RULEMAKING ISSUE
(Notation Vote)

June 25, 2007 SECY-07-0104

FOR: The Commissioners

FROM: Luis A. Reyes
Executive Director for Operations /RA/

SUBJECT: PROPOSED RULEMAKING — ALTERNATE FRACTURE TOUGHNESS REQUIREMENTS FOR PROTECTION AGAINST PRESSURIZED THERMAL SHOCK EVENTS (RIN 3150-AI01)

PURPOSE:

To obtain Commission approval to publish for public comment a proposed rule that would provide new fracture toughness requirements for pressurized water reactors (PWRs). This paper does not address any new commitments.

SUMMARY:

The enclosed proposed rule, Title 10 of the *Code of Federal Regulations*, Section 50.61a (10 CFR 50.61a), "Alternative Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events" (Enclosure 1), would amend the current regulations for PWRs by providing alternate fracture toughness requirements for protection against pressurized thermal shock (PTS) events. PWR licensees, including future holders of operating licenses and combined licenses, could choose to comply with the requirements of this new section as a voluntary alternative to the current PTS requirements of 10 CFR 50.61. The proposed amendment would reduce the regulatory burden on some licensees due to the unnecessarily conservative requirements of the current regulation while maintaining adequate safety. Several operating reactors that are projected to exceed the screening limits of 10 CFR 50.61 before the expiration of their renewed operating licenses would benefit from the new screening limits and correlations of 10 CFR 50.61a.

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BACKGROUND:

The PTS Rule (10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events") protects against brittle fracture of reactor vessels during severe cool-down events. This rule provides embrittlement correlations that licensees must use to determine a reference temperature for each vessel beltline material. The reference temperature is then compared to the rule's screening criterion. Licensees may not operate at a reference temperature in excess of the screening criterion without approval of the Nuclear Regulatory Commission's (NRC's) Director of the Office of Nuclear Reactor Regulation (NRR).

The screening criteria in the current rule are based on a conservative probabilistic fracture mechanics analysis developed in the 1980s. The analysis used conservative assumptions and a margin term to account for a limited data set and the limited computational resources and techniques available at that time. With the data and computational resources and analysis techniques available today, the NRC considers the screening criteria in the current rule to be unnecessarily conservative. Several licensees expect to exceed the current screening criteria before the expiration of their renewed operating licenses. Those licensees would need to take compensatory actions to avoid exceeding the screening criteria, which could result in costly analyses, modifications to the plant, reactor vessel thermal annealing, or cessation of plant operation.

The staff has completed a research program to provide the technical basis for 10 CFR 50.61a. This program is summarized in NUREG-1806, "Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limits in the PTS Rule (10 CFR 50.61): Summary Report," and NUREG-1874, "Recommended Screening Limits for Pressurized Thermal Shock (PTS)." The staff concluded that the risk of through-wall cracking due to PTS events is much lower than previously calculated. Thus, the screening criteria in 10 CFR 50.61 may impose an unnecessary burden on some licensees.

In SECY-06-0124, "Rulemaking Plan to Amend Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock," dated May 26, 2006 (ADAMS Accession No. ML060530624), the staff proposed to initiate a rulemaking to revise the existing regulation. In this paper, the staff provided four options for the Commission's consideration. The staff recommended that the Commission approve Option 3. Option 3 would allow licensees to voluntarily implement the less restrictive screening criteria based on the updated technical basis and insert the updated embrittlement correlation into 10 CFR 50.61 for regulatory consistency. Option 3 would apply the best available technology for 10 CFR 50.61 and 10 CFR 50.61a.

In response to SECY-06-0124, the Commission directed the staff to conduct a rulemaking as specified in Option 2 of the rulemaking plan in staff requirements memorandum (SRM) SRM-SECY-06-0124, dated June 30, 2006 (ADAMS Accession No. ML061810148). In Option 2, the staff would amend the regulation to allow licensees to voluntarily implement the less restrictive screening criteria based on the updated technical basis without implementing the updated embrittlement correlation in 10 CFR 50.61. In the SRM, the Commission requested the staff to seek early interaction with, and specific feedback from, stakeholders regarding the potential impacts (e.g., cost/benefit) of requiring all licensees to use the updated embrittlement correlation as described in Option 3. Further, the Commission directed the staff to assess the impacts of the updated correlation, identifying which reactors would predict a higher embrittlement and if any regulatory action would be needed. The staff should consider requiring new plants use the best available embrittlement correlation. The staff should ensure that the probabilistic assumptions used in the technical basis are consistent with those used in other risk-informed initiatives and that plant aging effects have been reasonably considered over the extended plant lifetimes. Finally, the Commission specifically requested the staff to seek Advisory Committee on Reactor Safeguards (ACRS) comment on the most important aspects of the probabilistic basis.

DISCUSSION:

Implementation of Option 2

The proposed rulemaking would implement the Commission's direction to amend the regulations to allow licensees to voluntarily implement the requirements of the new PTS rule without implementing the updated embrittlement correlation in 10 CFR 50.61. The current mandatory PTS requirements of 10 CFR 50.61 would continue to apply for any current or future PWR licensee. The proposed rule would provide a new section, 10 CFR 50.61a, which licensees could choose to comply with as a voluntary alternative to the requirements of the current PTS rule. This new section would require the use of the updated embrittlement correlations and their corresponding screening criteria. Implementation of the new rule would involve a licensee choosing to comply either with 10 CFR 50.61 or with 10 CFR 50.61a. The NRC would not require current operating reactor licensees or licensees referencing certified reactor designs based on the current rule to take additional action if they choose to continue complying with 10 CFR 50.61. However, those licensees may choose to voluntarily implement 10 CFR 50.61a as an alternative to compliance with 10 CFR 50.61. A future reactor licensee can choose to comply with either regulation in its request for an operating license or combined construction/operating license.

Stakeholder Interaction and Feedback

The staff did not seek early interaction with stakeholders on a possible requirement for all licensees to use the updated embrittlement correlation. The staff determined that requiring licensees to use the updated embrittlement correlation would be considered a backfit under 10 CFR 50.109. The staff believes that such backfitting would not fall under any of the three 10 CFR 50.109(a)(4) exceptions to preparation of a backfit analysis. In addition, based on available information, the staff believes that such backfitting could not be justified, using a quantitative methodology, as a substantial increase in protection of public health and safety and whose costs are justified in light of this increased level of protection. As a result, the staff did not pursue early interactions with stakeholders on a possible requirement for all licensees to use the updated embrittlement correlation. However, the proposed rule does request public comment on the possibility of imposing the updated embrittlement correlation in 10 CFR 50.61. The staff believes that the public comment process will provide adequate stakeholder feedback on the proposed rule. The staff understands that the Commission may decide, in its consideration of the draft proposed rule, that backfitting of the updated embrittlement correlation should be based upon qualitative factors. For example, in SRM-SECY-93-086, dated June 30, 1993, the Commission stated that a demonstration of a "substantial increase in safety" could be based upon consideration of qualitative factors.

If the Commission wishes to retain the option of adopting a final rule imposing the updated embrittlement correlation in 10 CFR 50.61 without the need for further renoticing, then the staff recommends, consistent with the June 30, 1993, SRM as well as applicable law on renoticing of proposed rules, that the backfitting discussion in the Federal Register Notice for the draft proposed rule in this paper be further modified to add a backfit analysis of such a requirement. The backfit analysis would rely upon qualitative factors in demonstrating that there is a substantial increase in safety from imposing the updated embrittlement correlation in 10 CFR 50.61, and that the costs of imposing the updated embrittlement correlation are justified in light of the increase in safety. However, it should be noted that if the Commission directs the staff to develop such a qualitative backfit analysis prior to the publication of the draft rule, this can be expected to have a significant impact on the staff's schedule for the publication of the draft rule. Furthermore, the staff's efforts may conclude that such a qualitative backfit analysis does not demonstrate a substantial increase in safety.

Detailed Assessment of the Impact of the Updated Correlation

The staff evaluated reactors that were projected to be above or near the current PTS screening criteria in 10 CFR 50.61. Table 1 identifies the reactors that are projected to be above the current PTS screening criteria using the embrittlement correlation in the current PTS rule and/or the updated embrittlement correlation. All reference temperature for pressurized thermal shock (RT_{PTS}) values are projected to the end of an extended operating period (60 years of operation). The RT_{PTS} values for the limiting materials for Beaver Valley Unit 1 and Three Mile Island Unit 1 are projected to exceed the current PTS screening criteria using the current PTS rule embrittlement correlation, but remain below the current PTS screening criteria using the updated embrittlement correlation. The RT_{PTS} values for the limiting materials for Palisades, Point Beach Unit 2, Indian Point Unit 3, and Diablo Canyon Unit 1 are projected to exceed the current PTS screening criteria using both the current PTS rule embrittlement correlation and using the updated embrittlement correlation. The limiting material for Salem Unit 1 changes from an axial weld to a plate. The axial weld RT_{PTS} values are projected to exceed the current PTS screening criteria using the current PTS rule embrittlement correlation, but remain below the current PTS screening criteria using the updated embrittlement correlation. The plate RT_{PTS} values are projected to remain below the current PTS screening criteria using the current PTS rule embrittlement correlation, but exceed the current PTS screening criteria using the updated embrittlement correlation. The RT_{PTS} values for the limiting material for Fort Calhoun (axial welds) are projected to remain below the current PTS screening criteria using the current PTS rule embrittlement correlation, but exceed the current PTS screening criteria using the updated embrittlement correlation.

Table 1 - Comparison of RT_{PTS} Values Using the Embrittlement Correlation in the Current PTS Rule, 10 CFR 50.61, and the Updated Embrittlement Correlation at the End of the Renewed License Period

Plant	Limiting Reactor Vessel Material	PTS Screening Criteria in 10 CFR 50.61 (°F)	RT_{PTS} Value[1] Using Correlation in 10 CFR 50.61 (°F)	RT_{PTS} Value[2] Using Updated Correlation (°F)
Beaver Valley Unit 1	Plate	270	290	255
Palisades	Axial Weld	270	287	283
Palisades	Circ Weld	300	302	278
Point Beach Unit 2	Circ Weld	300	315	307
Three Mile Island Unit 1	Axial Weld	270	289	253
Three Mile Island Unit 1	Circ Weld	300	316	265
Indian Point Unit 3	Plate	270	280	292
Salem Unit 1	Axial Weld	270	278	252
Salem Unit 1	Plate	270	256	277
Fort Calhoun	Axial Weld 1	270	256	282
Fort Calhoun	Axial Weld 2	270	245	282
Diablo Canyon Unit 1	Axial Weld	270	283	273

[1] RT_{PTS} values are calculated using the methodology in 10 CFR 50.61.
[2] RT_{PTS} values are calculated using Equations 1 and 2 in 10 CFR 50.61; where the ΔRT_{NDT} is calculated using equations 5, 6, and 7 in the proposed 10 CFR 50.61a and the standard deviation for ΔRT_{NDT} is the standard deviation of the residuals for the updated embrittlement correlation (from Table 5 in the proposed 10 CFR 50.61a).

The staff also evaluated the limiting materials for other units not included in the table. The staff evaluated the following reactors with reactor vessel materials that are projected to be near the PTS screening criteria in 10 CFR 50.61: Surry Unit 1, Oconee Unit 2, Turkey Point Units 3 and 4, H.B. Robinson Unit 2, Ginna, Calvert Cliffs Unit 1, Watts Bar Unit 1, Sequoyah Unit 1, and North Anna Units 1 and 2. The RT_{PTS} values for these plants' limiting materials are projected to remain below the current PTS screening criteria using both the current PTS rule embrittlement correlation and the updated embrittlement correlation.

In its evaluation of operating reactors, the staff observed that reactor vessel forgings with underclad cracks could have RT_{PTS} and RT_{MAX-X} values at the end of their licenses that are near the screening criteria in both 10 CFR 50.61 and 10 CFR 50.61a (Note: RT_{MAX-X} is the equivalent term for RT_{PTS} in 10 CFR 50.61a.). This condition occurs when surveillance data is used to determine the RT_{PTS} values. The staff has reviewed all of the operating reactors that are susceptible to underclad cracking and has determined that the only reactor vessel of potential concern is the Watts Bar Unit 1 reactor vessel. Watts Bar Unit 1 is the only operating reactor vessel fabricated with forgings that are susceptible to underclad cracks and has its forging material in its reactor vessel surveillance program. Watts Bar Unit 1 is projected to approach the PTS screening criteria in 10 CFR 50.61 at the expiration of its license. At this time, Watts Bar Unit 1 has insufficient surveillance data to determine if the issue is relevant to its reactor vessel. When the licensee for Watts Bar Unit 1 removes, tests, and reports the results of its next surveillance capsule, the staff will evaluate the data to determine if the Watts Bar Unit 1 reactor vessel is projected to exceed the screening criteria in 10 CFR 50.61 or 10 CFR 50.61a. However, Table 2 indicates that, without considering surveillance data, Watts Bar Unit 1 is not projected to exceed the screening criteria in 10 CFR 50.61a. The staff does not believe that underclad cracking will be a concern for newly fabricated reactor vessels, because licensees who follow the guidance in Regulatory Guide 1.43, “Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components,” should comply with the requirements in 10 CFR 50.61a. Reactor vessels fabricated in accordance with this regulatory guide should not be susceptible to underclad cracking in their forgings.

Table 2 - Impact of the Proposed Voluntary Rule, 10 CFR 50.61a, on Reactor Vessels Projected to be Above the Screening Criteria in Table 1

Plant	Limiting Reactor Vessel Material	PTS Screening Criteria in 10 CFR 50.61a (°F)	RT_{MAX-X} Value[1] Using Correlation in 10 CFR 50.61a (°F)
Beaver Valley Unit 1	Plate	356	212
Beaver Valley Unit 1	Plate + Axial Weld	538	424
Palisades	Axial Weld	269	220
Palisades	Plate + Axial Weld	538	408
Palisades	Circ Weld	312	215
Point Beach Unit 2	Circ Weld	312	245
Three Mile Island Unit 1	Axial Weld	269	187
Three Mile Island Unit 1	Plate + Axial Weld	538	271
Three Mile Island Unit 1	Circ Weld	312	198
Indian Point Unit 3	Plate	356	249
Indian Point Unit 3	Plate + Axial Weld	538	498
Salem Unit 1	Axial Weld	269	234
Salem Unit 1	Plate	356	234
Salem Unit 1	Plate + Axial Weld	538	468
Fort Calhoun	Axial Weld 1	269	219
Fort Calhoun	Axial Weld 2	269	219

Plant	Limiting Reactor Vessel Material	PTS Screening Criteria in 10 CFR 50.61a (°F)	RT_{MAX-X} Value[1] Using Correlation in 10 CFR 50.61a (°F)
Fort Calhoun	Plate + Axial Weld	538	361
Diablo Canyon Unit 1	Axial Weld	269	210
Diablo Canyon Unit 1	Plate + Axial Weld	538	355
Kewaunee	Circ Weld	312	252
Watts Bar Unit 1	Forging	246	206
[1] RT_{MAX-X} values are calculated using the methodology in the proposed 10 CFR 50.61a.			

The Kewaunee reactor vessel's RT_{PTS} value was projected to be near the current PTS screening criteria at the end of the extended operating period. However, this RT_{PTS} value was determined using a plant-specific embrittlement correlation with a methodology not in accordance with the methodology in 10 CFR 50.61. The licensee received approval for an exemption (ADAMS Accession No. ML011210180) from the rule. The updated embrittlement correlation, from a technical standpoint, cannot be directly imposed on the licensee-developed methodology due to the differences in methodologies. Hence, the staff did not include Kewaunee in its evaluation of the effects of using the updated embrittlement correlation on its RT_{PTS} value. The staff will discuss with the licensee for Kewaunee its plant-specific methodology for determining its RT_{PTS} value and whether this method will need to be changed based on the updated embrittlement correlation. The impact of using 10 CFR 50.61a on the Kewaunee reactor vessel is shown in Table 2.

All other operating PWRs have projected RT_{PTS} values so far below the current PTS screening criteria that the updated embrittlement correlation would have no regulatory impact with respect to compliance with the rule. However, if the NRC were to impose the updated embrittlement correlation on all PWR licensees, the regulatory impact would include expenditure of licensee resources to update and maintain documentation of each licensee's compliance with the updated embrittlement correlation.

The staff has determined that imposition of the new embrittlement correlation within the existing rule would have a substantive, negative regulatory impact on only Ft. Calhoun. The staff will discuss their assessment with the licensee and recommends taking the appropriate action on a plant-specific basis. In addition, although the limiting material may change at Salem Unit 1, its licensee would still be expected to voluntarily implement 10 CFR 50.61a because the plant has at least one material that exceeds the 10 CFR 50.61 criteria regardless of which embrittlement correlation is used. The voluntary implementation of 10 CFR 50.61a would require the licensee to evaluate all of their reactor vessel beltline materials.

Table 2 assesses the impact of the proposed rule (10 CFR 50.61a) on the limiting materials in the same reactor vessels as identified in Table 1, with the addition of Kewaunee and Watts Bar Unit 1. Table 2 provides the applicable screening criteria and the projected RT_{MAX-X} values for the reactor vessels to the end of an extended operating period (60 years of operation). RT_{MAX-X}

is defined in 10 CFR 50.61a and is the material property parameter that is calculated based on the projected amount of radiation embrittlement. The proposed regulations in 10 CFR 50.61a require that the RT_{MAX-X} values be compared to the screening criteria. Because all RT_{MAX-X} values in Table 2 are below the screening criteria, the staff expects that all of these licensees would be able to demonstrate that their reactor vessels will be able to comply with 10 CFR 50.61a, should they choose to implement 10 CFR 50.61a as proposed.

Probabilistic Assumptions

The staff used a systematic process to develop probabilistic assumptions and inputs adopted in the technical basis analyses to identify the significant sources of uncertainty in the calculations and, whenever possible, to address and quantify those sources. The staff applied available, standard probabilistic risk assessment practices to identify and represent uncertainties. Thus, the probabilistic assumptions are consistent, where appropriate, with those used in other risk-informed initiatives.

The proposed rulemaking also considers neutron irradiation embrittlement effects over extended plant lifetimes. The technical basis for this rulemaking evaluates plant fracture toughness against PTS events well beyond a 60-year lifetime.

ACRS

The staff was requested to seek ACRS comment on the most important aspects of the probabilistic basis. The staff has interacted with the ACRS during the development of the technical basis for the rulemaking, as well as prior to requesting Commission approval of the proposed rule. The ACRS, in a letter dated April 10, 2007 (ADAMS Accession No. ML071000105), has decided to defer their review of the proposed rule until after public comments have been considered. However, the ACRS has also asked the staff to plan to discuss the proposed rule with the ACRS subcommittee after publishing the proposed rule in the *Federal Register*. The staff plans to discuss the proposed rule with the ACRS subcommittee and consider their comments with any public comments received on the proposed rule.

<u>SCHEDULE</u>:

The staff plans to publish this proposed rule in the *Federal Register* in August 2007. After consideration of public comments, the staff plans to submit the final rule to the Commission for consideration in March 2008. This schedule was approved by the Executive Director for Operations on January 12, 2007.

<u>RESOURCES</u>:

Total resources required are 2.5 FTE and $50,000 for FY2007 and 1.7 FTE and $35,000 for FY2008. For NRR, approximately 2.1 FTE and $50,000 are needed for this rulemaking for FY 2007 through FY 2008. Of this amount, 1.2 FTE and $50,000 are budgeted for FY2007, and 0.9 FTE is budgeted for FY 2008. For the Office of Nuclear Regulatory Research, 1.5 FTE and $110,000 are needed for this rulemaking for FY 2007 through FY 2008. Of this amount, 1.0 FTE and $75,000 are budgeted for FY2007, and 0.5 FTE and $35,000 are pending approval for FY 2008. For the Office of General Counsel, 0.1 FTE per year is budgeted for FY 2007 through

FY 2008. For the Office of Information Services, 0.1 FTE per year is budgeted for FY 2007 through FY 2008. For the Office of Administration, 0.1 FTE per year is budgeted for FY 2007 through FY 2008.

RECOMMENDATIONS:

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

1. Approve for publication in the *Federal Register* the proposed amendment to 10 CFR Part 50 (Enclosure 1).

2. Certify that this rule, if promulgated, will not have a significant economic impact on a substantial number of small entities in order to satisfy requirements of the Regulatory Flexibility Act (5 U.S.C. 605(b)).

3. Take note of the following:

    a. The proposed rule will be published in the *Federal Register* for a 75-day comment period.

    b. A draft regulatory analysis has been prepared (Enclosure 2).

    c. A draft environmental assessment and finding of no significant impact has been prepared (Section VII of Enclosure 1).

    d. This proposed rule creates new information collection requirements that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 *et seq*.). The staff will submit this rule to the Office of Management and Budget (OMB) for review and approval of the paperwork requirements (Section VIII of Enclosure 1). A draft OMB supporting statement has been prepared (Enclosure 3).

    e. The Chief Counsel for Advocacy of the Small Business Administration will be informed of the certification regarding the economic impact on small entities and the reasons for the certification as required by the Regulatory Flexibility Act (Section XVII of Enclosure 1).

    f. The appropriate Congressional committees will be informed.

    g. The Office of Public Affairs will issue a press release.

COORDINATION:

The Office of the General Counsel has no legal objection to the proposed rule. The Office of the Chief Financial Officer has reviewed the proposed rule for resource implications and has no objections. The Office of Information Services has reviewed the proposed rule and has no objections to the changes in information collection requirements. The ACRS and the Committee to Review Generic Requirements (CRGR) have deferred their review of the proposed rule until after public comments have been considered. The ACRS and CRGR will have the opportunity to review this rulemaking at the final rule stage.

/RA/

Luis A. Reyes
Executive Director
for Operations

Enclosures:
1. Federal Register Notice
2. Draft Regulatory Analysis
3. Draft OMB Supporting Statement

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.

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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 5×10^{-6} per reactor year.

Any reactor vessel with materials predicted to exceed the screening criteria in 10 CFR 50.61 may not continue to operate without implementation of compensatory actions or additional plant-specific analyses unless the licensee receives an exemption from the requirements of the rule. Acceptable compensatory actions are neutron flux reduction, 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 $1x10^{-6}$ 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 $1x10^{-6}$ 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.

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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 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

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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 $RT_{MAX\text{-}CW}$ 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 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 $RT_{MAX\text{-}X}$ 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.

(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.

Enclosure 2

Regulatory Analysis for the Proposed Rule to Amend Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events

Regulatory Analysis for the Proposed Rule to Amend Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events (10 CFR 50.61a)

This document presents a draft regulatory analysis of proposed revisions to the Pressurized Thermal Shock (PTS) Rule as set forth by the U.S. Nuclear Regulatory Commission (NRC) in Title 10, Section 50.61, of the Code of Federal Regulations (10 CFR 50.61). The proposed rule is being undertaken as the result of a June 30, 2006, Staff Requirements Memorandum (SRM). In this SRM, the Commission directed the staff to pursue the rulemaking as described in Option 2 of the May 26, 2006, Commission paper, SECY-06-0124, containing the "Rulemaking Plan to Amend Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events (10 CFR 50.61)."

1.0 Statement of the Problem and Reasons for the Rulemaking:

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

The NRC staff has completed a research program to update the PTS regulations. The results of this research program conclude that the risk of through-wall cracking due to a PTS event is much lower than previously estimated. This finding indicates that the screening criteria in 10 CFR 50.61 are unnecessarily conservative and may impose an unnecessary burden on some licensees. Therefore, the NRC is proposing a new rule, 10 CFR 50.61a. The objective of the proposed rule is to provide alternative screening criteria and corresponding embrittlement correlations for licensees seeking regulatory relief from the overly conservative requirements of the current PTS regulation, 10 CFR 50.61.

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

2.0 Identification of Alternatives

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

Alternative 1: Take no action.

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

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

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

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

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

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

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

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

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

Under Alternative 4, the NRC would promulgate 10 CFR 50.61a as a voluntary alternative to the requirements of 10 CFR 50.61. All PWR licensees would be required to meet the conditions of 10 CFR 50.61, or as a voluntary alternative, would be required to comply with 10 CFR 50.61a. This alternative would not constitute a backfit for any licensee because 10 CFR 50.61a would be voluntarily implemented by any PWR licensee who found it advantageous to do so. PWR licensees who are projected to exceed the 10 CFR 50.61 screening criteria during the lifetime of their plant license would likely comply with

10 CFR 50.61a rather than implement the more expensive compensatory actions specified in 10 CFR 50.61. PWR licensees who are not projected to exceed the 10 CFR 50.61 screening criteria would not likely comply with 10 CFR 50.61a due to the unnecessary cost of implementation.

3.0 Estimation and Evaluation of Values and Impacts

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

3.1 Identification of Affected Attributes

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

- *Industry Implementation*. Implementation of the new rule would require a licensee to submit a license amendment to the NRC for review and approval, requesting that the licensee comply with the proposed rule. This license amendment request would include analyses of the licensee's vessel under the embrittlement correlations and screening criteria in the proposed rule through the plant's end of life (40 or 60 years, as applicable). This analysis is required to demonstrate the licensee's compliance with the new regulation. The licensee would be required to perform analyses of the volumetric

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

examination of the vessel to ensure that the proposed screening criteria and calculation methodology are applicable. Additionally, the licensee would be required to report the manganese and phosphorus content of the reactor vessel beltline materials.

- *Industry Operation*. If implemented, the amended rule would differ from 10 CFR 50.61 only in that, during plant operation, a licensee would be required to perform analyses of the volumetric examination of the vessel to ensure that the proposed screening criteria and calculation methodology are applicable.
- *NRC Implementation*. The NRC would be affected by this proposed rule because of the level of effort required to review and approve license amendment requests to comply with 10 CFR 50.61a.
- *NRC Operation*. The NRC would be required to review a licensee's submittal of the analysis of the volumetric examination inspection results of the vessel under the amendment.
- *Improvements in Knowledge*. The NRC and the nuclear industry would acquire additional data concerning vessel weld flaws due to the additional analyses of the volumetric examination inspection results under 10 CFR 50.61a. Each plant implementing the proposed rule would contribute to improvements in NRC's and industry's knowledge of how well the new PTS rule fracture toughness requirements apply to current reactor vessels. The additional insights gained from these inspections could be used in future research projects, with the potential for further revisions to the PTS rule.
- *Regulatory Efficiency*. The NRC staff is of the opinion that proposing the amendment as a voluntary rule is the more efficient approach. This is accomplished by allowing the licensees to select the option that best serves their situation without any affect on the

public health and safety and common defense and security.

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

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

3.2 Estimation of Values and Impacts

- Industry Implementation

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

- Industry Operation

The projected additional cost to a licensee performing the analysis of the volumetric examination inspection results is estimated at 0.3 FTE, or approximately $46,000, per analysis. This includes performing the analysis and submitting it for review and approval by the NRC. It would be performed with the vessel inspection frequency (currently every 10 years). This regulatory analysis assumes that the rule will take effect in 2008. The timeframe for which

[2] All cost estimates in this analysis are based on the NRC staff's most recent estimate of labor rates to be used in regulatory analyses of $105 per hour or an annual rate of $152,000 assuming 1446 hours worked in a year.

costs are estimated is based on the remaining operating lives of the relevant facilities. For this analysis, costs are estimated over an assumed period of 48 years, with costs discounted at a 7-percent and 3-percent discount rate every 10 years, as specified in NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook." The analysis makes a simplifying assumption that an average plant's next vessel inspection will occur 5 years from the rule's implementation date and every 10 years thereafter through the assumed lifetime of the plant, including license renewal.

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

- NRC Implementation

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

• NRC Operation

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

• Regulatory Efficiency

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

4.0 Presentation of Results

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

4.1 Benefits and Costs

The analyses performed in the technical basis for this rulemaking indicate that the degree of PTS challenge for anticipated lifetimes and operating conditions for current operating

PWRs is low. Further, the U.S. domestic commercial operating fleet of 69 PWRs has a low probability of exceeding either the limit on the maximum estimated mean through-wall crack frequency of $5x10^{-6}$/yr expressed by current PTS regulations or the proposed new value of $1x10^{-6}$/yr, consistent with the Commission's direction in their Staff Requirements Memorandum, SRM-SECY-06-0124, on the PTS Rulemaking Plan. As a result, the risk of PTS events is much lower than previously estimated. Therefore, the screening criteria in 10 CFR 50.61 are considered unnecessarily conservative and may impose unnecessary burden on licensees. These results provide the basis to support a relaxation of the current PTS regulations while continuing to provide adequate protection to public health and safety.

This proposed action, which would be voluntary for all current and future PWR licensees, would result in a burden reduction for some of those licensees with no increase in risk to the public's health and safety.

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

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

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

4.2 Disaggregation

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

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

the goals of the rule and, therefore, complies with the NRC's criteria for the treatment of individual requirements.

5.0 Decision Rationale for Selection of the Proposed Action

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

The NRC staff did not recommend Alternative 2. Under this alternative, all current PWR licensees and all new PWR licensees referencing certified designs would incur additional regulatory burden from the requirement to re-analyze the plant RT_{PTS} values under the new rule. These designs have previously been licensed or certified under the analysis methods and screening criteria defined in 10 CFR 50.61. As described previously, Alternative 2 would constitute a backfit under 10 CFR 50.109 for these licensees. New licensees not referencing a certified design would only perform the analyses required in 10 CFR 50.61a. This would not constitute a backfit because the licensee had not previously been granted approval of the plant design based on 10 CFR 50.61.

The NRC staff did not recommend Alternative 3. For the majority of current and future projected PWR licensees, this alternative would impose no additional regulatory burden to comply with 10 CFR 50.61a because implementation of 10 CFR 50.61a would be voluntary. Although the Commission directed the NRC staff to consider requiring that new reactors be required to comply with the new rule, the NRC staff determined there was no benefit in requiring mandatory implementation for future (non-licensed, non-design certified) licensees. This determination was based on the fact that 10 CFR 50.61 is considered conservative and sufficient. As a result, the NRC staff saw no benefit in requiring implementation of 10 CFR 50.61a for any licensee, regardless of the date of their license or design certification.

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

6.0 Implementation

This action is being published as a proposed rule for public comment in the *Federal Register*. After addressing public comments, the final rule would take effect upon Commission approval and publication of the final rule in the *Federal Register*.

Enclosure 3

Draft OMB Supporting Statement for Proposed Rule: Alternate Fracture Toughness Requirements for Protection Against Thermal Shock Events

DRAFT OMB SUPPORTING STATEMENT FOR
PROPOSED RULE: ALTERNATE FRACTURE TOUGHNESS REQUIREMENTS FOR PROTECTION AGAINST THERMAL SHOCK EVENTS
(10 CFR 50.61 and 50.61a)
(3150-0011)

DESCRIPTION OF THE INFORMATION COLLECTION

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 unless the licensee receives an exemption from the requirements of the rule. Acceptable compensatory actions are neutron flux reduction, plant-specific analyses, 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.

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 proposed rule would be voluntary for all current and future PWR licensees, 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 or applicants 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 alternate PTS rule (10 CFR 50.61a) permits PWR licensees to voluntarily implement the new screening limits and embrittlement correlations based on the updated technical basis. The requirements of the current PTS rule, 10 CFR 50.61, continue to apply to licensees that choose not to implement the new rule.

This rule contains a requirement for licensees to perform analyses of test results from the ASME Boiler and Pressure Vessel Code Section XI inservice inspection volumetric examination. The examination and analyses are to confirm that the flaw density and size in the licensee's reactor pressure vessel beltline are bounded by the flaw density and size in the technical basis.

10 CFR 50.61a(c)(1) requires each PWR licensee to have projected values of RT_{MAX-X}, accepted by the NRC, for each reactor vessel beltline material for the expiration date of the operating license (EOL) fluence of the material. The assessment must (1) use the calculation

procedures specified in 10 CFR 50.61a paragraphs (f)(1) and (g); (2) specify the bases for the projected value, including the assumptions regarding core loading patterns; and (3) specify the copper, phosphorus, manganese and nickel contents and the neutron flux and fluence values and full power cold leg temperature used in the calculation for each beltline material.

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

10 CFR 50.61a(d)(1) requires that licensees submit a re-assessment of RT_{MAX-X} values upon any significant change in the projected values of RT_{MAX-X}, or upon a request for a change in the expiration date for operation of the facility.

10 CFR 50.61a(d)(2) requires that licensees submit a re-analysis demonstrating a TWCF of less than $1x10^{-6}$ per reactor-year or a technical justification as required in 10 CFR 50.61a(e)(4) and (5).

10 CFR 50.61a(d)(3) requires consideration of submission and anticipated approval by the NRC of detailed plant-specific analyses submitted to demonstrate acceptable risk with RT_{MAX-X} above the screening limit due to plant modifications, new information, or new analysis techniques, in conjunction with implementing flux reduction programs that are reasonably practical to avoid exceeding the screening criteria.

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

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

10 CFR 50.61a(e) requires PWR licensees to verify that the screening criteria and calculation methodology are applicable to that particular reactor vessel. The analysis to be provided is based on results of the ASME Code volumetric examination.

10 CFR 50.61a(f)(7) requires PWR licensees to report to NRC any information believed to significantly improve the accuracy of the RT_{MAX-X} values. The burden is included in the estimates for RT_{MAX-X} assessment under Item 12 of this Supporting Statement.

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

A. JUSTIFICATION

1. Need for the Collection of Information

Maintaining the structural integrity of the reactor pressure vessel of light-water-cooled reactors is a critical concern related to the safe operation of nuclear power plants. To assure the structural integrity of reactor vessels, the NRC has developed regulations, including 10 CFR 50.61 and 10 CFR 50.61a, and regulatory guides, including Regulatory Guide 1.99, Revision 2, to provide analysis and measurement methods and procedures to establish that the reactor vessel has adequate safety margin for continued operation. The fracture toughness of the vessel materials varies with time. As the plant operates, neutrons escaping from the reactor core impact the vessel beltline materials causing embrittlement of those materials. The information collections in 10 CFR 50.61 and 10 CFR 50.61a, as well as those in 10 CFR 50.60 and 10 CFR 50 Appendix G and 10 CFR 50 Appendix H, provide estimates of the extent of the embrittlement, and evaluations of the consequences of the embrittlement, in terms of the structural integrity of the vessel. The NRC requires this information to ensure that no reactor, susceptible to the effects of pressurized thermal shock, will continue to operate without putting in place other mitigating measures.

2. Agency Use of the Information

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

3. Reduction of Burden Through Information Technology

There are no legal obstacles to reducing the burden associated with this information collection. The NRC encourages respondents to use information technology when it would be beneficial to them. NRC issued a regulation on October 10, 2003 (68 FR 58791), consistent with the Government Paperwork Elimination Act, which allows its licensees, vendors, applicants, and members of the public the option to make submissions electronically via CD-ROM, e-mail, special Web-based interface or other means. It is estimated that approximately 15% of the potential responses are filed electronically.

4. Effort to Identify Duplication and Use Similar Information

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

5. Effort to Reduce Small Business Burden

The requirements in this rule do not affect small businesses.

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

If this information, in combination with the information collection associated with 10 CFR 50.61, were not collected, the NRC could not verify that each reactor pressure vessel has an adequate safety margin for continued safe operation.

7. Circumstances Which Justify Variations from OMB Guidelines

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

8. Consultations Outside the NRC

The opportunity for public comment on this information collection has been published in the *Federal Register.*

9. Payment or Gift to Respondents

Not applicable.

10. Confidentiality of Information

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

11. Justification for Sensitive Questions

No sensitive information is requested in this rule.

12. Estimated Industry Burden and Burden Hour Cost

Currently Operating Pressurized Water Reactors

The requirements in 10 CFR 50.61a will only apply to those licensees that voluntarily choose compliance with this section as an alternative to compliance with the requirements specified in 10 CFR 50.61. Of the 69 currently operating PWRs, the staff projects that eight reactor vessels could exceed the screening criteria specified in 10 CFR 50.61 during their extended (60 year) lifetimes. The NRC expects that each of these licensees will elect to apply the less stringent embrittlement correlations and screening criteria in 10 CFR 50.61a rather than applying the compensatory measures of 10 CFR 50.61(b)(3) through (b)(7). The NRC assumes that, subsequent to the effective date of the final rule, one operating reactor licensee per year will choose to comply with 10 CFR 50.61a for the following eight years.

Thus, in the three years following the effective date of this rule, three operating reactors would be affected by the RT_{MAX-X} assessment; none would perform the flux reduction analyses, and none would perform the reactor vessel thermal annealing. The number of annual responses to the NRC is expected to be two (one response for the RT_{MAX-X} assessment and one response for the analysis of ASME BPV inservice ultrasonic testing results) and the estimated number of annual respondents is also expected to be one.

(1) RT_{MAX-X} assessment - The NRC estimates that the reporting burden would be 120 staff hours per plant. Thus the annualized burden over three years would be 3 plants x 120 hours per plant ÷ 3 years, or 120 staff hours per year. The recordkeeping burden is expected to be approximately 10% of the reporting burden and is estimated to be 3 plants x 12 hours per plant ÷ 3 years or 12 staff hours per year

(2) Flux reduction analyses - None expected.

(3) Safety analysis - None expected.

(4) Reactor vessel thermal annealing - None expected.

(5) Analysis of ASME BPV inservice ultrasonic testing results. For the purpose of this supporting statement, the NRC is assuming that the reporting and record keeping burden for this requirement is the same as for the reporting and recordkeeping requirements for the RT_{MAX-X} assessment (*i.e.*, 120 hours per year for reporting and 12 hours per year for recordkeeping) for the three current licensees expected to voluntarily implement the new rule over the next three years.

The total estimated annual industry burden for reporting would be approximately 240 hours or $52,080 (240 hours X $217 per hour) per year over the next 3 years.[1] years. The total estimated annual industry burden for recordkeeping would be approximately 24 hours or $5,208 (24 hours X $217 per hour) per year over the next 3 years.

<u>New Combined License Applications</u>

The NRC is currently estimating that in the next several years it will receive 20 combined license applications for 29 new reactor units, 20 of which are expected to be PWRs. However, the requirements in 10 CFR 50.61a are voluntary, and they are less familiar to licensees than the current 10 CFR 50.61 requirements. Also, while the requirements in 10 CFR 50.61a are less restrictive, there are additional requirements to perform and document,. Therefore, the NRC believes that all new COL applicants will choose to comply with the requirements of 10 CFR 50.61 rather

[1]The information collection burden for the three plants discussed here is reduced from the information collection burden for 10 CFR 50.61 to avoid double counting. That information burden is reduced by 120 hours per year.

than the voluntary alternate requirements of 10 CFR 50.61a. Therefore, the following are projected to apply:

(1) RT_{MAX-X} assessment - The NRC estimates that this information collection burden would be 120 staff hours per plant. The annualized burden over three years would be 0 plants X 120 hours per plant ÷ 3 years, or 0 staff hours.

(2) Flux reduction analyses - None expected.

(3) Safety analysis - None expected.

(4) Reactor vessel thermal annealing - None expected.

(5) Analysis of ASME BPV inservice volumetric testing results- None expected.

The total estimated annual industry burden for this rulemaking would be approximately 264 hours (240 hours for reporting and 24 hours for recordkeeping) or $57,288 (264 hours X $217 per hour) per year over the next 3 years.

13. <u>Estimate of Other Additional Costs</u>

The quantity of records to be maintained is roughly proportional to the record keeping burden and therefore can be used to calculate approximate records storage costs. Based on the number of pages maintained for a typical clearance, the records storage cost has been determined to be .0004 times the record keeping burden cost. Therefore, the storage cost of this clearance is insignificant (24 recordkeeping hours x $217/hr.x .0004 = $2).

14. <u>Estimated Annualized Cost to the Federal Government</u>

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

(1) RT_{MAX-X} Assessment: The staff estimates that reevaluations of RT_{MAX-X} values will be submitted by 3 PWR licensees within the 3-year clearance period. On average, 40 hours are estimated for the review of each submittal. Total review time is estimated at 120 staff hours at an estimated cost of x $26,040 (3 submittals x 40 hours/submittal x $217/hour) over the 3-year clearance period. Thus, the estimated annualized burden is 40 hours at a cost of $8,680.

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

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

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

(5) The estimated total annual federal cost, which is the sum of items (1) through

(4) above, is $8,680.

15. Reasons for Changes in Burden or Cost

    The only change in burden is incurred by those licensees choosing to voluntarily implement 10 CFR 50.61a, which includes an additional evaluation of ASME BPV inservice volumetric testing results. The base burden cost changes from $156 to $217 per hour.

16. Publication for Statistical Use

    The collected information is not published for statistical purposes.

17. Reason for Not Displaying the Expiration Date

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

18. Exceptions to the Certification Statement

    None.

B. COLLECTIONS OF INFORMATION EMPLOYING STATISTICAL METHODS

Not applicable.