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. <u>Effort to Reduce Small Business Burden</u>

   The requirements in this rule do not affect small businesses.

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

   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. <u>Circumstances Which Justify Variations from OMB Guidelines</u>

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

8. <u>Consultations Outside the NRC</u>

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

9. <u>Payment or Gift to Respondents</u>

   Not applicable.

10. <u>Confidentiality of Information</u>

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

11. <u>Justification for Sensitive Questions</u>

    No sensitive information is requested in this rule.

12. <u>Estimated Industry Burden and Burden Hour Cost</u>

    <u>Currently Operating Pressurized Water Reactors</u>

    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. Estimate of Other Additional Costs

The quantity of records to be maintained is roughly proportional to the record keeping burden and therefore can be used to calculate approximate records storage costs. Based on the number of pages maintained for a typical clearance, the records storage cost has been determined to be .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. Estimated Annualized Cost to the Federal Government

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

(1) RT_{MAX-X} Assessment: The staff estimates that reevaluations of RT_{MAX-X} values will be submitted by 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.