead of the 95 percent upper bounds. [PWROG-26, EPRI-26]

NRC RESPONSE: The NRC disagrees that the mean values of TWCF should be used instead of the 95^{th} percentile upper bounds, whether the technical basis is redone or not. There are several differences between the risk metrics in RG 1.174 and the 1×10^{-6} per reactor year TWCF criteria. For example, the RG 1.174 acceptance guidelines identify increases in risk that would normally be acceptable but the TWCF is a total estimated frequency. The risk guidelines from RG 1.174 are discussed briefly in the TWCF technical basis in order to provide a reference to a quantitative frequency that is normally considered acceptably small for an undesired event. In addition to the difference in the estimated parameter, the PTS analysis is an evaluation to support a change to a regulation which is subject to Commission review and approval and to public comment. Consequently, the PTS work need only follow the RG guidance to the extent that the NRC determines that the RG guidance is applicable and appropriate. During the development of the PTS RT_{MAX-X} limits, the NRC noted that the highly skewed distribution resulted in mean values that lay in the higher percentile values and were always greater than the 75^{th} percentile. The NRC concluded that the 95^{th} percentile upper bound is preferable to the mean because more consistent RT_{MAX-X} limits could be derived by using the fixed percentile than by using the mean value whose percentile changes with the embrittlement level.

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

COMMENT: Clarify if the NRC would be receptive to licensees pursuing an exemption request to § 50.61 given the significant amount of resources required to reevaluate the vessel in accordance with the requirements in § 50.61a and that some plants have already made a significant investment determining RT using an alternative method (i.e., Master Curve).

The commenters suggested that the NRC continue to allow exemptions in the future for determining RT_{PTS} using the current PTS rule. [PWROG-1, EPRI-1]

NRC RESPONSE: Section 50.12 of the 10 CFR identifies the requirements for allowing exemptions to NRC regulations. NRC will evaluate exemption requests submitted by licensees in accordance with the requirements of § 50.12.

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

COMMENT: Add a statement that states that this rule is applicable to the current PWR fleet and not the new plant designs. 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. [PWROG-5, EPRI-5]

NRC 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 the effective date of the 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 the effective date of the final rule and whose reactor vessel was designed and fabricated to the 1998 Edition or earlier of the ASME Code.

COMMENT: Revise the rule language to read "The information required by §§ 50.61a(c)(1) and (c)(3) must be submitted for review and approval by the Director of NRR at least three years before the limiting RT_{PTS} value calculated under § 50.61 is projected to exceed the PTS screening criteria in § 50.61 for plants licensed under 10 CFR Part 50 or 10 CFR Part 52. A schedule to provide the information required by § 50.61a(c)(2) shall be submitted at the same time."

Section 50.61a(c) of the proposed rule states the information required by §§ 50.61a(c)(1), (c)(2), and (c)(3) must be submitted for review and approval by the Director of NRR at least three years before the limiting RT_{PTS} value calculated under § 50.61 is projected to exceed the PTS screening criteria in § 50.61. In the case of Palisades, this information is required to be submitted by December 31, 2010. Palisades has two refueling outages scheduled prior to that date (i.e., Spring 2009 and Fall 2010). Given that the Fall 2010 outage is close to the required submittal date, the Spring 2009 outage is the preferred date for performing the inspection. Performing an inservice inspection on such short notice is certainly an enormous and unexpected misuse of resources. The licensees attempt to operate using at least a five year planning horizon. [PWROG-6, EPRI-6]

NRC RESPONSE: The NRC does not agree with the commenters' proposed change. The requirement to submit all the assessment information to the Director of NRR at least three years before the limiting RT_{PTS} value calculated under § 50.61 is projected to exceed the PTS screening criteria in § 50.61 for plants licensed under 10 CFR Part 50 is necessary because it allows the NRC adequate time to review the information and resolve any issues prior to the reactor vessel exceeding the screening criteria. The commenters have not provided any information that would indicate a generic or reoccurring problem exists that would require the NRC to provide criteria in the rule to allow licensees to submit the required inspection information less than three years before the limiting RT_{PTS} value is projected to exceed the PTS screening criteria in § 50.61. The NRC recognizes that licensees may, under the provisions of § 50.12, seek an exemption from § 50.61a(c) to request permission to modify the timing of submittals required by this rule. The NRC will evaluate any requests for exemption from the submittal requirements of § 50.61a(c) on a case-by-case basis. Therefore, the NRC has decided not to adopt the commenters' proposed change.

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

COMMENT: Clarify if the beltline area to be examined per § 50.61a(c)(2) of the proposed rule is limited to the "limiting materials" or if this requires the entire beltline under an owner's inservice inspection program to be evaluated. [PWROG-7a, EPRI-7a]

NRC RESPONSE: The requirement is to examine the entire beltline. The area to be examined includes all of the welds and adjacent base material defined by the ASME Code, Section XI inspection requirements. Examination results may be utilized to meet flaw density, size and location requirements provided that the examination satisfies the criteria described in § 50.61a(e) of the rule (i.e., examination performed using procedures, equipment and personnel that have been qualified under the ASME Code, Section XI, Appendix VIII, Supplements 4 and 6, as specified in § 50.55a(b)(2)(xv)).

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

COMMENT: Clarify if § 50.61a(c)(2) of the proposed rule imposes a stand-alone special examination, or if the most recent ASME Code, Section XI examination can be used to satisfy this requirement. [PWROG-7b, EPRI-7b]

NRC RESPONSE: The ASME Code, Section XI examination may be used to satisfy this section of the rule if the most recent examination was performed using procedures, equipment and personnel that have been qualified under the ASME Code, Section XI Appendix VIII, Supplements 4 and 6, as specified in § 50.55a(b)(2)(xv).

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

COMMENT: Clarify what is defined as a "significant" change in RT_{MAX-X}.

Revise the proposed rule to define a significant change in RT_{MAX-X} as one where there is an increase in projected fluence greater than 20 percent. A 20 percent increase is equivalent to the uncertainty allowed in RG 1.190 and also equivalent to 2 standard deviations on the global fluence that is input to the FAVOR evaluations in NUREG-1874. [PWROG-8, EPRI-8]

NRC RESPONSE: The NRC does not agree with the comment. A clarification is not necessary because § 50.61a(d)(1) of the proposed rule already defines a significant change in the projected value of RT_{MAX-X} as one where "the previous value, the current value or both values, exceed the screening criteria prior to the expiration of the plant operating license." The definition of a "significant change" that is being applied in § 50.61a is consistent with the definition in § 50.61, footnote 2, regarding a significant change to a facility's RT_{PTS} value.

The NRC does not agree with the proposed change because defining a significant change in RT_{MAX-X} as one where there is an increase in projected fluence greater than 20 percent, equivalent to 2 standard deviation of the global fluence that is input to the FAVOR evaluation, is not acceptable because a percentage change in neutron fluence does not define a significant regulatory change. A significant regulatory change is one when the value of RT_{MAX-X} is projected to exceed the screening criteria.

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

COMMENT: Modify § 50.61a(d)(2) to clarify that the 120 days applies only for subsequent applications of the PTS rule (i.e., after the initial application of the voluntary PTS Rule). [PWROG-9, EPRI-9]

NRC RESPONSE: The NRC believes that the rule language is clear and that a change is not necessary. Paragraph (c) of § 50.61a describes the actions needed to request an approval to use this rule. This paragraph states that licensees must submit a license amendment requesting approval to use this rule. Specifically, § 50.61a(c)(2) identifies the flaw assessment information that that must be included with the license amendment request.

Further, § 50.61a(d) describes subsequent actions to be taken by licensees whose license amendment request to utilize § 50.61a has been approved. Specifically, § 50.61a(d)(2) establishes the requirements for submitting the flaw assessment results that are determined after the requirements of § 50.61a(c) have been completed. Paragraph (d)(2) indicates that the subsequent flaw assessment shall be submitted for review and approval to the NRC within 120 days after completing a volumetric examination of reactor vessel beltline materials as required by ASME Code, Section XI.

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

COMMENT: Clarify that the ASME Code, Section XI, Edition and Addenda to be used is the one that the licensee is currently working to in their inservice inspection program. If this is not the case, specify which Edition or Addenda shall be used and the basis for requiring this. [PWROG-11, EPRI-11]

NRC RESPONSE: The NRC agrees that the proposed rule should be modified to specify which edition or addenda shall be used. The ASME Code, Section XI, edition and addenda to be used is specified in § 50.55a(b)(2)(xv). 10 CFR 50.55a(b)(2)(xv) identifies NRC approved editions of the ASME Code and modifications to ASME Code, Section XI, Appendix VIII ultrasonic qualification requirements. Section (e) of the proposed rule has been modified to include the requirements of § 50.55a(b)(2)(xv). 10 CFR 50.55a(g)(6)(ii)(C) requires licensees to implement ASME Code, Section XI, Appendix VIII, Supplements 4 and 6 as of November 22, 2000.

COMMENT: Make the technical basis document for the proposed correlation available to the public. If a plant is to compare to data from other surveillance programs, it is preferred that the data (e.g., ΔT_{30}) be determined consistently (e.g., the same tanh curve shaping method). [PWROG-20, EPRI-20]

NRC RESPONSE: The NRC's technical basis for the correlation in the rule is documented in ADAMS Accession Nos. ML081000629 and ML081000630.

Paragraph (f)(6)(i)(A) specifies the ΔT_{30}, whether it be determined from the plant's surveillance program of other surveillance program, must be determined by the requirements of 10 CFR Part 50, Appendix H. Therefore, the NRC concludes that a clarification is not necessary.

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

COMMENT: Modify equation 8 from "Residual ® =…" to "Residual (r) =…." [PWROG-22, EPRI-22]

NRC RESPONSE: The NRC agrees that this is a typographical error. Equation 8 has been corrected in the rule as a result of this comment.

COMMENT: Revise the proposed rule to provide clarification regarding forging materials susceptible to underclad cracking. Table 1 of the proposed rule provides different PTS screening criteria for "forging without underclad cracks" and "forging with underclad cracks". NUREG-1874 provides clarification that reactor vessels that have been fabricated in accordance with RG 1.43 can be considered to not be susceptible to underclad cracking. No guidance or criteria is provided in the proposed rule for determining whether or not the forging material is susceptible to underclad cracking. [PWROG-23, EPRI-23]

NRC RESPONSE: The NRC agrees that the rule language should be clarified to ensure that licensees have adequate guidance to determine whether forgings have or do not have underclad cracks. The category of "forging without underclad cracks" applies to forgings for which no underclad cracks have been detected by examination and which were fabricated in accordance with RG 1.43. The category of "forging with underclad cracks" applies to forgings that either have had underclad cracks detected by examination or were not fabricated in accordance with RG 1.43. As a result of this comment, the NRC has added footnotes 6 and 8 to Table 1 to clarify the rule language.

COMMENT: The proposed rule states that "Each licensee shall compare the projected RT_{MAX-X} values for plates, forgings, axial welds, and circumferential welds to the PTS screening criteria ..." Add the term "in Table 1 of this section" after the term "screening criteria". [PWROG-24, EPRI-24]

NRC RESPONSE: The NRC agrees with this comment is acceptable because it provides a cross-reference to Table 1 and clarifies the rule language. As a result of this comment, § 50.61a(c)(3) of the rule language has been modified.

COMMENT: Clarify in § 50.61a(c)(3) of the proposed rule that for reactor vessels with plates and axial welds, the screening criteria of the RT_{MAX-AW}, RT_{MAX-PL}, and combination must be met. The proposed rule states that "Each licensee shall compare the projected RT_{MAX-X} values for plates, forgings, axial welds, and circumferential welds to the PTS screening criteria…" However, Table 1 also includes a screening criterion for a combination of RT_{MAX-AW} and RT_{MAX-PL} that may be more restrictive than the separate RT_{MAX-AW} and RT_{MAX-PL} criteria. [PWROG-25, EPRI-25]

NRC RESPONSE: The NRC agrees with the comment. The NRC believes that the use of the term "projected RT_{MAX-X} values or sum of RT_{MAX-AW} and RT_{MAX-PL} values" provides clarification to the rule. As a result of this comment, the definition of the term RT_{MAX-X} in § 50.61a(a)(6) has been modified.

COMMENT: If the screening limits for RT_{MAX} in Table 1 are not satisfied, then the same compensatory measures identified in the existing PTS rule, § 50.61 (i.e., paragraphs from flux reduction through thermal annealing) must be submitted with the requests for review and approval by the Director of NRR and implemented prior to when the limits are projected to be violated. Note that the option of calculating the TWCF using the maximum RT_{MAX} values for each type of beltline material (i.e., axial or circumferential weld, plate or forging) with the curve-fit equations 3-5 in NUREG-1874, Section 3.3.1.3, and showing that it is less than the risk limit of $1x10^{-6}$ events per year is not included. [PWROG-29a, EPRI-29a]

NRC RESPONSE: The NRC does not agree with the comment. The option to perform an evaluation using the curve-fit equation 3-5 in NUREG-1874, Section 3.3.1.3, which demonstrates that a particular vessel is below the risk limit of $1x10^{-6}$ is included within the scope of § 50.61a(d)(4) of the proposed rule. The approach suggested in the proposed change is one of many possible alternatives that a licensee might choose to use on a case-by-case basis.

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

COMMENT: Revise § 50.61a(d)(3) of the proposed rule to include the option of first calculating the TWCF using the maximum RT_{MAX} values for each type of beltline material with the curve-fit equations NUREG-1874 and showing that it is less than the risk limit of $1x10^{-6}$ events per year. If this is not successful, then the remaining options in § 50.61a(d)(3) would be invoked. [PWROG-29b, EPRI-29b]

NRC RESPONSE: The NRC does not agree with the commenters' proposal. Paragraphs (d)(4) and (d)(5) describe the fact that analyses that demonstrate acceptable frequency of through-wall cracking may be performed and submitted for approval. Licensees are not precluded by the language in §§ 50.61a(d)(3), (d)(4), or (d)(5) from performing such analyses prior to, or in parallel with, the other actions described in §§ 50.61a(d)(3), (d)(4), or (d)(5). The commenters have not shown, and the NRC has not identified, a reason why the rule should be modified. Hence, the NRC declines to adopt the commenters' proposal.

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

COMMENT: The default Mn values in Table 4 of the proposed rule should be consistent with the mean values in Tables 3.3 and 3.4 and the root mean square value of global and local standard deviations in NUREG-1874, Appendix A, Task 1.6. The default limits on Manganese

(Mn) in Table 4 look high, especially for welds and forgings, relative to their stated intent (mean plus one sigma) and the actual data in NUREG-1874. [PWROG-36, EPRI-36]

NRC RESPONSE: The NRC does not agree that the values in Table 4 of the proposed rule should be consistent with the values in NUREG-1874 because the two sets of values are used for different purposes. The values in Table 4 are to be used as default values when licensees do not know the value for their welds, plates, or forgings. The values in NUREG-1874 were used to characterize material properties in the FAVOR code while simulating multiple flaws within a single weld, plate or forging. There is much less variability in the chemical composition of the materials around different flaws within a single weld, plate and forging than there is between different welds, plates and forgings across the fleet of PWRs. The values in Table 4 are composite values developed from information on the chemical properties from the variety of welds, plates, and forging from the population that is actually in service. The use of mean plus one sigma values was determined by the NRC to be appropriately conservative when no material-specific value for manganese is available. The basis for these values can be found in the technical basis which is documented in ADAMS Accession Nos. ML081000629 and ML081000630.

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

COMMENT: Clarify why there are no operating PWRs in column 2 of Table 1 of the proposed rule. [PWROG-38, EPRI-38]

NRC RESPONSE: The NRC recognizes that no currently operating PWRs fall into the thickness bin represented by column 2 of Table 1. This column was included only in the interest of completeness.

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

COMMENT: Replace the term, "Appendix VIII, Supplement 4," with "IWB-2000." The AMSE Code, Section XI, Appendix VIII, does not provide the examination volume for Inservice Inspections. [PWROG-39, EPRI-39]

NRC RESPONSE: The NRC agrees with the comment but does not agree with the proposed change. Based on the results from the PTS study, flaws within the one inch of the clad-to-base metal interface are the flaws that contribute most significantly to TWCF. Therefore, instead of inspecting the entire volume of weld, as recommended in IWB-2000, the flaw assessment should concentrate on flaws within one inch of the clad-to-base metal interface. Therefore, the inspection volume in IWB-2000 is not suitable for PTS significant flaws. However, in response to other public comments the NRC modified § 50.61a(e) of the rule language to clarify the definition of "inspection volume."

COMMENTS RELATED TO THE SUPPLEMENTAL PROPOSED RULE

Comments Related to Adjustments of Volumetric Examination Data

COMMENT: Modify § 50.61a(e) 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. [PWROG2-1 and EPRI2-1]

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.

NRC 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 number requirements of Tables 2 and 3. This requirement would be accomplished by requiring licensees to base the 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 to the Director of NRR. 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.

Comments Related to Surveillance Data

COMMENT: Remove test reactor data from the definition of "surveillance data" in §§ 50.61a(a)(10) or (f)(6) should be amended to limit the required evaluations to surveillance data generated in commercial power reactor surveillance programs.

Test reactor data is included under the definitions of surveillance data (§ 50.61a(a)(10)). This seems to imply that test reactor data should be included in the evaluations described in § 50.61a(f)(6). The commenters believe that it is not technically correct to require evaluation of test reactor data in conjunction with power reactor data. The technical basis for the revision of RG 1.99 (ADAMS Accession Number ML081120289) shows that test reactor data significantly deviates from the power reactor data at high fluence and would likely cause impacted heats to violate the criteria in § 50.61a(f)(6)(ii). [PWROG2-2 and EPRI2-2]

NRC RESPONSE: The NRC agrees with the comment and the proposed change. Licensees should not be required to evaluate test reactor data in conjunction with power reactor data. Test reactor data may not be directly applicable to commercial power reactors since the radiation environment (e.g., neutron flux and spectrum) of the test reactor can be significantly different than the radiation environment of the power reactor. The NRC's intent for the evaluation of the surveillance data in § 50.61a(f)(6) was to require licensees to use surveillance data from material capsules that were removed from commercial power reactors. Hence, the surveillance data definition in § 50.61a(a)(10) was modified to eliminate the phrase "data from test reactors." Test reactor data may, however, be used if a licensee demonstrates its applicability to the commercial power reactor vessel materials being evaluated.

COMMENT: The proposed methodology for assessing potentially significant deviations of actual surveillance data for plant-specific heats from the predicted values has not been extensively tested by industry. It is apparent that guidance will be needed to perform the evaluation required in § 50.61a(f)(6)(vi). The industry intends to prepare such guidance for licensees to perform the data review and evaluation discussed in § 50.61a(f)(6)(vi) when these types of deviations are identified. This guidance will be provided to the Director of NRR for review and approval. [PWROG2-3a and EPRI2-3a]

NRC RESPONSE: The NRC agrees with the commenters' statement that that the methodology for assessing potentially significant deviations of actual surveillance data for plant-specific heats from the predicted values has not been extensively tested by industry. Therefore, the NRC understands that the number of plants who may potentially fail the §§ 50.61a(f)(6)(ii), (iii) or (iv) criteria and be required to apply § 50.61a(f)(6)(vi) has not been identified. However, the rule language clearly states the information that must specifically be provided for NRC review and approval if a licensee performs the evaluation in § 50.61a(f)(6)(vi). If industry guidance documents are developed, the NRC will consider them when submitted for review and approval by the Director of NRR.

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

COMMENT: The required surveillance checks cover three types of potential deviations from trend curve predictions. The first surveillance check is to address an offset bias and the third is to address significant outliers. Although no changes in these surveillance checks are proposed, guidance will be needed to perform the evaluation required in § 50.61a(f)(6)(vi). [PWROG2-3b, EPRI2-3b, PWROG2-3d and EPRI2-3d]

NRC RESPONSE: The NRC disagrees with the commeters suggestion that guidance will be needed to perform the § 50.61a(f)(6)(vi) evaluation. The final rule language clearly states the information that must specifically be provided for NRC review and approval if a licensee performs the evaluation in § 50.61a(f)(6)(vi). The NRC understands that the commenters will be developing guidance documents [as stated in PWROG2-3a and EPRI2-3a]. If industry guidance documents are developed, the NRC will consider them when submitted for review and approval by the Director of NRR.

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

COMMENT: Eliminate the second surveillance check from the rule since the slope change evaluation appears to be of limited value.

The second required surveillance check is to address a slope change. The intent of this section appears to be 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 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 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. [PRWOG2-3c and EPRI2-3c]

NRC 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. Since 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.

Final Regulatory Analysis Related to Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events (10 CFR 50.61a)

RIN 3150-AI01
[NRC-2007-0008]

This document presents a final regulatory analysis for the proposed revisions to the Pressurized Thermal Shock (PTS) Rule as set forth by the U.S. Nuclear Regulatory Commission (NRC) in Title 10, Section 50.61, of the *Code of Federal Regulations* (10 CFR 50.61). The proposed rule was undertaken as the result of a June 30, 2006, staff requirements memorandum (SRM), "Staff Requirements - SECY-06-0124 - Rulemaking Plan to Amend Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events (10 CFR 50.61)." In this SRM, the Commission directed the staff to pursue the rulemaking as described in Option 2 of the May 26, 2006, Commission paper, SECY-06-0124, containing the "Rulemaking Plan to Amend Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events (10 CFR 50.61)." The NRC published the proposed rulemaking on the alternate fracture toughness requirements for protection against PTS events for public comments on October 3, 2007 (72 FR 56275).

During the development of the PTS final rule, the NRC determined that several significant changes to the proposed rule language would be needed to adequately address the stakeholders' comments and their associated implementation concerns. Two of the modifications are significant changes to the proposed rule language on which external stakeholders did not have an opportunity to comment. The NRC concluded that obtaining stakeholder feedback on these provisions through the use of a supplemental proposed rule was appropriate. The supplemental proposed rule was published for public comments on August 11, 2008 (73 FR 46557). The two modifications addressed in the supplemental proposed rule did not have a measurable impact in this regulatory analysis. However, in the supplemental proposed rule, the NRC proposed limiting the applicability and the use of 10 CFR 50.61a. The regulatory analysis was modified to reflect this change. The NRC has decided to adopt the PTS final rule. The final rule incorporates the two proposed modifications described in the supplemental proposed rule. Therefore, the technical aspects of this regulatory analysis remain unchanged from those issued in the regulatory analysis for the supplemental proposed rule.

1.0 Statement of the Problem and Reasons for the Rulemaking:

The PTS rule, 10 CFR 50.61, adopted on July 23, 1985 (50 FR 29937), establishes screening criteria to evaluate when a reactor vessel may be susceptible to failure due to a PTS event. The screening criteria define a limiting level of embrittlement beyond which operation cannot continue without further plant-specific evaluation. Any pressurized water reactor (PWR) vessel with materials predicted to exceed the screening criteria in 10 CFR 50.61 may not continue operation without implementation of compensatory actions or plant-specific analyses unless the licensee receives an exemption from the requirements of the rule. No currently operating PWR is projected to exceed the 10 CFR 50.61 screening criteria before the expiration of its 40 year operating license. However, several plants are approaching the screening criteria, while others are likely to exceed the screening criteria during their first license renewal periods.

The NRC staff has completed a research program to update the PTS regulations. The results of this research program conclude 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 10 CFR 50.61 are unnecessarily conservative and may impose an unnecessary burden on some licensees. Therefore, the NRC is adopting a new rule, 10 CFR 50.61a. The objective of the final rule is to provide alternative screening criteria and corresponding embrittlement correlations for licensees seeking regulatory relief from the overly conservative requirements of the current PTS regulation, 10 CFR 50.61.

Further, the NRC has determined that the backfit rule, 10 CFR 50.109, does not apply to this final rule because compliance with the requirements of the final rule (10 CFR 50.61a) would be an alternative to compliance with the requirements of the current PTS rule (10 CFR 50.61). Due to the voluntary implementation of this amendment, this final rule does not constitute backfitting as defined in 10 CFR 50.109(a)(1), and a backfit analysis is not required.

2.0 Identification of Alternatives

Following the Commission's direction contained in the June 30, 2006, SRM, the staff considered several alternatives to amend the regulation.

Alternative 1: Take no action.

Under Alternative 1, the "no action" alternative, the NRC would not amend the current regulations regarding PTS events. The current requirements of 10 CFR 50.61 would remain in effect and would continue to apply to all current and future PWR licensees.

The "no action" alternative serves as the baseline against which the costs and benefits of the other alternatives are measured. Under the current rule, licensees with reactor pressure vessels (RPVs) that do not meet the current screening limits may implement several compensatory measures, such as flux reduction, submission of plant-specific analyses, and vessel annealing, each of which impose a cost burden on the licensee. Alternatively, licensees may request exemptions from 10 CFR 50.61 to use, for example, plant-specific toughness analyses different from those required by the current rule. Absent the compensatory measures, licensees who exceed the screening limits would be required to cease operation.

Alternative 2: Require all PWRs to implement the requirements in 10 CFR 50.61a.

Under Alternative 2, the NRC would promulgate a new PTS rule which would require all PWR licensees to apply the updated PTS screening criteria and embrittlement correlations. The requirements in this proposed rule would replace the requirements in the current 10 CFR 50.61. All PWR licensees would be required to meet the requirements of the new rule. As a result, current licensees would be required to perform analyses to evaluate their plant(s) using the new embrittlement correlations to assess compliance with the new screening criteria, thereby demonstrating their compliance with the new regulation. Future licensees referencing a certified design would be required to perform similar re-analyses under the new rule. Future licensees not referencing a certified design would be required to comply with the new rule.

All current PWR licensees and future licensees referencing certified designs would be required to comply with the new rule and would incur additional regulatory burden. This additional burden would be caused by the requirement to re-analyze the plant PTS reference temperature (RT_{MAX-X}) values under the new rule, where the design has previously been licensed or certified under the analysis methods and screening criteria defined in the current rule. This would constitute a backfit under 10 CFR 50.109 for those licensees. Future licensees not referencing a certified design would only perform the analyses required in the new rule. However, in this case, it would not constitute a backfit because the licensee had not previously been granted approval of the plant design based on the current rule.

Alternative 3: Permit voluntary compliance with a new PTS rule for existing PWR licensees and require mandatory compliance with the new rule for new PWR licensees.

Under Alternative 3, the NRC would promulgate 10 CFR 50.61a which would be (1) an alternative to requirements of 10 CFR 50.61 for any current PWR reactor with an operating license or combined license in place before the effective date of the rule or new PWR reactor referencing a design certified before the effective date of the rule, and (2) mandatory for any new PWR reactor with an operating license or combined license in place after the effective date of the rule. All PWR licensees would be required to meet the requirements of 10 CFR 50.61 or 10 CFR 50.61a, depending on the date of their license or design certification and whether they choose to implement the new rule.

Licensees under (1) described above would incur no additional regulatory burden, since the rule would be voluntarily implemented. Licensees under (2) described above would be required to comply with 10 CFR 50.61a, but this would not be a backfit because the licensee had not previously been granted approval of the plant design based on the current rule.

Alternative 4: Permit all PWR licensees to implement either the current 10 CFR 50.61 or the proposed 10 CFR 50.61a.

Under Alternative 4, the NRC would promulgate 10 CFR 50.61a as an alternative to the requirements of 10 CFR 50.61. All PWR licensees would be required to meet the conditions of 10 CFR 50.61, or as an alternative, would be required to comply with 10 CFR 50.61a. This alternative would not constitute a backfit for any licensee because 10 CFR 50.61a would be implemented by any PWR licensee who found it advantageous to do so. PWR licensees who are projected to exceed the 10 CFR 50.61 screening criteria during the lifetime of their plant license would likely comply with 10 CFR 50.61a rather than implement the more expensive compensatory actions specified in 10 CFR 50.61. PWR licensees who are not projected to exceed the 10 CFR 50.61 screening criteria would not likely comply with 10 CFR 50.61a due to the unnecessary cost of implementation.

3.0 Estimation and Evaluation of Values and Impacts

This section describes the analysis conducted to identify and evaluate the benefits (values) and costs (impacts) of this final rule. Section 3.1 identifies the attributes that the final rule is expected to affect. Section 3.2 describes the methodology used to analyze the benefits and costs associated with changes to the affected attributes. The results of the analysis are presented in Section 4.

3.1 Identification of Affected Attributes

This section identifies the factors that the rulemaking is expected to affect. These factors are classified as "attributes" using the list of potential attributes provided in Chapter 5 of the NRC's "Regulatory Analysis Technical Evaluation Handbook."[1] Affected attributes from the handbook include the following:

- Industry Implementation

  Implementation of the final rule would require a licensee to submit a license amendment to the NRC for review and approval. This license amendment request would include analyses of the licensee's vessel under the embrittlement correlations and screening criteria in the final rule through the plant's end of life (40 or 60 years, as applicable). This analysis is required to demonstrate the licensee's compliance with the new regulation. The licensee would be required to perform analyses of the volumetric examination of the vessel to ensure that the screening criteria and calculation methodology are applicable. Additionally, the licensee would be required to report the manganese and phosphorus content of the reactor vessel beltline materials.

- Industry Operation

  If implemented, the final rule would differ from 10 CFR 50.61 only in that, during plant operation, a licensee would be required to perform analyses of the volumetric examination of the vessel to ensure that the screening criteria and calculation methodology are applicable.

- NRC Implementation

  The NRC would be required to review and approve license amendment requests, including the submittal of the analysis of the volumetric examination inspection results of the vessel under the amendment, to comply with 10 CFR 50.61a.

- NRC Operation

  The NRC would be required to review and approve the submittal of subsequent volumetric examinations of the vessel to ensure that the screening criteria and calculation methodology are applicable.

- Improvements in Knowledge

  The NRC and the nuclear industry would acquire additional data concerning vessel weld flaws due to the additional analyses of the volumetric examination inspection results under 10 CFR 50.61a. Each plant implementing the final rule would contribute to improvements in NRC's and industry's knowledge of how well the new PTS rule fracture toughness requirements apply to current reactor vessels. The additional insights gained

[1] NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook: Final Report," U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, January 1997.

from these inspections could be used in future research projects, with the potential for further revisions to the PTS rule.

- Regulatory Efficiency

  The NRC staff is of the opinion that adopting the final rule as an alternative rule is the most efficient approach. This is accomplished by allowing the licensees to select the option that best serves their situation without any effect on the public health and safety and common defense and security.

The final rule is *not* expected to affect the following attributes:

- public health (accident and routine)
- occupational health (accident and routine)
- property (onsite and offsite)
- other government
- general public
- antitrust considerations
- safeguards and security considerations; and
- environmental considerations.

3.2 Estimation of Values and Impacts

- Industry Implementation

  The projected cost of a licensee implementing the final PTS rule is estimated at 0.6 full-time equivalent (FTE)[2], or approximately $90,000. This implementation consists of performing the required analyses, preparing the associated license amendment request, and submitting it for review and approval by the NRC.

- Industry Operation

  The projected additional cost to a licensee performing the analysis of the volumetric examination inspection results is estimated at 0.3 FTE, or approximately $46,000, per analysis. This includes performing the analysis and submitting it for review and approval by the NRC. It would be performed with the vessel inspection frequency (currently every 10 years). This regulatory analysis assumes that the rule will take effect in 2009. The timeframe for which costs are estimated is based on the remaining operating lives of the relevant facilities. For this analysis, costs are estimated over an assumed period of 48 years, with costs discounted at a 7-percent and 3-percent discount rate every 10 years, as specified in NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook." The analysis makes a simplifying assumption that an average plant's next vessel inspection will occur 5 years from the rule's implementation date and every 10 years thereafter through the assumed lifetime of the plant, including license renewal.

[2] All cost estimates in this analysis are based on the NRC estimate of labor rates of $105 per hour or an annual rate of $152,000 assuming 1446 hours worked in a year. The labor rates used were the ones in effect at the time the analysis was performed.

Assuming the 3-percent discount rate results in a discounted flow of funds of approximately $116,200, while the 7-percent rate gives an estimated value of around $60,200. Therefore, operating under 10 CFR 50.61a, those licensees would incur costs projected to exceed those for operating under 10 CFR 50.61. For licensees not projected to exceed the current PTS screening criteria within their plant lifetime, the NRC staff does not expect that any licensees would benefit from implementing and operating under 10 CFR 50.61a due to the cost of implementation and the inspection results analyses required as described earlier, and would not change the licensee's cost.

- NRC Implementation

  The NRC implementation costs are estimated at 0.5 FTE or $76,000 in labor costs to review each license amendment request. However, this cost must be compared with the NRC's costs of having licensees operate under the existing rule. Each licensee projected to exceed the current PTS screening criteria within their plant lifetime would be expected to take compensatory actions in 10 CFR 50.61. The extent of NRC resources would depend on the compensatory actions taken. The NRC staff estimates that the resources required (per licensee) could range from 0.1 to 2.0 FTE or from $15,000 to $300,000. Therefore, for this attribute, the impact could range from a small savings to an increase in costs to the NRC when a licensee would opt for the using the existing 10 CFR 50.61 instead of the amended option.

- NRC Operation

  The projected additional cost to the NRC for reviewing the analysis required by the final rule is estimated at 0.1 FTE or $15,000 and would be performed with the vessel inspection frequency (currently once every 10 years). Assuming the same timeframe as used in the derivation of the industry operation costs, the discounted flow of funds for NRC implementation per licensee is estimated at $38,000 using a 3-percent discount rate and estimated at $20,000 using the 7-percent rate. There are no alternatives to operating under the final rule after it has been implemented. For licensees not projected to exceed the current PTS screening criteria within their plant lifetime, the NRC staff does not expect that any current or future licensees would benefit from implementing and operating under the final rule, due to the additional implementation and inspection results analyses required as described earlier, and would not change the NRC's cost.

- Regulatory Efficiency

  Regulatory efficiency is attained by permitting PWR licensees to select the option that is most suitable for their situation without affecting public health and safety or common defense and security. Further, the impact on the NRC is minimal.

4.0 Presentation of Results

This section presents the estimates of the benefits and costs in Section 4.1 and the disaggregation analysis in Section 4.2

4.1 Benefits and Costs

The analyses performed in the technical basis for this rulemaking indicate that the degree of PTS challenge for anticipated lifetimes and operating conditions for current operating PWRs is low. Further, the U.S. domestic commercial operating fleet of 69 PWRs has a low probability of exceeding either the limit on the maximum estimated mean through-wall crack frequency of $5x10^{-6}$ per year expressed by current PTS regulations or the final rule value of $1x10^{-6}$ per year, consistent with the Commission's direction in their SRM for SECY-06-0124, on the PTS Rulemaking Plan. As a result, the risk of PTS events is much lower than previously estimated. Therefore, the screening criteria in 10 CFR 50.61 are considered unnecessarily conservative and may impose unnecessary burden on licensees. These results provide the basis to support a relaxation of the current PTS regulations while continuing to provide adequate protection to public health and safety.

This rulemaking action, which would be adopted as an alternative to the current requirements, would result in a burden reduction for some of those licensees with no increase in risk to the public's health and safety.

The current PTS rule, 10 CFR 50.61, requires licensees to take compensatory actions when the value of RT_{PTS} for any material in the beltline is projected to exceed the PTS screening criterion using the plant's projected end of license (EOL) fluence. First, the licensee shall implement those flux reduction programs that are reasonably practical to avoid exceeding the PTS screening criteria. If a licensee has no reasonably practical flux reduction program that will prevent RT_{PTS} from exceeding the PTS screening criteria using the EOL fluence, the licensee is required to submit a safety analysis to determine what, if any, modifications to equipment, systems, and operation are necessary to prevent potential failure of the reactor vessel as a result of the postulated PTS events if continued operation beyond the screening criteria is allowed. Reactor vessel annealing may also be implemented by a licensee to prevent exceeding the screening criteria.

Under the proposed 10 CFR 50.61a, licensees that are projected to exceed the existing requirements in 10 CFR 50.61 before the expiration of their licenses would not be required to comply with the compensatory action requirements described in the preceding paragraph.

However, the alternatives to implementing the final rule for licensees that are projected to exceed the PTS screening criteria within their plant lifetime are to either perform the compensatory actions or to cease operation under 10 CFR 50.61. The cost of compensatory actions in 10 CFR 50.61, including performing flux reduction, vessel annealing, and other analyses, are estimated at $50 million, well exceeding the cost of implementing the final rule. Further, the cost of ceasing operation and purchasing replacement power would exceed the cost of implementing the final rule, because the replacement energy cost is estimated at $1 million per day. Therefore, implementing the final PTS rule, 10 CFR 50.61a, would provide savings to licensees projected to exceed the PTS screening criteria during their plant lifetimes. For licensees not projected to exceed the PTS screening criteria within their plant lifetime, the NRC staff does not expect that any licensees would benefit from implementing 10 CFR 50.61a, due to the additional costs associated with the required implementation analyses as described earlier.

4.2 Disaggregation

In order to comply with the guidance provided in Section 4.3.2 ("Criteria for the Treatment of Individual Requirements") of the Regulatory Analysis Guidelines[3], the NRC conducted a screening review to ensure that the aggregate analysis does not mask the inclusion of individual rule provisions that are not cost beneficial when considered individually and not necessary to meet the goals of the rulemaking. The NRC has determined that this final rule does not contain any individual rule provisions which are not necessary to meet the goals of the rule and; therefore, complies with the NRC's criteria for the treatment of individual requirements.

5.0 Decision Rationale for Selection of the Proposed Action

The NRC staff did not recommend Alternative 1, the no action option. The Commission, in the SRM for SECY-06-0124, approved the rulemaking plan which directed the staff to proceed with preparing a proposed rule. Further, licensees whose plants are projected to exceed the PTS screening limits in the current rule would be required to implement the costly, mandatory compensatory actions with no other alternative available. This option neither satisfies the Commission's SRM direction nor provides regulatory relief for some PWR licensees.

The NRC staff did not recommend Alternative 2. Under this alternative, all current PWR licensees would incur additional regulatory burden from the requirement to re-analyze the plant RT_{PTS} values under the new rule. These designs have previously been licensed or certified under the analysis methods and screening criteria defined in 10 CFR 50.61. As described previously, Alternative 2 would constitute a backfit under 10 CFR 50.109 for these licensees. Further, the NRC determined that the characteristics of advanced PWR designs were not considered in the analysis of this final rule. The NRC cannot be assured that reactors that commence commercial power operation after the effective date of this rule will have operating characteristics and materials of fabrication similar to those evaluated as part of the technical basis for the final rule. The NRC believes that applicants referencing certified designs should not be allowed to use the alternatives provided by 10 CFR 50.61a. Therefore, the NRC determined that it would be prudent to limit the applicability and use of 10 CFR 50.61a to licensees whose construction permits were issued prior to the effective date of the 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.

The NRC staff did not recommend Alternative 3. For the majority of PWR licensees, this alternative would impose no additional regulatory burden to comply with 10 CFR 50.61a because implementation of 10 CFR 50.61a would be voluntary. Although the Commission directed the NRC staff to consider requiring that new reactors be required to comply with the final rule, the NRC staff determined there was no benefit in requiring mandatory implementation for applicants (i.e., non-licensed, non-design certified). This determination was based on the fact that 10 CFR 50.61 is considered conservative and sufficient. As a result, the NRC staff saw no benefit in requiring implementation of 10 CFR 50.61a for any licensee. Further, as stated previously, the NRC determined that the characteristics of PWR reactors that commence

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[3] NUREG/BR-0058, Revision 4, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, September 2004.

commercial power operation after the effective date of the final rule designs were not considered in the analysis of this rule. Therefore, NRC concluded that the use of 10 CFR 50.61a should be limited to licensees whose construction permits were issued prior to the effective date of the final rule and whose reactor vessels were designed and fabricated to the ASME Code, 1998 Edition or earlier.

The NRC staff recommends Alternative 4. This alternative complies with the Commission's SRM that approved the rulemaking plan to prepare a proposed rule. Also, this alternative retains the requirements of the current rule for all PWR licensees, while providing alternative requirements for PWR licensees choosing to implement these requirements. Further, this alternative provides the necessary regulatory flexibility that some current PWR licensees will need to continue to operate throughout their extended lifetimes. Although PWR applicants will not be allowed to use the alternatives provided in the final rule, the current 10 CFR 50.61 is conservative but sufficient, and its requirements do not change as a result of this rulemaking. The final rule, 10 CFR 50.61a, is more realistic yet sufficiently safe, and can be implemented by any current PWR licensee. Therefore, the NRC staff recommends Alternative 4.

6.0 Implementation

This action is being published as a final rule, which would take effect upon Commission approval and publication in the *Federal Register*.

Final OMB Supporting Statement Related to Final Rule:

Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events (10 CFR 50.61 and 50.61a) (3150-0011)

RIN 3150-AI01 [NRC-2007-0008]

DESCRIPTION OF THE INFORMATION COLLECTION

The pressurized thermal shock (PTS) proposed rule was published in the *Federal Register* on October 3, 2007 (72 FR 56275). A supplemental proposed rule was also published in August 11, 2008 (73 FR 46557). The supplemental proposed rule was issued to request stakeholders' feedback on modifications made to the rule language as a result of public comments received on the October 2007 publication. The information collection requirements issued in October 2007 were updated with the information contained in the supporting statement issued for the supplemental proposed rule. The Nuclear Regulatory Commission (NRC) has decided to adopt the PTS final rule. The final rule incorporates the modifications described in the supplemental proposed rule. Therefore, the information collection requirements of this supporting statement remain unchanged from those issued in the supporting statement for the supplemental proposed rule.

PTS events are system transients in a pressurized water reactor (PWR) in which severe overcooling occurs coincident with high pressure. The thermal stresses are caused by rapid cooling of the reactor vessel inside surface combined with the 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 PTS rule, described in Title 10 of the *Code of Federal Regulations* Section 50.61 (10 CFR 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. The screening criteria effectively define a limiting level of embrittlement beyond which operation cannot continue without further plant-specific evaluation. Regulatory Guide (RG) 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Analysis Reports for Pressurized Water Reactors," indicates that reactor vessels that exceed the screening criteria in 10 CFR 50.61 may continue to operate provided they can demonstrate a mean through-wall crack frequency (TWCF) from PTS-related events of no greater than 5×10^{-6} per reactor year.

Any reactor vessel with materials predicted to exceed the screening criteria in 10 CFR 50.61 may not continue to operate without implementation of 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 PTS event probability or severity, and reactor vessel annealing, which are addressed in

10 CFR 50.61(b)(3), (b)(4), and (b)(7); and 10 CFR 50.66, "Requirements for Thermal Annealing of the Reactor Pressure Vessel."

Currently, no operating PWR reactor vessel is projected to exceed the 10 CFR 50.61 screening criteria before the expiration of its 40 year operating license. However, several PWR reactor vessels are approaching the screening criteria, while others are likely to exceed the screening criteria during their first license renewal periods.

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 10 CFR 50.61 are unnecessarily conservative and may impose an unnecessary burden on some licensees. Therefore, the NRC created a new rule, 10 CFR 50.61a, "Alternate Fracture Toughness Requirements for Protection against Thermal Shock Events," which provides 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 because the Commission directed the NRC staff, in a staff requirements memorandum (SRM), "Staff Requirements - SECY-06-0124 - Rulemaking Plan to Amend Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events (10 CFR 50.61)," dated June 30, 2006, to prepare a rulemaking which would allow current PWR licensees to implement the new requirements of 10 CFR 50.61a or continue to comply with the current requirements of 10 CFR 50.61. Alternatively, the NRC could have revised 10 CFR 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 alternate PTS proposed rulemaking for public comment in the *Federal Register* on October 3, 2007 (72 FR 56275). The proposed rule provided an alternative to the current rule in 10 CFR 50.61, which further prompted the NRC to keep the current, mandatory requirements separate from the new requirements. As a result, the proposed rule retained the current requirements in 10 CFR 50.61 for PWR licensees choosing not to implement the less restrictive screening limits, and presented new requirements in 10 CFR 50.61a as a relaxation for PWR licensees.

During the development of the PTS final rule, the NRC determined that several significant changes to the proposed rule language would be needed to adequately address stakeholders' comments, including concerns related to the applicability of the rule. The NRC considered the adoption of these provisions as an alternative to the provisions previously noticed in the *Federal Register*. Because these modifications were significant changes to the proposed rule language on which external stakeholders did not have an opportunity to comment, the NRC concluded that obtaining stakeholder feedback on these provisions through the use of a supplemental proposed rule is appropriate. The supplemental proposed rule was published in August 11, 2008 (73 FR 46557). Consequently, the information collection requirements from the proposed rule were updated in its entirety with the information collection requirements provided in the supporting statement for the supplemental proposed rule.

The NRC has decided to adopt the PTS final rule. The final rule incorporates the modifications described in the supplemental proposed rule. Therefore, the information collection requirements of this supporting statement remain unchanged from those issued in the supporting statement for the supplemental proposed rule.

The technical basis for this rulemaking is documented in the following reports: (1) “Statistical Procedures for Assessing Surveillance Data for 10 CFR Part 50.61a,” (Agencywide Documents Access and Management System (ADAMS) Accession No. ML081290654), (2) “A Physically Based Correlation of Irradiation Induced Transition Temperature Shifts for RPV Steel,” (ADAMS Accession No. ML081000630), (3) 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), (4) NUREG-1874, “Recommended Screening Limits for Pressurized Thermal Shock (PTS)” (ADAMS Accession No. ML070860156), and (5) 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). These reports summarize and reference several additional reports on the same topics.

The updated technical basis indicates that, after 60 years of operation, the risk of reactor vessel failure due to a PTS event is much lower than previously estimated. The updated analyses were based on 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 TWCF results and resultant reference temperature-based screening criteria developed from the analysis of 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 that was characteristic of the three plants that were modeled in detail.

The updated technical basis 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 10 CFR 50.61 and those in the final rule are similar, the form of the revised embrittlement correlation in the final rule differs substantially from the correlation in the existing 10 CFR 50.61. The correlation in the 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.

A. JUSTIFICATION

1. Need for the Collection of Information

Maintaining the structural integrity of the reactor pressure vessel of light-water-cooled reactors is a critical concern related to the safe operation of nuclear power plants. To assure the structural integrity of reactor vessels, the NRC has developed regulations, including 10 CFR 50.61 and 10 CFR 50.61a, and regulatory guides, including Regulatory Guide 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel

Materials," and Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," to provide analysis and measurement methods and procedures to establish that the reactor vessel has adequate safety margin for continued operation. The fracture toughness of the vessel materials varies with time. As the plant operates, neutrons escaping from the reactor core impact the vessel beltline materials causing embrittlement of those materials. The information collection requirements in 10 CFR 50.61 and the final 10 CFR 50.61a, as well as those in 10 CFR 50.60 and 10 CFR Part 50, Appendices G and H, provide estimates of the extent of the embrittlement, and evaluations of the consequences of the embrittlement, in terms of the structural integrity of the vessel. The NRC requires this information to ensure that no reactor, susceptible to the effects of PTS, will continue to operate without the licensee putting in place other mitigating measures.

Specific requirements for reporting and recording in the final rule are identified below.

<u>10 CFR 50.61a(c)</u> requires that each PWR licensee submit a license amendment to request NRC approval to use the requirements of 10 CFR 50.61a three years before exceeding the screening criteria in 10 CFR 50.61. The specific requirements for this amendment are described in paragraphs (c)(1), (c)(2), and (c)(3).

<u>10 CFR 50.61a(c)(1)</u> requires licensees to project the values of RT_{MAX-X} for each reactor vessel beltline material for the expiration date of the operating license fluence of the material. The assessment must (1) use the calculation procedures specified in 10 CFR 50.61a paragraphs (f) and (g); (2) specify the bases for the projected value, including the assumptions regarding core loading patterns; and (3) specify the copper, phosphorus, manganese and nickel contents and the neutron flux and fluence values and full power cold leg temperature used in the calculation for each beltline material.

<u>10 CFR 50.61a(c)(2)</u> requires licensees to assess the flaws in the reactor vessel beltline in accordance with paragraph (e). This assessment is required to be completed at least three years before values of RT_{MAX-X} are projected to exceed the 10 CFR 50.61 screening criteria.

<u>10 CFR 50.61a(c)(3)</u> requires licensees to compare the projected RT_{MAX-X} values with the screening criteria to evaluate the vessel's susceptibility to fracture due to a PTS event.

<u>10 CFR 50.61a(d)(1)</u> requires licensees to submit a re-assessment of RT_{MAX-X} values upon any significant change in the projected values of RT_{MAX-X}, or upon a request for a change in the expiration date for operation of the facility. The specific requirements for this assessment are described in paragraphs (c)(1) and (c)(3).

<u>10 CFR 50.61a(d)(2)</u> requires licensees to submit a re-analysis demonstrating a TWCF of less than $1x10^{-6}$ per reactor year using the assessment requirements in paragraphs (e)(1), (e)(2), and (e)(3). If licensees are required to perform assessments under paragraphs (e)(4), (e)(5), and (e)(6), a report must be submitted to the NRC.

<u>10 CFR 50.61a(d)(3)</u> requires consideration of submission and anticipated approval by the NRC of detailed plant-specific analyses submitted to demonstrate acceptable risk with RT_{MAX-X} above the screening limit due to plant modifications, new information, or

new analysis techniques, in conjunction with implementing flux reduction programs that are reasonably practical to avoid exceeding the screening criteria.

10 CFR 50.61a(d)(4) requires licensees, for which the analysis required by paragraph (d)(3) indicates that no reasonably practical flux reduction program will prevent RT_{MAX-X} from exceeding the screening criteria, to submit a safety analysis to determine what, if any, modifications to equipment, systems, and operation are necessary to prevent potential failure of the reactor vessel as a result of postulated PTS events if continued operation beyond the screening criteria is allowed. This analysis must be submitted at least three years before RT_{MAX-X} is projected to exceed the screening criteria.

10 CFR 50.61a(d)(6) states that if NRC concludes that operation of the facility with RT_{MAX-X} in excess of the screening criteria cannot be approved on the basis of the licensee's analyses submitted under paragraphs (d)(3) and (d)(4), the licensee shall request a license amendment and receive approval by NRC prior to any operation beyond the screening criteria.

10 CFR 50.61a(e) requires licensees to verify that the screening criteria and calculation methodology are applicable to that particular reactor vessel. The analysis to be provided is based on results of the ASME Code volumetric examination. The record-keeping requirements for this verification are described in paragraphs (e)(1), through (e)(4).

10 CFR 50.61a(e)(1) requires licensees to verify that the flaw density and size do not exceed the screening criteria.

10 CFR 50.61a(e)(2) requires licensees to identify flaws that are equal to or greater than 0.075 inches and verify that they do not open to the inside surface of the vessel.

10 CFR 50.61a(e)(3) requires licensees to verify 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.

10 CFR 50.61a(e)(4) requires licensees to perform analyses to demonstrate that the reactor vessel will have a TWCF of less than $1x10^{-6}$ per reactor year.

10 CFR 50.61a(f) requires licensees to calculate RT_{MAX-X} values in accordance with the requirements of paragraphs (f)(1) through (f)(6).

10 CFR 50.61a(f)(4) requires licensees to obtain approval from the NRC if they use a method other than the one specified to calculate $RT_{NDT(U)}$.

10 CFR 50.61a(f)(6)(vi) requires that PWR licensees submit an evaluation of the surveillance data and propose ΔT_{30} and RT_{MAX-X} values if the criteria described in paragraph (f)(6)(v) are not satisfied. These evaluations shall be submitted for NRC approval at the time of the initial application. The licensees shall submit the analysis required by (e)(6) for each surveillance capsule removed from the reactor vessel after the submittal of the initial application for NRC approval.

10 CFR 50.61a(f)(7) requires PWR licensees to report to NRC any information believed to significantly influence the RT_{MAX-X} values. The burden is included in the estimates for RT_{MAX-X} assessment in item 12 of this supporting statement.

Note that this rulemaking makes no changes to the requirements in 10 CFR 50.61, although paragraph (b)(1) of this section is revised to include the option of complying with 10 CFR 50.61a. However, the effect of 10 CFR 50.61a is to shift some of the information collection burden from 10 CFR 50.61 to 10 CFR 50.61a. This shift in burden is discussed in item 12 of this supporting statement.

2. Agency Use of the Information

The information and analyses required by 10 CFR 50.61a will be reported on the plant's docket pursuant to the provisions of 10 CFR 50.4 and reviewed by the NRC to ensure the requirements of the regulation are met. The information collection requirements described above involve a safety issue. By reviewing the submittals from the PWR licensees, the NRC will verify that licensees are aware of (a) the potential threat to the integrity of their reactor vessel from PTS events, and (b) the need to consider additional compensatory measures in order to remain below the screening criterion.

3. Reduction of Burden Through Information Technology

There are no legal obstacles to reducing the burden associated with this information collection. The NRC encourages respondents to use information technology when it would be beneficial to them. NRC issued a regulation on October 10, 2003 (68 FR 58791), consistent with the Government Paperwork Elimination Act, which allows its licensees, vendors, applicants, and members of the public the option to make submissions electronically via CD-ROM, e-mail, special Web-based interface or other means. It is estimated that approximately 90 percent of the potential responses are filed electronically.

4. Effort to Identify Duplication and Use Similar Information

There is no duplication of requirements. NRC has in place an ongoing program to examine all information collections with the goal of eliminating all duplication and/or unnecessary information collections. There are no other NRC or Federal government requirements regarding analyses for flux reduction or plant PTS safety analyses.

5. Effort to Reduce Small Business Burden

The requirements in this rule do not affect small businesses.

6. Consequences to Federal Program or Policy Activities if the Collection is not Conducted or is Conducted Less Frequently

If this information was not collected, the NRC could not verify that each reactor pressure vessel has an adequate safety margin for continued safe operation.

7. Circumstances Which Justify Variations from OMB Guidelines

There are no variations from OMB guidelines in this collection of information.

8. Consultations Outside the NRC

The NRC extended the period for comments on the information collection requirements in November 21, 2007 (72 FR 65470). The NRC received a total of 54 comments for the notices published in October and November 2007.

Opportunity for public comment on the proposed rule was published in the *Federal Register* on October 3, 2007 (72 FR 56275). The NRC published a notice in the *Federal Register* in November 21, 2007 (72 FR 65470) to increase the period for comment on the information collection requirements originally published in October 2007 from 30 days to 75 days. The period for comments for both the proposed rule and the information collection requirements closed on December 17, 2007. The NRC received a total of 54 comments for the notices published in October and November 2007. The NRC also published a supplemental proposed rule on August 11, 2008 (73 FR 46557) and received 5 comments.

For the proposed rule, NRC considered comments submitted by representatives of the nuclear industry including the Nuclear Energy Institute, the Pressurized Water Reactors Owners Group (PWROG), the Electric Power Research Institute (EPRI), the Strategic Teaming and Resource Sharing Alliance and Duke Energy.

The commenters requested that the NRC eliminate from the proposed rule the reporting requirements for embedded flaws violating the sizing criteria because this information is already evaluated and reported to the NRC in the vessel inspection summary reports that are issued to fulfill the requirements of ASME Code, Section XI.

The NRC considered these comments and concluded that the reporting requirements should remain unchanged. The NRC understands that some of the information required to be submitted by the rule may be provided in some, but not all, inservice inspection summary reports to the NRC. For example, the inservice inspection summary report does not necessarily include information about flaw sizes and locations when the flaw sizes are less than the reportable sizes. In addition, the NRC needs to know the size and location of all flaws that exceed the screening criteria in the rule to evaluate the licensee's assessment of the impact of these flaws. If these reporting requirements would be eliminated, the licensee would have to provide to the NRC a flaw-by-flaw reference to the information previously submitted in the inservice inspection summary reports. The NRC has determined that it would require the same level of effort to provide the actual description of each flaw as it would take to provide the flaw-by-flaw reference information. Further, eliminating the time needed for the NRC to search through different summary reports will also increase the efficiency of the NRC evaluation process. Since the rule requires the licensee to provide flaw assessments on vessels that exceed the limits, the additional requirement to identify flaw size and location is a minimal, if not a negligible, additional burden.

In the supplemental proposed rule, the information collection requirements were updated in its entirety. The NRC considered comments submitted by representatives of the nuclear industry including PWROG, EPRI, and FirstEnergy Nuclear Operating Company. There were no significant comments related to the information collection requirements.

9. Payment or Gift to Respondents

Not applicable.

10. Confidentiality of Information

Proprietary or confidential information is protected in accordance with NRC regulations described in 10 CFR 2.390(b) and 10 CFR 9.17(a).

11. Justification for Sensitive Questions

No sensitive information is requested in this rule.

12. Estimated Industry Burden and Burden Hour Cost

Currently Operating Pressurized Water Reactors

The requirements in 10 CFR 50.61a will only apply to those licensees that choose compliance with this section as an alternative to compliance with the requirements specified in 10 CFR 50.61. Of the 69 currently operating PWRs, the NRC projects that eight reactor vessels could exceed the screening criteria specified in 10 CFR 50.61 during their extended lifetimes (i.e., 60 years of operation). The NRC expects that each of these licensees will elect to apply the less stringent embrittlement correlations and screening criteria in 10 CFR 50.61a rather than applying the compensatory measures of 10 CFR 50.61(b)(3) through (b)(7). Because it could take approximately up to 3 years to prepare the package for initial submittal to the NRC, the NRC estimates that only one licensee is expected to apply during the initial 3 year clearance period. The NRC assumes that, 3 years subsequent to the effective date of the final rule, one operating reactor licensee per year will choose to comply with 10 CFR 50.61a for the following eight years. Thus, in the three years following the effective date of this rule, one operating reactor would be affected by the RT_{MAX-X} assessment and inservice testing and none would perform the flux reduction analyses nor the reactor vessel thermal annealing. Therefore, the estimated number of annual respondents is expected to be 0.333 per year and each respondent would provide two responses (i.e., one for the RT_{MAX-X} assessment and one for the analysis of ASME Code inservice ultrasonic testing results).

The NRC staff estimates through their experience that the cumulative burden per licensee complying with 10 CFR 50.61a will be 1,600 hours. In accordance with the information collection estimates currently approved for 10 CFR 50.61; 90 percent of the burden is attributable to record-keeping requirements and 10 percent to the reporting requirements. In accordance with the information provided in the supplemental proposed rule, the NRC estimates that the reporting requirements in 10 CFR 50.61a would be greater than those in 10 CFR 50.61, as the alternate PTS rule would require the licensees to report flaw distributions, inservice inspection data, enhanced surveillance data analysis, and material properties, among others. Therefore, the NRC estimates that from these 1,600 hours, 70 percent would be attributable to record-keeping requirements and 30 percent to the reporting requirements. Additionally, the NRC estimates that from these 1,600 hours, 1,000 hours would be associated to the initial submittal to the NRC and the remainder burden would be associated to subsequent submittals to be made during the remainder of their lifetimes (e.g., 20 years or 30 hours per year).

(1) RT_{MAX-X} assessment - The record-keeping burden is estimated to be approximately 70 percent of the total burden. For RT_{MAX-X} assessment, the record-keeping burden for the initial submittal it is estimated to be 350 staff hours per plant for the three year period. Thus the annualized initial submittal burden over three years would be 1 plant x 350 hours per plant ÷ 3 years, or 116 staff hours per year. An additional 10.5 hours of record-keeping burden per year are required for subsequent submittals to be made during the remainder of the plant's lifetime.

The reporting burden is expected to be approximately 30 percent of the total burden. For RT_{MAX-X} assessment, the reporting burden for the initial submittal is estimated to be 1 plant x 150 hours per plant ÷ 3 years or 50 staff hours per year. An additional 4.5 hours per year are required for reporting required for subsequent submittals to be made during the remainder of the plant's lifetime. Therefore, the total burden is 181 hours (116 + 10.5 + 50 + 4.5 hours). (The burden for RT_{MAX-X} assessments is captured under sections 50.61a(c), (c)(1), (c)(2), (c)(3), (d)(1), (d)(3), (f), (f)(4), (f)(6) and (f)(7).)

(2) Flux reduction analyses - None expected. (The burden for flux reduction analyses is captured under sections 50.61a(d)(3) and(d)(6).)

(3) Safety analysis - None expected. (The burden for safety analysis is captured under sections 50.61a(d)(4) and (d)(6).)

(4) Reactor vessel thermal annealing - None expected. (The burden for reactor vessel thermal annealing is captured under section 50.61a(d)(4).)

(5) Analysis of ASME Code inservice ultrasonic testing results. For the purpose of this supporting statement, the NRC is assuming that the record-keeping and reporting burden for this requirement is the same as for the burden requirements for the RT_{MAX-X} assessment (i.e., 116 hours per year for recordkeeping and 50 hours per year for reporting) for the one current licensee expected to implement the new rule over the next three years. In addition, 10.5 hours for record-keeping and 4.5 hours for reporting are required for any subsequent submittals to be made during the remainder of the plant's lifetime. (The burden for analysis of ASME BPV inservice ultrasonic testing is captured under sections 50.61a(c), (c)(2), (d)(2), (e), (e)(1), (e)(2), (e)(3), and (e)(4).)

The total estimated annual industry burden for record-keeping would be approximately 253 hours or $60,214 (i.e., 253 hours x $238 per hour) per year over the next 3 years. The total estimated annual industry burden for reporting would be approximately 110 hours or $26,180 (i.e., 110 hours x $238 per hour) per year over the next 3 years. Please see Tables 1, 2 and 3 for further clarification.

If licensees elect to apply the less stringent embrittlement correlations and screening criteria in 10 CFR 50.61a, the compensatory measures of 10 CFR 50.61(b)(3) through (b)(7) would not have to be implemented. Therefore, the information collection burden in 10 CFR 50.61 will be reduced.

In the next three years, if one licensee elects to use the alternate screening criteria in 10 CFR 50.61a; the total estimated annual industry burden in 10 CFR 50.61 would be reduced by 840 hours per year, for a net savings of 477 hours per year. In accordance with the information collection estimates for 10 CFR 50.61; 90 percent of the burden is attributable to recordkeeping (i.e., 756 hours), and 10 percent is associated to the reporting requirements (i.e., 84 hours). Hence, in 10 CFR 50.61, the estimated annual recordkeeping burden is reduced to 1,512 hours (i.e., 2,268 - 756), the estimated annual reporting burden is reduced to 168 hours (i.e., 252 - 84), for an annual estimated burden of 1,680 hours (i.e., 2,520 - 840).

New PWRs or PWR applicants

The NRC is limiting the use of 10 CR 50.61a to licensees whose construction permits were issued prior to the effective date of the 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. Therefore, there is no expected burden for new PWRs or PWR applicants.

13. Estimate of Other Additional Costs

The quantity of records to be maintained is roughly proportional to the record-keeping burden and therefore can be used to calculate approximate records storage costs. Based on the number of pages maintained for a typical clearance, the records storage cost has been determined to be 0.0004 times the record-keeping burden cost. Therefore, the storage cost of this clearance is negligible (i.e., 253 record-keeping hours x $238 per hour x 0.0004 = $24).

14. Estimated Annualized Cost to the Federal Government

Licensee submittals will be evaluated by the staff at the estimated cost given below:

(1) RT_{MAX-X} assessment: The staff estimates that reevaluations of RT_{MAX-X} values will be submitted by 1 PWR licensee within the 3-year clearance period. On average, 45 hours are estimated for the review of each submittal. Total review time is estimated at 45 staff hours at an estimated cost of x $10,710 (i.e., 1 submittals x 45 hours per submittal x $238 per hour) over the 3-year clearance period. Thus, the estimated annualized burden is 45 hours at a cost of $3,570.

(2) It is estimated that no licensee will submit an analysis for implementation of a flux reduction program, and thus no staff resources are assumed for this effort.

(3) It is estimated that no licensee will submit an analysis for plant modifications, and thus no staff resources are assumed for this effort.

(4) It is estimated that no licensee will implement reactor vessel thermal annealing, and thus no staff resources are assumed for this effort.

(5) The estimated total annual federal cost, which is the sum of items (1) through (4) above, is $3,570. Please see Table 4 for further clarification.

15. Reasons for Changes in Burden or Cost

There is change in burden that will be incurred by those licensees choosing to implement 10 CFR 50.61a, which includes an additional evaluation of ASME Code inservice volumetric testing results.

If licensees elect to apply the less stringent embrittlement correlations and screening criteria in 10 CFR 50.61a, the compensatory measures of 10 CFR 50.61 would not have to be implemented. Therefore, the information collection burden in 10 CFR 50.61 will be reduced.

The NRC expects that eight licensees will elect to apply the less stringent embrittlement correlations and screening criteria in 10 CFR 50.61a rather than applying the compensatory measures of 10 CFR 50.61(b)(3) through (b)(7). Because it could take approximately up to 3 years to prepare the package for submittal to the NRC, only one licensee is expected to apply during the initial 3 year clearance period. The NRC assumes that, 3 years subsequent to the effective date of the final rule, one operating reactor licensee per year will choose to comply with 10 CFR 50.61a for the following eight years.

In the three years following the effective date of this rule, only one operating reactor is expected to be affected by the RT_{MAX-X} assessment and inservice testing and none would perform the flux reduction analyses nor the reactor vessel thermal annealing.

The expected annual reduction in burden during the next 3 years for licensees implementing the new section 50.61a is expected to be 477 hours (i.e., 840 - 363 hours).

The base burden cost has also changed from $156 to $238 per hour.

16. Publication for Statistical Use

The collected information is not published for statistical purposes.

17. Reason for Not Displaying the Expiration Date

The requirement is contained in a regulation. Amending the Code of Federal Regulations to display information that, in an annual publication, could become obsolete would be unduly burdensome and too difficult to keep current.

18. Exceptions to the Certification Statement

None.

B. COLLECTIONS OF INFORMATION EMPLOYING STATISTICAL METHODS

Not applicable.

TABLE 1a[1]
ANNUAL RECORDKEEPING INFORMATION COLLECTION BURDENS, 10 CFR 50.61a
CURRENTLY OPERATING LICENSEES
RT_{MAX-X} ASSESSMENT

Section	Number of Record-keepers	Burden Hrs Per Record-keeper	Total Hr/Yr
§§ 50.61a(c), (c)(1), (c)(2) and (c)(3)	1	63.25	63.25
§ 50.61a(f)	1	63.25	63.25
§ 50.61a(d)(3)	0	0	0
TOTAL	1		126.5

TABLE 1b[1]
ANNUAL REPORTING INFORMATION COLLECTION BURDENS, 10 CFR 50.61a
CURRENTLY OPERATING LICENSEES
RT_{MAX-X} ASSESSMENT

Section	Number of Respondents	Responses Per Respondent	Total Responses/ yr	Burden Hrs/ Response	Total Burden hr/yr
§ 50.61a(d)(1)	1	0.333	0.333	81.75	27.5
§§ 50.61a(f)(4), (f)(6)(vi) and (f)(7)	1	0.333	0.333	81.75	27.5
§ 50.61a(d)(4)	0	0	0	0	0
§ 50.61a(d)(6)	0	0	0	0	0
TOTALS	1			163.5	55

[1] Burden based on an estimate of 1600 hours per licensee.

TABLE 2a[1]
ANNUAL RECORDKEEPING INFORMATION COLLECTION BURDENS, 10 CFR 50.61a
CURRENTLY OPERATING LICENSEES
ASME CODE INSERVICE ULTRASONIC TESING RESULTS

Section	Number of Record-keepers	Burden Hrs Per Record-keeper	Total Hr/Yr
§§ 50.61a(e), (e)(1), (e)(2), (e)(3) and (e)(4)	1	126.5	126.5
TOTAL	1		126.5

TABLE 2b[2]
ANNUAL REPORTING INFORMATION COLLECTION BURDENS, 10 CFR 50.61a
CURRENTLY OPERATING LICENSEES
ASME CODE INSERVICE ULTRASONIC TESING RESULTS

Section	Number of Respondents	Responses/ Respondent	Total Responses/ yr	Burden Hrs/ response	Total Burden hr/yr
§§ 50.61a(c), (c)(1), and (d)(2)	1	0.333	0.333	163.5	55
TOTALS	1			163.5	55

TABLE 3[2]
TOTAL INFORMATION COLLECTION BURDENS, 10 CFR 50.61a

Section	Number of Respondents	Number of Record-keepers	Total Burden hr/yr
Record-Keeping - RT_{MAX-X}		1	126.5
Record-Keeping - ASME		1	126.5
Reporting	1		55
Reporting	1		55
TOTALS	1	1	363

[2] Burden based on an estimate of 1600 hours per licensee.