Enclosure 1

Proposed Rule on Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events

[7590-01-P]

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

RIN 3150-AI01

Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events

AGENCY: Nuclear Regulatory Commission.

ACTION: Proposed rule.

SUMMARY: The Nuclear Regulatory Commission (NRC) is proposing to amend its regulations to provide updated fracture toughness requirements for protection against pressurized thermal shock (PTS) events for pressurized water reactor (PWR) pressure vessels. The proposed rule would provide 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 voluntarily utilized by any PWR licensee as an alternative to complying with the existing requirements.

DATES: Submit comments by [INSERT DATE 75 DAYS AFTER PUBLICATION IN THE FEDERAL REGISTER]. Submit comments specific to the information collection aspects of this rule by [INSERT DATE 30 DAYS AFTER PUBLICATION IN THE FEDERAL REGISTER].

Comments received after these dates will be considered if it is practical to do so, but assurance of consideration cannot be given to comments received after these dates.

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 on rulemakings submitted in writing or in electronic form will be made available for public inspection. Because your comment 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.

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

E-mail comments to: SECY@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. You may also submit comments via the NRC's rulemaking website at http://ruleforum.llnl.gov. Address questions about our rulemaking web site to Carol Gallagher (301) 415-5905; Email CAG@nrc.gov. Comments can also be submitted via the Federal e-Rulemaking Portal http://www.regulations.gov.

Hand deliver comments to: 11555 Rockville Pike, Rockville, Maryland 20852, between 7:30 a.m. and 4:15 p.m. Federal workdays (telephone (301) 415-1966).

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

You may submit comments on the information collections by the methods indicated in the Paperwork Reduction Act Statement.

Publicly available documents related to this rulemaking may be viewed electronically on the public computers located at the NRC's Public Document Room (PDR), O1-F21, One White Flint North, 11555 Rockville Pike, Rockville, MD 20852-2738. The PDR reproduction contractor will copy documents for a fee. Selected documents, including comments, may be viewed and downloaded electronically via the NRC rulemaking web site at http://ruleforum.llnl.gov.

Publicly available documents created or received at the NRC after November 1, 1999, are available electronically at the NRC's Electronic Reading Room at http://www.nrc.gov/reading-rm/adams.html. From this site, the public can gain entry into the NRC's Agencywide Document Access and Management System (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 PDR Reference staff at 1-800-397-4209, 301-415-4737, or by e-mail to PDR@nrc.gov.

FOR FURTHER INFORMATION CONTACT: Mr. George Tartal, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; telephone (301) 415-0016; e-mail: GMT1@nrc.gov, or Mr. Barry Elliot, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; telephone (301) 415-2709; e-mail: BJE@nrc.gov.

SUPPLEMENTARY INFORMATION

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I. Background

Pressurized thermal shock events are system transients in a pressurized water reactor (PWR) in which severe overcooling occurs coincident with high pressure. The thermal stresses caused by rapid cooling of the reactor vessel inside surface combine with the 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 adjacent to the core where neutron radiation gradually embrittles the material over the lifetime of the plant may be such a material.

The toughness of ferritic reactor vessel materials is characterized by a "reference temperature for nil ductility transition" (RT_{NDT}). RT_{NDT} is referred to as a ductile-to-brittle transition temperature. At temperatures below RT_{NDT} fracture occurs very rapidly, by cleavage, a behavior referred to as "brittle." As temperatures increase above RT_{NDT}, progressively larger amounts of deformation occur before rapid cleavage fracture occurs. Eventually, at temperatures above approximately RT_{NDT}+60°F, there is no longer adequate stress intensification to promote cleavage and fracture occurs by the slower mechanism of micro-void initiation, growth, and coalescence into the crack, a behavior referred to as "ductile."

At normal operating temperature, ferritic reactor vessel materials are usually tough. However, neutron radiation embrittles the material over time, causing a shift in RT_{NDT} to higher temperatures. Correlations based on test results for unirradiated and irradiated specimens have been developed to calculate the shift in RT_{NDT} as a function of neutron fluence (the integrated neutron flux over a specified time of plant operation) for various material compositions. The value of RT_{NDT} at a given time in a reactor vessel's life is used in fracture mechanics calculations to determine the probability that assumed pre-existing flaws would propagate when the reactor vessel is stressed.

The Pressurized Thermal Shock (PTS) rule, 10 CFR 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 the rule 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 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, other 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, respectively.

No currently 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.

Technical Basis for the Proposed Amendment

The NRC's Office of Nuclear Regulatory Research (RES) 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 proposing a new rule, 10 CFR 50.61a, which would provide alternative screening criteria and corresponding embrittlement correlations based on the updated technical basis. The updated embrittlement correlation is the projected increase in the Charpy V-notch 30 ft-lb transition temperature for reactor vessel materials resulting from neutron radiation and is calculated using equations 5 through 7 of the proposed rule. The proposed rule would be voluntary for all holders of a PWR operating license under 10 CFR Part 50 or a combined license under 10 CFR 52, although it is intended for licensees with reactor vessels that cannot demonstrate compliance with the more restrictive criteria in 10 CFR 50.61. The requirements of 10 CFR 50.61 would continue to apply to licensees who choose not to implement 10 CFR 50.61a.

The following two reports provide the technical basis for this rulemaking: (1) NUREG-1806, "Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61): Summary Report," and (2) NUREG-1874, "Recommended Screening Limits for Pressurized Thermal Shock (PTS)." These reports summarize and reference several additional reports on the same topic. 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 (i.e., 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 concludes that the TWCF results and resultant RT-based screening criteria developed from their 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/frequency characteristic of the three plants that were modeled in detail.

The current guidance provided by Regulatory Guide 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," for large early release frequency (LERF) was used to relate the PTS screening criteria in 10 CFR 50.61a to an acceptable yearly limit of $1X10^{-6}$ per reactor year on reactor vessel TWCF. Although many post-through-wall cracking accident progressions are expected to lead only to core damage (which suggests a $1X10^{-5}$ events per year limit on TWCF per Regulatory Guide 1.174), uncertainties in the accident progression analysis led to the recommendation of adopting the more conservative TWCF limit of $1X10^{-6}$ per reactor year based on LERF.

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 the proposed rule are similar, the form of the revised embrittlement correlation differs substantially from the correlation in the existing 10 CFR 50.61. The correlation in 10 CFR 50.61a 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.

This proposed rule would differ from the current rule in that it would contain a requirement for licensees who choose to follow its requirements to analyze the results from the

American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section XI inservice inspection volumetric examinations. This requirement would be provided in paragraph (e) of the proposed rule. The examinations and analyses would confirm that the flaw density and size in the licensee's reactor vessel beltline are bounded by the flaw density and size utilized in the technical basis. The technical basis was developed using a flaw density, spatial distribution, and size distribution determined from a small amount of experimental data, as well as from physical models and expert elicitation. The experimental data included 22,210 cubic inches of weld metal, 3845 cubic inches of plate, and 1650 cubic inches of clad. The experimental data were obtained from samples removed from reactor vessel materials from cancelled plants (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 proposed rule may be applied are consistent with those in the technical basis because the experimental data was obtained from a limited number of reactor vessels.

Paragraph (g)(6)(ii)(c) of 10 CFR 50.55a requires licensees to implement Supplements 4 and 6 in Appendix VIII to ASME BPV Code Section XI after November 22, 2000. 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.

The performance of inspectors who have gone through the Supplement 4 qualification process has been documented in a paper by Becker (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). Analysis of the results reported in this paper indicates that an inspector using a Supplement 4 qualification procedure would have an 80 percent probability of detecting a flaw with a through-wall extent of 0.1 inch and would have an approximately 99 percent probability of detecting a flaw with a through-wall extent of 0.3 inch. Therefore, there is an 80 percent or greater probability of detecting a flaw that contributes to crack initiation from PTS events in reactor vessels with embrittlement conditions characteristic of 1×10^{-6} per reactor-year TWCF when they are inspected using ASME BPV Code Section XI, Appendix VIII, Supplement 4 requirements.

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 proposed rule would require licensees to determine if:

(1) The indication density and size within the weld and base metal inservice inspection volume from the clad-to-base metal interface to the inner 1.0 inch or 10 percent of the vessel thickness are within the flaw density and size distributions that were used in the technical basis represented in Tables 2 and 3 in the proposed rule;

(2) Any indications within the weld and base metal inservice inspection volume from the clad-to-base metal interface to the inner 1.0 inch or 10 percent of the vessel thickness are larger than the sizes in Tables 2 and 3;

(3) Any indications between the clad-to-base metal interface and three-eights of the vessel thickness exceed the size allowable in ASME BPV Code Section XI, Table IWB-3510-1; or

(4) Any linear indications that penetrate through the clad into the welds or the adjacent base metal.

The technical basis for the proposed rule concludes that flaws as small as 0.1 inch deep contribute to TWCF and that nearly all of the contributions come from flaws in the range below 1 inch deep for reactor vessels with embrittlement characteristics of TWCF equal to 1×10^{-6} per reactor year. The peak contribution comes from flaws between 0.1 and 0.2 inch deep, because that is the range that has the maximum combined effect from the number of flaws, which is decreasing with flaw size, and their susceptibility to brittle fracture, which is increasing with flaw size. For weld flaws that exceed the sizes in the table, the risk analysis indicates that a single flaw can be expected to contribute a significant fraction of the $1x10^{-6}$/reactor-year limit on TWCF. Therefore, if a flaw of that size is found in a reactor vessel, it is important to more accurately assess if its size and location with respect to the local level of embrittlement challenge the regulatory limit.

The technical basis for the proposed rule indicates that flaws buried deeper than 1 inch from the inner surface of the reactor vessel are not as susceptible to brittle fracture as similar size flaws located closer to the inner surface. Therefore, the proposed rule would not require the comparison of the density of such flaws, but still would require large flaws, if discovered, to be evaluated for contributions to TWCF if they are within the inner three-eights of the vessel thickness. This requirement would be provided in paragraph (e)(4)(iv) of the proposed rule. 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, this proposed rule would require these flaws to be evaluated for contribution to TWCF in addition to the other evaluations for such flaws that are prescribed in the ASME Code.

The numerical values in Tables 2 and 3 of the proposed rule would represent the number of flaws in each size range that were derived from the technical basis. Table 2 for the weld flaws is limited to flaw sizes that are frequent enough to be expected to occur in most plants. Similarly, Table 3 for the plate and forging flaws stops at the maximum flaw size that was modeled for these materials in the technical basis. 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 for that reactor vessel to exceed the regulatory limit.

Surface cracks that penetrate through the stainless steel clad 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 an operating PWR reactor 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"). The observed cracks had a maximum depth into the base metal of approximately 6 mm (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 10 CFR 50.61. The cracking at Quad Cities Unit 2 was attributed to intergranular stress corrosion cracking (IGSCC) of the stainless steel cladding, which has not been observed in PWR reactor vessels, and hot cracking of the low alloy steel metal base. 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 proposed rule would require licensees to determine if cracks of this type exist in the beltline weld region at each ASME Code Section XI ultrasonic examination. This requirement would be provided in paragraph (e)(2) of the proposed rule.

Development of Tables 2 and 3 Flaw Density and Size Screening Criteria

The ASME Code specifies that the dimension of flaws detected by nondestructive examination be expressed to the nearest 0.05 inch for indications less than 1 inch. Hence, the examination results from the ASME Code volumetric examination will be reported in multiples of 0.05 inch with a range of ±0.025 inch. Therefore, Tables 2 and 3 in the proposed rule describe the flaw density in multiples of 0.05 inch with a size range of ±0.025 inch.

The ASME Code standard for reporting flaw sizes did not match the size increments in the technical basis. Therefore, the NRC staff developed a procedure to distribute the flaws used in the technical basis into ASME Code-sized ranges. This is explained in greater detail in the NRC staff document “Development of Flaw Size Distribution Tables for Draft Proposed Title 10 of the Code of Federal Regulations (10 CFR) 50.61a” (refer to ADAMS accession number ML070950392).

The values in Tables 2 and 3 of the proposed rule exceed the values for those size ranges that were developed from the laboratory analyses of the two reactor vessels. It was decided to allow licensees to use the Table 2 and 3 values instead of the values that would come from the laboratory results because it is still conservative to model all of the flaws as if they were the largest size for each of the ASME Code size ranges. In effect, some of the conservatism that was in the original risk modeling is being made available to licensees for demonstrating that the results of an individual plant’s ASME Code examinations are consistent with the underlying technical basis.

Rulemaking Initiation

In SECY-06-0124, dated May 26, 2006, the NRC staff presented a rulemaking plan to the Commission to amend fracture toughness requirements for PWRs. In this SECY paper, the NRC staff proposed four options for rulemaking. The NRC staff recommended Option 3, which would allow licensees to voluntarily implement the less restrictive screening limits based on the updated technical basis and insert the updated embrittlement correlation into 10 CFR 50.61 to maintain regulatory consistency and implement the best state-of-the-art embrittlement correlation in both 10 CFR 50.61 and 10 CFR 50.61a. This recommendation was based on providing the necessary relief to licensees that would otherwise expend considerable resources to justify continued plant operation beyond the screening criteria in 10 CFR 50.61 (via compensatory actions, plant-specific analyses, annealing or exemption), while also requiring all licensees to recalculate their embrittlement metric to ensure that all plants’ analyses are consistent.

In a Staff Requirements Memorandum (SRM) dated June 30, 2006, the Commission approved the initiation of the rulemaking as specified in Option 2 of the rulemaking plan. This option would require licensees to continue to meet the requirements of 10 CFR 50.61, which provides adequate protection against PTS events, without implementing the updated embrittlement correlation. For licensees whose reactor vessels do not meet the requirements of 10 CFR 50.61, Option 2 would allow licensees to voluntarily implement 10 CFR 50.61a which utilizes the less restrictive screening limits based on the updated technical basis as well as the updated embrittlement correlation. Accordingly, the proposed rule provides for a voluntary alternative to the current set of PTS requirements for any PWR licensee. The NRC considered requiring new plants to use the best available embrittlement correlation (i.e., the embrittlement correlation developed for the new rule). The NRC believes that such a requirement was not necessary to provide adequate protection of public health and safety. The NRC believes that imposing the existing 10 CFR 50.61, without modification, on new reactors would ensure that adequate protection concerns would be met. The NRC believes that the proposed rule's requirements should be a voluntary alternative available to new plants, if needed.

In implementing the rulemaking plan, the proposed rule would provide a new section, 10 CFR 50.61a, for the new set of fracture toughness requirements. 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 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 proposed rule would provide a voluntary alternative to the current rule, which further prompted the NRC to keep the current, mandatory requirements separate from the new, voluntarily-implemented requirements. As a result, the proposed new rule would retain the current requirements in §50.61 for PWR licensees choosing not to implement the less restrictive screening limits, and would present new requirements in § 50.61a as a voluntary relaxation for any PWR licensee.

II. Section-by-Section Analysis

Section 50.61–Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events

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

Section 50.61a–Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events

Proposed new § 50.61a would contain pressurized thermal shock screening limits based on updated probabilistic fracture mechanics analyses. This new section would provide similar requirements to that of § 50.61, fracture toughness requirements for protection against pressurized thermal shock events for pressurized water nuclear power reactors. However, § 50.61a would differ 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 proposed rule would require 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 proposed § 50.61a would provide 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 would 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 mechanisms, and engineering judgement. The embrittlement data base used for this analysis was derived primarily from the Power Reactor Embrittlement Data Base (PR-EDB) developed at Oak Ridge National Laboratory. The updated RT_{MAX-X} estimation procedures provide a more realistic (compared to the existing regulation) method for estimating the fracture toughness of reactor vessel materials over the lifetime of the plant.

Paragraph (a) would contain definitions for terms used in § 50.61a. It would also provide that terms defined in § 50.61 also have the same meaning in § 50.61a unless otherwise noted.

Paragraph (b) would describe the applicability of § 50.61a to PWRs as an alternative to the requirements of § 50.61. The requirements of this section would provide a voluntarily-implemented alternative to the current requirements of § 50.61 for any current PWR licensee or future holder of a PWR operating license or combined license.

Paragraph (c) would set forth the requirements governing NRC approval of a licensee's use of § 50.61a. The licensee would make the formal request to the NRC via a license amendment, and only upon approval of the license amendment by the NRC would a licensee be permitted to implement § 50.61a. In the licensee's amendment request, the required information would include a) calculating the values of RT_{MAX-X} values as required by paragraph (c)(1), b) examining and assessing flaws discovered by ASME Code inspections as required by paragraph (c)(2), and c) comparing the RT_{MAX-X} values against the applicable screening criteria as required by paragraph (c)(3). In doing so, the licensee would also be required to utilize paragraphs (e)(1) through (e)(3), paragraph (f), and paragraph (g) in order to perform the necessary calculations, comparisons, examinations, assessments, and analyses.

Paragraph (d) would define the requirements for subsequent examinations and flaw assessments after initial approval to use § 50.61a has been obtained under the requirements of paragraph (c). It would also define the required compensatory measures or analyses to be taken if a licensee determines that the screening criteria will be exceeded. Paragraph (d)(1) would define the requirements for subsequent RT_{MAX-X} assessments consistent with the requirements of paragraphs (c)(1) and (c)(3). Paragraph (d)(2) would define the requirements for subsequent examination and flaw assessments utilizing the requirements of paragraphs (e)(1), (e)(1)(i), (e)(1)(ii), (e)(2), and (e)(3). Paragraphs (d)(3) through (d)(7) would 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.

Paragraph (e) would define the requirements for verifying that the PTS screening criteria in § 50.61a are applicable to a particular reactor vessel. The proposed rule would require that verification be based on an analysis of test results from ultrasonic examination of the reactor vessel beltline materials required by Section XI of the ASME Code.

Paragraph (e)(1) would establish cumulative limits on flaw density and size within the ASME Code, Section XI, Appendix VIII, Supplement 4 inspection volume, which corresponds to a depth of approximately one inch from the clad-to-base metal interface. The allowable number of flaws provided in Tables 2 and 3 are cumulative values. If flaws exist in larger increments, the allowable number of flaws is the value in Table 2 or 3 for that increment minus the total number of flaws in all larger increments. Flaws in this inspection volume contribute approximately 97-99 percent to the TWCF at the screening limit.

Paragraph (e)(1)(i) would describe the flaw density limits for welds.

Paragraph (e)(1)(ii) would describe the flaw density limits for plates and forgings.

Paragraph (e)(1)(iii) would describe the specific ultrasonic examination and neutron fluence information to be submitted to the NRC. The NRC would utilize this information to evaluate whether plant-specific information gathered in accordance with this rule suggests that the NRC staff should generically re-examine the technical basis for the rule.

Paragraph (e)(2) would require that licensees verify that no clad-base metal interface flaws within the ASME Code, Section XI, Appendix VIII, Supplement 4 inspection volume open to the vessel inside surface. These types of flaws could have a substantial effect on the TWCF.

Paragraph (e)(3) would establish limits on flaw density and size beyond the ASME Code, Section XI, Appendix VIII, Supplement 4 inspection volume to three-eights of the reactor vessel thickness from the interior surface. Flaws in this inspection volume contribute approximately 1-3 percent to the TWCF at the screening criteria. Flaws exceeding this limit could affect the TWCF. Flaws greater than three-eights of the reactor vessel thickness from the interior surface do not contribute to the TWCF at the screening limit.

Paragraph (e)(4) would establish requirements to be met if flaws exceed the limits in (e)(1) and (e)(3) or open to the inside surface of the reactor vessel. This section requires an analysis to demonstrate 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, 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. This paragraph would be required to be implemented if the requirements of (e)(1) through (e)(3) are not satisfied.

Paragraph (e)(5) would describe the critical parameters to be addressed if flaws exceed the limits in (e)(1) and (e)(3) or if the flaws would open to the inside surface of the reactor vessel. This paragraph would be required to be implemented if the requirements of (e)(1) through (e)(3) are not satisfied.

Paragraph (f) would define the process for calculating RT_{MAX-X} values. These values would be 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.

Paragraph (g) would provide the necessary equations and variables required by paragraph (f) of this section.

Table 1 would provide the PTS screening criteria for comparison with the licensee's calculated RT_{MAX-X} values. Tables 2 and 3 would provide values to be used in paragraph (e) of this section. Tables 4 and 5 would provide values to be used in paragraph (f) of this section.

III. 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; 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* (10 CFR). 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, but does not confer regulatory authority on the State.

IV. Availability of Documents

The following table lists documents relating to this rulemaking which are available to the public and how they may be obtained.

Public Document Room (PDR). The NRC's Public Document Room is located at the NRC's headquarters at 11555 Rockville Pike, Rockville, MD 20852.

Rulemaking Website (Web). The NRC's interactive rulemaking Website is located at http://ruleforum.llnl.gov. These documents may be viewed and downloaded electronically via this Website.

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

Document	**PDR**	**Web**	**ERR** (ADAMS)
Regulatory Analysis	x	x	ML070570383
OMB Supporting Statement	x	x	ML070570446
SECY-06-0124, May 26, 2006, Rulemaking Plan Request for Commission Approval	x		ML060530624
SRM-SECY-06-0124, June 30, 2006, Staff Requirements - Commission Approval of Rulemaking Plan	x		ML061810148
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"	x		ML043060581
NUREG-1806, "Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limits in the PTS Rule (10CFR50.61): Summary Report"	x		ML061580318
NUREG-1874, "Recommended Screening Limits for Pressurized Thermal Shock (PTS)"	x		ML070860156

Document	PDR	Web	ERR (ADAMS)
Regulatory Guide 1.154, “Format and Content of Plant-Specific Pressurized Thermal Shock Analysis Reports for Pressurized Water Reactors”	x		ML003740028
Regulatory Guide 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis”	x		ML023240437
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

V. Plain Language

The Presidential memorandum dated June 1, 1998, entitled “Plain Language in Government Writing” directed that the Government’s writing be in plain language. This memorandum was published on June 10, 1998 (63 FR 31883). 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 address listed under the ADDRESSES caption of the preamble of this document.

VI. 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 considered using 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 proposed rule. However, the correlation developed by RES 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 proposed rule because it does not represent the broad range of conditions necessary to justify a revision to the regulations.

American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requirements are utilized as part of the volumetric examination analysis requirements of the 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 50 Appendix H and utilized to determine 30-foot-pound transition temperatures. These standards were selected for use in the proposed rule based on their use in other regulations within Part 50 and their applicability to the subject of the desired requirements.

The NRC will consider using other voluntary consensus standards if appropriate standards are identified.

VII. Finding of No Significant Environmental Impact: Environmental Assessment

The Commission has determined under the National Environmental Policy Act of 1969, as amended, and the Commission's regulations in Subpart A of 10 CFR part 51, 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 basis for this determination is as follows:

Environmental Impacts of the Action:

This environmental assessment focuses on those aspects of § 50.61a where there is a potential for an environmental impact. The NRC has concluded that there will be no significant radiological environmental impacts associated with implementation of the rule requirements for the following reasons:

(1) Section 50.61a would maintain the same functional requirements for the facility as the existing PTS rule in § 50.61 as a voluntary alternative to the existing rule. This proposed rule would establish screening criteria, limiting levels of embrittlement beyond which operation cannot continue without further plant-specific evaluation or modifications, as well as require calculation of the maximum embrittlement predicted at the end of the licensed period of operation. The screening criteria provide reasonable assurance that licensees operating below (predicted embrittlement less than) the screening criteria could endure a pressurized thermal shock event without fracture of vessel materials, thus assuring integrity of the reactor pressure vessel.

(2) The new rule is risk-informed and in accordance with the NRC's 1995 PRA policy statement and risk-informed regulation guidance. Sufficient safety margins are maintained to ensure that any potential increases in core damage frequency (CDF) and large early release frequency (LERF) resulting from implementation of § 50.61a are negligible.

The action 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 action.

With regard to potential nonradiological impacts, implementation of the rule requirements has no impact on the facility other than to provide a more realistic method of calculating PWR vessel fracture toughness with associated limits. Nonradiological plant effluents are not affected and there are no other environmental impacts. Therefore, the NRC concludes that there are no significant environmental impacts associated with the action.

<u>Alternatives to the Action:</u>

As an alternative to the rulemaking described above, the NRC considered not taking the action (i.e., the “no-action” alternative). Not adopting the more realistic and less conservative regulation would result in no change in environmental impacts for current PWRs or those that would be expected for future PWRs under 10 CFR 50.61.

Agencies and Persons Consulted:

The NRC staff developed the proposed rule and this environmental assessment. Under the NRC’s stated policy, a copy of this environmental assessment will be provided to the state liaison officials as part of the publication of the proposed rule for public comment.

Conclusion

On the basis of this environmental assessment, the NRC concludes that the action would not have a significant effect on the quality of the human environment. Accordingly, the NRC has determined not to prepare an environmental impact statement for the action.

The determination of this environmental assessment is that no significant offsite impact to the public from this action would occur. However, the general public should note that the NRC is seeking public participation. Comments on any aspect of the environmental assessment may be submitted to the NRC as indicated under the ADDRESSES heading.

The NRC has sent a copy of this proposed rule to every State Liaison Officer and requested their comments on the environmental assessment.

VIII. Paperwork Reduction Act Statement

This 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 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 60.61 and 50.61a)" 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 a voluntary alternative to the requirements of 10 CFR 50.61. Collections would also be required, after voluntary implementation of the new § 50.61a, when any change is made to the design or operation of the facility that affects the calculated RT_{MAX-X} value. Collections would also be required during the scheduled periodic ultrasonic examination of beltline welds.

Who will be required or asked to report: Any PWR licensee voluntarily 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: 264 hours (24 hours annually for recordkeeping plus 240 hours annually for reporting)

Abstract: The NRC is proposing to amend its regulations to provide updated fracture toughness requirements for protection against pressurized thermal shock (PTS) events for pressurized water reactor (PWR) pressure vessels. The 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 would 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 voluntarily utilized by any PWR licensee as an alternative to complying with the existing requirements.

The U.S. Nuclear Regulatory Commission is seeking public comment on the potential impact of the information collections contained in this 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-1 F21, 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 for 60 days after the signature date of this notice and are also available at the rule forum site, http://ruleforum.llnl.gov.

Send comments on any aspect of these proposed information collections, including suggestions for reducing the burden and on the above issues, by [INSERT DATE 30 DAYS AFTER PUBLICATION IN THE FEDERAL REGISTER] to the Records and FOIA/Privacy Services Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by Internet electronic mail to INFOCOLLECTS@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. 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. You may also comment by telephone at (202) 395-3087.

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.

IX. Regulatory Analysis

The Commission has prepared a draft regulatory analysis on this proposed regulation. The analysis examines the costs and benefits of the alternatives considered by the Commission. The Commission requests public comments on this draft regulatory analysis. Availability of the regulatory analysis is provided in Section IV. Comments on the draft regulatory analysis may be submitted to the NRC as indicated under the ADDRESSES heading of this document.

In addition, the Commission also requests public comments on the cost and benefit of requiring PWR licensees to revise their vessel analyses if the updated embrittlement correlation were imposed in 10 CFR 50.61. This would differ from the proposed rule, which leaves the technical content of 10 CFR 50.61 unchanged.

X. Regulatory Flexibility Certification

Under the Regulatory Flexibility Act (5 U.S.C. 605(b)), the Commission certifies that this rule would not, if promulgated, have a significant economic impact on a substantial number of small entities. This proposed rule would affect only the licensing and operation of 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).

XI. Backfit Analysis

The NRC has determined that the requirements in this proposed rule do 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 proposed rule. The requirements of the proposed PTS rule, 10 CFR 50.61a, would not be required, but could be voluntarily utilized, by any PWR licensee. Licensees choosing to implement the proposed PTS rule would be required to comply with its requirements as a voluntary 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 by any PWR licensee, the NRC finds that this amendment would not constitute backfitting.

List of Subjects in 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). Section 50.7 also issued under Pub. L. 95-601, sec. 10, 92 Stat. 2951 (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. 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 certifications required under § 50.82(a)(1) have been submitted, the licensee shall have projected values of RT_{PTS} or $RT_{MAX\text{-}X}$, accepted by the NRC, for each reactor vessel beltline material for the EOL fluence of the material in accordance with this section or § 50.61a. For a licensee choosing to comply with this section, the assessment of RT_{PTS} must use the calculation procedures given in paragraph (c)(1) of this section, except as provided 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.

* * * * *

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) $RT_{MAX\text{-}AW}$ 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-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 holder of an operating license under this part or holder of a combined license under part 52 of this chapter of a pressurized water nuclear power reactor 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 10 CFR Part 50 or 10 CFR Part 52.

(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 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

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[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.

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 $1x10^{-6}$ 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 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

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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.

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)(iv) 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 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}

[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.

value used for calculating the plate, forging, and circumferential weld RT_{MAX-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 specifications 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-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 weighted time average of the reactor cold leg temperature under normal operating full power conditions from

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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.

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 has been deemed consistent as judged by the following criteria:

(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 50 Appendix H.

(B) If three or more surveillance data points exist for a specific material, the surveillance data must be evaluated for consistency with the model in Equations 5, 6, and 7 as specified by paragraph (f)(6)(ii) of this section. If fewer than three surveillance data points exist for a specific material, then Equations 5, 6, and 7 of this section must be used without performing the consistency check.

(ii) The licensee shall estimate the mean deviation from the model (Equations 5, 6 and 7 of this section) for the specific data set (i.e., a group of surveillance data points representative of a given material). The mean deviation from the 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 Equation 10 of this section and based on the material group into which the surveillance material falls and the number of available data points. The licensee shall determine, based on this comparison, if the surveillance data show a significantly different trend than the model predicts. The surveillance data analysis must follow the criteria in paragraphs (f)(6)(iii) through (f)(6)(iv) 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 comply with Table 5 of this section.

(iii) If the mean deviation from the model for the data set is equal to or less than the value in Table 5 or the value using Equation 10 of this section, then the ΔT_{30} value must be determined using Equations 5, 6, and 7 of this section.

(iv) If the mean deviation from the model for the data set is greater than the value in Table 5 or the value using Equation 10 of this section, the ΔT_{30} value must be determined using the surveillance data. If the mean deviation from the model for the data set is outside the limits specified in Equation 10 of this section or in Table 5 of this section, 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 its ΔT_{30} and RT_{MAX-X} values for review and approval by the Director no later than one year after the surveillance capsule is withdrawn from the reactor 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: $RT_{MAX-AW} = MAX\ \{[RT_{NDT(u)\text{ - plate}} + \Delta T_{30\text{ - plate}}(\varphi t_{FL})],$

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

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

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

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

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

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

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

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

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

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 φ greater than or equal to 4.39×10^{10} n/cm^2/sec
= $\varphi t \cdot (4.39 \times 10^{10}/\varphi)^{0.2595}$ for φ less than 4.39×10^{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] = $\varphi \cdot t$

$f(Cu_e,P)$ = 0 for Cu $\leq$ 0.072
= $[Cu_e - 0.072]^{0.668}$ for Cu > 0.072 and P $\leq$ 0.008

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

and Cu_e = 0 for Cu $\leq$ 0.072

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

and maximum Cu_e = 0.243 for Linde 80 welds

= 0.301 for all other materials

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

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 = $\sum_{i=1}^{n} r_i / n$

Equation 10: Maximum credible heat-average residual = $3\sigma/n^{0.5}$

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.

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})		
	$T_{WALL} \leq$ 9.5in.	9.5in. < $T_{WALL} \leq$ 10.5in.	10.5in. < $T_{WALL} \leq$ 11.5in.
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 $1x10^{-8}$ 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	$0.025 \leq TWE < 0.075$	Unlimited
0.10	$0.075 \leq TWE < 0.125$	166.70
0.15	$0.125 \leq TWE < 0.175$	90.80
0.20	$0.175 \leq TWE < 0.225$	22.82
0.25	$0.225 \leq TWE < 0.275$	8.66
0.30	$0.275 \leq TWE < 0.325$	4.01
0.35	$0.325 \leq TWE < 0.375$	3.01
0.40	$0.375 \leq TWE < 0.425$	1.49
0.45	$0.425 \leq TWE < 0.475$	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	$0.025 \leq TWE < 0.075$	Unlimited
0.10	$0.075 \leq TWE < 0.125$	8.049
0.15	$0.125 \leq TWE < 0.175$	3.146
0.20	$0.175 \leq TWE < 0.225$	0.853
0.25	$0.225 \leq TWE < 0.275$	0.293
0.30	$0.275 \leq TWE < 0.325$	0.0756
0.35	$0.325 \leq TWE < 0.375$	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

Material Group	σ [°F]	Number of available data points					
		3	**4**	**5**	**6**	**7**	**8**
Welds, for Cu > 0.072	26.4	45.7	39.6	35.4	32.3	29.9	28.0
Plates, for Cu > 0.072	21.2	36.7	31.8	28.4	26.0	24.0	22.5
Forgings, for Cu > 0.072	19.6	33.9	29.4	26.3	24.0	22.2	20.8
Weld, Plate or Forging, for Cu ≤ 0.072	18.6	32.2	27.9	25.0	22.8	21.1	19.7

Dated at Rockville, Maryland, this _________ day of _________, 2007.

For the Nuclear Regulatory Commission.

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