[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: Supplemental Proposed Rule.

SUMMARY: The Nuclear Regulatory Commission (NRC) is considering the adoption of provisions regarding applicability of the rule and new provisions regarding procedures to perform surveillance data checks related to the updated fracture toughness requirements for protection against pressurized thermal shock (PTS) events for pressurized water reactor (PWR) pressure vessels. The NRC is considering these provisions as an alternative to the provisions previously noticed for public comment on October 3, 2007 (72 FR 56275).

DATES: Submit comments on this proposed rule by September 10, 2008. Submit comments on the information collection aspects on this proposed rule by September 10, 2008.

ADDRESSES: You may submit comments by any one of the following methods. Please include the following number RIN 3150-AI01 in the subject line of your comments. Comments submitted in writing or in electronic form will be made available for public inspection. Because your comments will not be edited to remove any identifying or contact information, the NRC cautions you against including any information in your submission that you do not want to be publicly disclosed.

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 (301) 415-5905; e-mail Carol.Gallager@nrc.gov.

Mail comments to: Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, ATTN: Rulemakings and Adjudications Staff.

E-mail comments to: Rulemaking.Comments@nrc.gov. If you do not receive a reply e-mail confirming that we have received your comments, contact us directly at (301) 415-1966.

Hand deliver comments to: 11555 Rockville Pike, Rockville, Maryland 20852, between 7:30 am and 4:15 pm during Federal workdays. (Telephone (301) 415-1966).

Fax comments to: Secretary, U.S. Nuclear Regulatory Commission at (301) 415-1101.

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FOR FURTHER INFORMATION CONTACT: Ms. Veronica M. Rodriguez, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; telephone (301) 415-3703; e-mail: Veronica.Rodriguez@nrc.gov, Mr. Barry Elliot, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; telephone (301) 415-2709; e-mail: Barry.Elliot@nrc.gov, or Mr. Mark Kirk, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; telephone (301) 415-6015; e-mail: Mark.Kirk@nrc.gov.

SUPPLEMENTARY INFORMATION:

I. Introduction

The NRC published a proposed rule on alternate fracture toughness requirements for protection against Pressurized Thermal Shock (PTS) for public comments in the *Federal Register* on October 3, 2007 (72 FR 56275). This rule provides new 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 would reduce regulatory burden for licensees, specifically those licensees that expect to exceed the existing requirements before the expiration of their licenses, while maintaining adequate safety. These new requirements would be utilized by any Pressurized Water Reactor (PWR) licensee as an alternative to complying with the existing requirements.

During the development of the PTS final rule, the NRC determined that several changes to the proposed rule language may be needed to adequately address issues raised in stakeholder's comments. The NRC also determined, in response to a stakeholder comment, that the characteristics of advanced PWR designs were not considered in the technical analysis made for the proposed rule. The NRC does not have assurance 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 proposed rule. Therefore, the NRC has concluded that it would be prudent to limit the applicability and the use of § 50.61a to currently-operating plants only, and proposes to modify the applicability provisions of the proposed rule accordingly.

Also, several stakeholders questioned the accuracy and validity of the generic embrittlement curves in the proposed rule. The NRC wants to ensure that the predicted values from the proposed embrittlement trend curves provide an adequate basis for implementation of the rule. Therefore, 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 is considering enhancing the procedure described in paragraph § 50.61a(f)(6) to, among other things, detect signs from the plant- and heat-specific surveillance data of embrittlement trends that are not reflected by Equations 5, 6 and 7 of the rule that may emerge at high fluences.

Because these proposed modifications may not represent a logical outgrowth from the October 2007 proposed rule's provisions, the NRC concludes that obtaining stakeholder feedback on the proposed alternative provisions through the use of a supplemental proposed rule is appropriate. As discussed in Section VI of this notice, the NRC will consider comments on §§ 50.61a(b); (f)(6)(i) through (f)(6)(vi); Equations 10, 11, and 12 in § 50.61a(g); and Tables 5, 6, and 7 of this supplemental proposed rule. The NRC is also requesting comments on whether there should be additional language added to § 50.61a(e) to allow licensees to account for the effects of sizing errors. This supplemental proposed rule does not reflect other modifications or editorial and conforming changes that the NRC is considering to incorporate in the final rule as a result of the public comments on the October 2007 proposed rule.

II. Background

PTS events are system transients in a PWR in which severe overcooling occurs coincident with high pressure. The thermal stresses are caused by rapid cooling of the reactor vessel inside surface, which combine 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 § 50.61, 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 § 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 $5x10^{-6}$ per reactor year.

Any reactor vessel with materials predicted to exceed the screening criteria in § 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 §§ 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 reactor vessel is projected to exceed the § 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 § 50.61 are unnecessarily conservative and may impose an unnecessary burden on some licensees. Therefore, the NRC created a new rule, § 50.61a, 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) dated June 30, 2006, to prepare a rulemaking which would allow current PWR licensees to implement the new requirements of § 50.61a or continue to comply with the current requirements of § 50.61. Alternatively, 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 rulemaking on the alternate fracture toughness requirements for protection against PTS for public comment in the *Federal Register* on October 3, 2007 (72 FR 56275). The proposed rule provided an alternative to the current rule, which a licensee may choose to adopt. This prompted the NRC to keep the current requirements separate from the new alternative requirements. As a result, the proposed rule retained the current requirements in § 50.61 for PWR licensees choosing not to implement the less restrictive screening limits, and presented new requirements in § 50.61a as an alternative relaxation for PWR licensees.

III. Discussion

The NRC published a proposed new rule, § 50.61a (October 3, 2007, 72 FR 56275) that would provide new PTS requirements based on updated analysis methods because the existing requirements are based on unnecessarily conservative probabilistic fracture mechanics analyses. Stakeholders' comments raised concerns related to the applicability of the rule and the accuracy and validity of the generic embrittlement curves. The NRC reconsidered the technical and regulatory issues in these areas and is considering adopting the modified provisions regarding the applicability of the rule and new provisions regarding procedures to perform surveillance data checks described in this supplemental proposed rule. The NRC will consider comments on §§ 50.61a(b), (f)(6)(i) through (f)(6)(vi); Equations 10, 11 and 12 in § 50.61a(g); and Tables 5, 6, and 7 of this supplemental proposed rule. As described in Section VI of this notice, the NRC is also requesting comments on whether there should be additional language added to § 50.61a(e) to allow licensees to account for the effects of sizing errors. The NRC will consider the October 2007 proposed rule, the supplemental proposed rule, and the comments received in response to both, when deciding whether to adopt a final PTS rule.

Applicability of the Proposed Rule, § 50.61a(b):

The supplemental proposed rule differs from the proposed rule and from § 50.61 in that it proposes to limit the use of § 50.61a to currently-operating plants only. It cannot be demonstrated, *a priori*, that reactors which commence commercial power operation after the effective date of this rule will have operating characteristics, in particular identified PTS event sequences and thermal-hydraulic responses, which are consistent with the reactors which were evaluated as part of the technical basis for this rule. Other factors, including materials of fabrication and welding methods, could also vary. Hence, the use of § 50.61a would be limited to currently operating PWR facilities which are known to have characteristics consistent with those assumed in the technical basis. The NRC also proposes to allow the holder of the operating license for Watts Bar Unit 2 to adopt the requirements in § 50.61a as this facility has operating characteristics consistent to those assumed in the technical basis. The NRC recognizes that licensees for reactors who commence commercial power operation after the effective date of this rule 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 operating characteristics, materials of fabrications, and welding methods is provided.

Surveillance Data, § 50.61a(f):

Section 50.61a(f), of the proposed rule defines the process for calculating the values for the material properties (i.e., RT_{MAX-X}) for a particular reactor vessel. These values would 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}).

Section 50.61a(f), of the proposed rule included 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 normally be the case. 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 trends, and to estimate RT_{MAX-X}, for the conditions being assessed. However, alternative methods for temperature shift prediction are not prescribed by § 50.61a(f) of the proposed rule.

Although standard and accepted procedures exist to assess the *statistical* significance of the differences between heat-specific surveillance data and the generic embrittlement trend curve, similarly standard and acceptable procedures are not available to assess the *practical* importance of such differences. The practical importance of statistically significant deviations is best assessed by licensees on a case-by-case basis, which would be submitted for the review of the Director of NRR, as prescribed by § 50.61a(f).

The method described in the proposed rulemaking to compare the heat-specific surveillance data collected as part of 10 CFR Part 50, Appendix H surveillance programs to the generic temperature shift embrittlement trend curve included a single statistical test. This statistical test was set forth by Equations 9 and 10, and Table 5. This test determined if, on average, the temperature shift from the surveillance data was significantly higher than the temperature shift of the generic embrittlement trend curve. The NRC has determined that, while necessary, this single test is not sufficient to ensure that the temperature shift predicted by the embrittlement trend curve well represents 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. To address this potential deficiency, which could be particularly important during a plant's period of extended operation, the NRC added two more statistical tests in this supplemental proposed rulemaking, which are expressed by Equations 11 and 12 and by Tables 6 and 7. Together, these two additional tests determine if the surveillance data from a particular heat show a more rapid increase after significant radiation exposure than the progression predicted by the generic embrittlement trend curve.

The NRC documented the technical basis for proposed alternative in the following reports: (1) "Statistical Procedures for Assessing Surveillance Data for 10 CFR Part 50.61a," (ADAMS Accession No. ML081290654), and (2) "A Physically Based Correlation of Irradiation Induced Transition Temperature Shifts for RPV Steel," (ADAMS Accession No. ML081000630).

IV. Responses to Comments on the Proposed Rule

The NRC received 5 comment letters on the proposed 10 CFR 50.61a rule published on October 3, 2007 (72 FR 56275). The following paragraphs discuss those comments which are directly associated with the supplemental proposed rule's provisions on the applicability of the rule and surveillance data procedures. The remainder of the comments and the NRC responses will be provided in the *Federal Register* notice for the final rule.

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 to state that this rule is applicable to the current PWR fleet and not the new plant designs. [PWROG-5, EPRI-5]

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 reactors that commence commercial power operation after the effective date of this rule will have operating characteristics, in particular identified PTS event sequences and thermal-hydraulic responses, which are 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, could also vary. Therefore, the NRC agrees with the commenters that it would be prudent to restrict the use of § 50.61a to current plants. As a result of this comment, the NRC proposes to modify § 50.61a(b) and the statement of considerations of the rule to reflect this position to limit the use of the rule to currently operating plants.

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 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.

The commenters suggested removing the requirement to assess surveillance data, including Table 5, of this rule. [PWROG-4, EPRI-4, NEI-2]

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. 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 is considering further enhancing the procedure described in paragraph (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.

V. Section-by-Section Analysis

The following section-by-section analysis only discusses the modifications in the provisions related to the applicability of the rule and surveillance data procedures that the NRC is considering as an alternative in this supplemental proposed rule. The NRC is only seeking comments on these alternative provisions. This supplemental proposed rule does not reflect other modifications or editorial and conforming changes that the NRC is considering to incorporate as a result of the public comments on the proposed rule that were not discussed in this notice as they will be provided in the *Federal Register* notice for the final rule.

Proposed § 50.61a(b)

The proposed language for § 50.61a(b) would establish the applicability of the rule. The NRC proposes to modify this paragraph to limit the use of this rule to currently-operating plants only.

Proposed § 50.61a(f)(6)(i)

The proposed language for § 50.61a(f)(6)(i) would establish 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. The NRC proposes to modify § 50.61a(f)(6)(i)(B) to state that licensees would 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) of the supplemental proposed rule.

Proposed § 50.61a(f)(6)(ii)

The proposed language for § 50.61a(f)(6)(ii) would establish the requirements to perform an estimate of the mean deviation of the data set from the embrittlement model. The mean deviation for the data set would be compared to values given in Table 5 or Equation 10 of this section. The NRC proposes to modify this paragraph to state that the surveillance data analysis would follow the criteria in §§ 50.61a(f)(6)(v) and (f)(6)(vi) of the supplemental proposed rule.

Proposed § 50.61a(f)(6)(iii)

The NRC proposes to modify § 50.61a(f)(6)(iii) to establish 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 would estimate the slope using Equation 11 and compare this value to the maximum permissible value in Table 6, both from the supplemental proposed rule. This surveillance data analysis would follow the criteria in §§ 50.61a(f)(6)(v) and (f)(6)(vi) of the supplemental proposed rule.

Proposed § 50.61a(f)(6)(iv)

The NRC proposes to modify § 50.61a(f)(6)(iv) to establish the requirements to estimate an outlier deviation from the embrittlement model for the specific data set using Equations 8 and 12. The licensee would compare the normalized residuals to the allowable values in Table 7 of the supplemental proposed rule. This surveillance data analysis would follow the criteria in §§ 50.61a(f)(6)(v) and (f)(6)(vi) of the supplemental proposed rule.

Proposed § 50.61a(f)(6)(v)

The NRC proposes to add paragraph (f)(6)(v) to establish the criteria to be satisfied in order to calculate the ΔT_{30} shift values.

Proposed § 50.61a(f)(6)(vi)

The NRC proposes to add paragraph (f)(6)(vi) to establish the actions to be taken by a licensee if the criteria in paragraph (f)(6)(v) of this section are not met. The licensee would need to submit an evaluation of the surveillance data and propose values for ΔT_{30}, considering their plant-specific surveillance data, for the review and approval by the Director. The licensee would need to 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 no later than 2 years after the capsule is withdrawn from the vessel.

Proposed § 50.61a(g)

The proposed language for § 50.61a(g) would provide the necessary equations and variables required by the proposed changes in § 50.61a(f)(6). The NRC proposes to modify Equation 10 to account for 1 percent of significance level. Equations 11 and 12 would be added to provide the means for estimating the slope and the outlier deviation from the embrittlement model.

Proposed Tables 5, 6, and 7

Tables 5, 6, and 7 would provide values to be used in the proposed changes in § 50.61a(f)(6). The NRC proposes to modify Table 5 to account for the use of a 1 percent of significance level. Tables 6 and 7 would be added to provide the threshold values for the slope and the outlier deviation tests.

VI. Specific Request for Comments

The NRC seeks comments on §§ 50.61a(b), (f)(6)(i) through (f)(6)(vi); Equations 10, 11, and 12 in § 50.61a(g), and Tables 5, 6, and 7 of the supplemental proposed rule. The NRC is not seeking comments on any other provisions of the proposed § 50.61a which remain unchanged from the October 2007 proposed rule. In addition, the NRC also requests comments on the following question:

Adjustments of the Inservice Inspection Volumetric Examination and Flaw Assessments:

The flaw sizes in Tables 2 and 3 are selected so that reactor vessels with flaw sizes less than or equal to those in the tables will have a TWCF less than or equal to 1×10^{-6} per reactor year at the maximum permissible embrittlement. The NRC recognizes that the flaw sizes in these tables 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. Therefore, NRC is considering to permit flaw sizes to be adjusted to account for the effects of sizing error before comparing the estimated size and density distribution to the acceptable size and density distributions in Tables 2 and 3. This would be accomplished by requiring licensees to base the methodology to account for the effects of sizing error on statistical data collected from ASME Code inspector qualification tests. An acceptable method would include a demonstration, that accounting for the effects of sizing error, is unlikely to result in accepting actual flaw size distribution that cause the TWCF to exceed the acceptance criteria. Adjusting flaw sizes to account for sizing error can change an unacceptable examination result into an acceptable result; further, collecting, evaluating, and using data from ASME Code inspector qualification tests will require extensive engineering judgment. Therefore, the methodology would have to be reviewed and approved by the Director of the NRC's Office of Nuclear Reactor Regulation (NRR) to ensure that the risk associated with PTS is acceptable. The NRC requests specific comments on whether there should be additional language added to 10 CFR 50.61a(e) to allow licensees to account for the effects of sizing errors.

VII. Availability of Documents

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

Public Document Room (PDR). The NRC Public Document Room 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

Document	PDR	Web	**ERR** (ADAMS)
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
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
“Statistical Procedures for Assessing Surveillance Data for 10 CFR Part 50.61a”	x		ML081290654
“A Physically Based Correlation of Irradiation Induced Transition Temperature Shifts for RPV Steel”	x		ML081000630
Supplemental Regulatory Analysis	x	NRC-2007-0008	ML081440673
Supplemental OMB Supporting Statement	x	NRC-2007-0008	ML081440736

Document	**PDR**	**Web**	**ERR** (ADAMS)
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

VIII. Plain Language

The Presidential memorandum “Plain Language in Government Writing” published in June 10, 1998 (63 FR 31883), directed that the Government’s documents be in clear and accessible language. The NRC requests comments on the proposed rule specifically with respect to the clarity and effectiveness of the language used. Comments should be sent to the NRC as explained in the **ADDRESSES** heading of this notice.

IX. 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 utilized 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 supplemental proposed 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 supplemental proposed rule because it does not represent the broad range of conditions necessary to justify a revision to the regulations.

The ASME Code requirements are utilized as part of the volumetric examination analysis requirements of the supplemental proposed 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 utilized to determine 30-foot-pound transition temperatures. These standards were selected for use in the supplemental proposed rule based on their use in other regulations within 10 CFR Part 50 and their applicability to the subject of the desired requirements.

The NRC will consider using a voluntary consensus standard in the final rule if an appropriate standard is identified in the public comment period for this supplemental proposed rule.

X. 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, if adopted, would not be 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. This determination was made as part of the proposed rulemaking issued on October 3, 2007 (72 FR 56275), and remains applicable to this supplemental proposed rulemaking.

XI. Paperwork Reduction Act Statement

This supplemental proposed rule would contain new or amended information collection requirements that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501, *et seq*). This supplemental proposed rule has been submitted to the Office of Management and Budget for review and approval of the information collection requirements.

Type of submission, new or revision: Revision

The title of the information collection: 10 CFR Part 50, "Alternate Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events (10 CFR 50.61 and 50.61a)" supplemental proposed rule.

The form number if applicable: Not applicable

How often the collection is required: Collections would be initially required for PWR licensees utilizing the requirements of 10 CFR 50.61a as an alternative to the requirements of 10 CFR 50.61. Collections would also be required, after implementation of the new 10 CFR 50.61a, when any change is made to the design or operation of the facility that affects the calculated $RT_{MAX\text{-}X}$ value. Collections would also be required during the scheduled periodic ultrasonic examination of beltline welds.

Who will be required or asked to report: Licensees of currently operating PWRs utilizing the requirements of 10 CFR 50.61a in lieu of the requirements of 10 CFR 50.61 would be subject to all of the proposed requirements in this rulemaking.

An estimate of the number of annual responses: 2

The estimated number of annual respondents: 1

An estimate of the total number of hours needed annually to complete the requirement or request: 363 hours (253 hours annually for record keeping plus 110 hours annually for reporting)

Abstract: The NRC is proposing to amend its regulations to provide updated fracture toughness requirements for protection against PTS events for PWR pressure vessels. The supplemental proposed rule would provide new PTS requirements based on updated analysis methods. This action is necessary because the existing requirements are based on unnecessarily conservative probabilistic fracture mechanics analyses. This action is expected to reduce regulatory burden for licensees, specifically those licensees that expect to exceed the existing requirements before the expiration of their licenses. These new requirements would be utilized by licensees of currently operating PWRs as an alternative to complying with the existing requirements.

The NRC is seeking public comment on the potential impact of the information collections contained in this supplemental proposed rule and on the following issues:

1. Is the proposed information collection necessary for the proper performance of the functions of the NRC, including whether the information will have practical utility?
2. Estimate of burden?
3. Is there a way to enhance the quality, utility, and clarity of the information to be collected?
4. How can the burden of the information collection be minimized, including the use of automated collection techniques?

A copy of the OMB clearance package may be viewed free of charge at the NRC Public Document Room, One White Flint North, 11555 Rockville Pike, Room O-1F21, Rockville, MD 20852. The OMB clearance package and rule are available at the NRC worldwide Web site: http://www.nrc.gov/public-involve/doc-comment/omb/index.html. The document will be available on the NRC home page site for 60 days after the signature date of this notice.

Send comments on any aspect of these proposed information collections, including suggestions for reducing the burden and on the above issues, by September 10, 2008. Comments received after this date will be considered if it is practical to do so, but assurance of consideration cannot be given to comments received after this date. Comments submitted in writing or in electronic form will be made available for public inspection. Because your comments will not be edited to remove any identifying or contact information, the NRC cautions you against including any information in your submission that you do not want to be publicly disclosed. Comments submitted should reference Docket No. NRC-2007-0008. Comments can be submitted in electronic form via the Federal e-Rulemaking Portal at http://www.regulations.gov by search for Docket No. NRC-2007-0008. Comments can be mailed to NRC Clearance Officer, Russell Nichols (T-5F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001. Questions about the information collection requirements may be directed to the NRC Clearance Officer, Russell Nichols (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, by telephone at (301) 415-6874, or by email to INFOCOLLECTS.Resource@nrc.gov. Comments can be mailed 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, or by telephone at (202) 395-7345.

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.

XII. Regulatory Analysis

The NRC has issued a supplemental regulatory analysis for this supplemental proposed rulemaking. The analysis examines the costs and benefits of the alternatives considered by the NRC. The NRC requests public comments on this supplemental draft regulatory analysis. Availability of the supplemental regulatory analysis is provided in Section VII of this notice. Comments on the supplemental draft regulatory analysis may be submitted to the NRC as indicated under the **ADDRESSES** heading of this notice.

XIII. 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, if promulgated, have a significant economic impact on a substantial number of small entities. This supplemental proposed 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).

XIV. Backfit Analysis

The NRC has determined that the requirements in this supplemental proposed rule would not constitute backfitting as defined in 10 CFR 50.109(a)(1). Therefore, a backfit analysis has not been prepared for this proposed 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 supplemental proposed rule. The requirements of the proposed PTS rule, including those in the supplemental proposed rule, would not be required, but could be utilized by PWR licensees with currently operating plants.

Licensees choosing to implement the proposed PTS rule would be required to comply with its requirements as an alternative to complying with the requirements of the current PTS rule. Because the proposed PTS rule would not be mandatory for any PWR licensee, but rather could be voluntarily implemented, the NRC finds that this amendment would not constitute backfitting.

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. 553; the NRC is proposing to adopt 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); sec. 651(e), Pub. L. 109-58, 119 Stat. 806-810 (42 U.S.C. 2014, 2021, 2021b, 2111).

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

(2) RT_{MAX-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-AW} is determined under the provisions of paragraph (f) of this section and has units of °F.

(3) RT_{MAX-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-PL} is determined under the provisions of paragraph (f) of this section and has units of °F.

(4) RT_{MAX-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-FO} is determined under the provisions of paragraph (f) of this section and has units of °F.

(5) RT_{MAX-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-CW} is determined under the provisions of paragraph (f) of this section and has units of °F.

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

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

(8) *φ* means average neutron flux. φ is determined 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 produced by irradiation defined at the 30 ft-lb energy level. The ΔT_{30} value is determined 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, data from test reactors or 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.

(b) *Applicability*. Each licensee of a pressurized water nuclear power reactor, whose original operating license was issued prior to [EFFECTIVE DATE OF FINAL RULE], and the holder of any operating license issued under this part or part 54 for the Watts Bar Unit 2 facility, may utilize the requirements of this section as an alternative to the requirements of 10 CFR 50.61.

(c) *Request for Approval*. Prior to implementation of this section, each licensee shall submit a request for approval in the form of a license amendment together with the documentation required by paragraphs (c)(1), (c)(2), and (c)(3) of this section for review and approval to the Director, Office of Nuclear Reactor Regulation (Director). The information required by paragraphs (c)(1), (c)(2), and (c)(3) of this section 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, except as provided in paragraphs (f)(6) and (f)(7) 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, etc.); 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.

(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)(1) through (e)(3) have been met and submit all documented indications and the neutron fluence map required by paragraph (e)(1)(iii) to the Director in its application to utilize 10 CFR 50.61a. If analyses performed under paragraph (e)(4) of this section are used to justify continued operation of the facility, approval by the Director is required prior to implementation.

(3) Each licensee shall compare the projected RT_{MAX-X} values for plates, forgings, axial welds, and circumferential welds to the PTS screening criteria 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 utilize 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}, such that the previous value, the current value, or both values, exceed the screening criteria prior to 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 for review and approval to the Director. 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, prior to operation beyond the PTS screening criteria in Table 1 of this section.

(2) Licensees shall determine the impact of the subsequent flaw assessments required by paragraphs (e)(1)(i), (e)(1)(ii), (e)(2), and (e)(3) of this section and shall submit the assessment for review and approval to the Director within 120 days after completing a volumetric examination of reactor vessel beltline materials as required by Section XI of the ASME Code. If a licensee is required to implement paragraphs (e)(4) and (e)(5) of this section, a re-analysis in accordance with paragraphs (e)(4) and (e)(5) of this section is required within one year of the subsequent ASME Code inspection.

(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 if continued operation beyond the PTS screening criteria is to be allowed. 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 must be submitted to the Director 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.

(6) If the Director concludes, under paragraph (d)(5) of this section, that operation of the facility with RT_{MAX-X} values in excess of the PTS screening criteria cannot be approved on the basis of the licensee's analyses submitted under paragraphs (d)(3) and (d)(4) of this section, then the licensee shall request a license amendment, and receive approval by the Director, prior to 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.

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

(e) *Examination and Flaw Assessment Requirements*. The volumetric examinations 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.

(1) The licensee shall verify that the indication density and size distributions within the ASME Code, Section XI, Appendix VIII, Supplement 4 inspection volume[1] are within the flaw density and size distributions in Tables 2 and 3 of this section based on the test results from the volumetric examination. The allowable number of flaws specified in Tables 2 and 3 of this section represent a cumulative flaw size distribution for each ASME flaw size increment. The allowable number of flaws for a particular ASME flaw size increment represents the maximum total number of flaws in that and all larger ASME flaw size increments. The licensee shall also demonstrate that no flaw exceeds the size limitations specified in Tables 2 and 3 of this section.

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

(iii) For each indication detected in the ASME Code, Section XI, Appendix VIII, Supplement 4 inspection volume, the licensee shall document the dimensions of the indication, including depth and length, the orientation of the indication relative to the axial direction, and the

[1] The ASME Code, Section XI, Appendix VIII, Supplement 4 weld volume is the weld volume from the clad-to-base metal interface to the inner 1.0 inch or 10 percent of the vessel thickness, whichever is greater.

location within the reactor vessel, including its azimuthal and axial positions and its depth embedded from the clad-to-base metal interface. The licensee shall also document 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 indications.

(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 indications within the ASME Code, Section XI, Appendix VIII, Supplement 4 inspection volume that are located at the clad-to-base metal interface. The licensee shall verify that such indications do not open to the vessel inside surface using a qualified surface or visual examination.

(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, all indications 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 through-wall crack frequency (TWCF) of less than 1×10^{-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 indication density and size in the ASME Code, Section XI, Appendix VIII, Supplement 4 inspection volume is not within the flaw density and size limitations specified in Tables 2 and 3 of this section;

(ii) Any indication in the ASME Code, Section XI, Appendix VIII, Supplement 4 inspection volume that is larger[2] than the sizes in Tables 2 and 3 of this section;

(iii) There are linear indications that penetrate through the clad into the low alloy steel reactor vessel shell; or

(iv) Any indications between the clad-to-base metal interface and three-eights[3] 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 indications 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.

(f) *Calculation of RT_{MAX-X} values*. Each licensee shall calculate RT_{MAX-X} values for each reactor vessel beltline material using φt. φt must be calculated using an NRC-approved methodology.

(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.

(2) The values of ΔT_{30} must be determined using Equations 5 through 7 of this section, unless the conditions specified in paragraph (f)(6)(vi) of this section are met, for each axial weld fusion line, plate, and circumferential weld fusion line. The ΔT_{30} value for each axial weld fusion line calculated as specified by Equation 1 of this section must be calculated for the maximum fluence (φt_{FL}) occurring along a particular axial weld fusion line. The ΔT_{30} value for each plate

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[2] Table 2 for the weld flaws is limited to flaw sizes that are expected to occur and were modeled from the technical basis supporting this rule. Similarly, Table 3 for the plate and forging flaws stops at the maximum flaw size modeled for these materials in the technical basis supporting this rule.

[3] 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.

calculated as specified by Equation 1 of this section must be calculated for φt_{FL} occurring along a particular axial weld fusion line. The ΔT_{30} value for each plate or forging calculated as specified by Equations 2 and 3 of this section are calculated for the maximum fluence (φt_{MAX}) occurring at the clad-to-base metal interface of each plate or forging. In Equation 4, the φt_{FL} value used for calculating the plate, forging, and circumferential weld $RT_{MAX\text{-}CW}$ value is the maximum φt occurring for each material along the circumferential weld fusion line.

(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 weight percentages. 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 specification to which the vessel material was fabricated, or conservative estimates (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 $RT_{MAX\text{-}X}$ 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.

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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 period 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.

(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 weighted time 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. The licensee shall consider plant-specific information that could affect the use of Equations 5 though 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. The surveillance data analysis must use the criteria in paragraphs (f)(6)(v) and (f)(6)(vi) of this section. For surveillance data sets with greater than 8 shift 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. The surveillance data analysis must follow the criteria in paragraphs (f)(6)(v) and (f)(6)(vi) of this section. For surveillance data sets with greater than 15 shift 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. The surveillance data analysis must follow the criteria in paragraphs (f)(6)(v) and (f)(6)(vi) of this section.

(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.

(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 4, 5, and 6 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 and shall, on the basis of this review, 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 shall be submitted for the review and approval by the Director at the time of the initial application. For each surveillance capsule removed from the reactor vessel after the submittal of the initial application, the licensee shall perform the analyses required by paragraph (f)(6) of this section. The analyses must be submitted for the review and approval by the Director in the form of a license amendment, and must be submitted no later than two years after the capsule is withdrawn from the vessel.

(7) The licensee shall report any information that significantly improves the accuracy of the RT_{MAX-X} value to the Director. Any value of RT_{MAX-X} that has been modified as specified in paragraph (f)(6)(iv) of this section is subject to the approval of the Director when used as provided in this section.

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

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

$[RT_{NDT(u)\text{ - axial weld}} + \Delta T_{30\text{ - axialweld}} (\varphi t_{FL})]\}$

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

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

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

$[RT_{NDT(u)\text{ - circweld}} + \Delta T_{30\text{ - circweld}} (\varphi t_{MAX})],$

$[RT_{NDT(u)\text{ - forging}} + \Delta T_{30\text{ - forging}} (\varphi t_{MAX})]\}$

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

φte = 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] = φ x 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 shift 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 = 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.

$se(m)$ = 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}	356	305	293
Axial Weld and Plate, RT_{MAX-AW} + RT_{MAX-PL}	538	476	445
Circumferential Weld, RT_{MAX-CW}[7]	312	277	269
Forging with underclad cracks, RT_{MAX-FO}	246	241	239

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

[7] RT_{PTS} limits contributes 1×10^{-8} per reactor year to the reactor vessel TWCF.

Table 2 - Allowable Number of Flaws in Welds

ASME Section XI Flaw Size per IWA-3200	Range of Through-wall Extent (TWE) of Flaw [in.]	Allowable Number of Cumulative Flaws per 1000 Inches of Weld Length in the ASME Section XI Appendix VIII Supplement 4 Inspection Volume
0.05	0.025 ≤ TWE < 0.075	Unlimited
0.10	0.075 ≤ TWE < 0.125	166.70
0.15	0.125 ≤ TWE < 0.175	90.80
0.20	0.175 ≤ TWE < 0.225	22.82
0.25	0.225 ≤ TWE < 0.275	8.66
0.30	0.275 ≤ TWE < 0.325	4.01
0.35	0.325 ≤ TWE < 0.375	3.01
0.40	0.375 ≤ TWE < 0.425	1.49
0.45	0.425 ≤ TWE < 0.475	1.00

Table 3 - Allowable Number of Flaws in Plates or Forging

ASME Section XI Flaw Size per IWA-3200	Range of Through-wall Extent (TWE) of Flaw [in.]	Allowable Number of Cumulative Flaws per 1000 Square Inches of Inside Diameter Surface Area in Forgings or Plates in the ASME Section XI Appendix VIII Supplement 4 Inspection Volume[8]
0.05	0.025 ≤ TWE < 0.075	Unlimited
0.10	0.075 ≤ TWE < 0.125	8.049
0.15	0.125 ≤ TWE < 0.175	3.146
0.20	0.175 ≤ TWE < 0.225	0.853
0.25	0.225 ≤ TWE < 0.275	0.293
0.30	0.275 ≤ TWE < 0.325	0.0756
0.35	0.325 ≤ TWE < 0.375	0.0144

[8] Excluding underclad cracks in forgings.

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 24th day of July 2008.

For the Nuclear Regulatory Commission.

/RA/

R. W. Borchardt,
Executive Director for Operations.

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 24th day of July 2008.

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Executive Director for Operations.

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