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

RIN 3150-AI01
[NRC-2007-0008]

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

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

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

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

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

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

(1) Embrittlement Trend Curves and Fluence Maps

(2) Surveillence Data

(3) Flaw Limits and Flaw Density Determinations

(4) Adjustments of Volumetric Examination Data

(5) Miscellaneous

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

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

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

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

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

COMMENTS RELATED TO THE PROPOSED RULE LANGUAGE

Comments Related to Embrittlement Trend Curves and Fluence Maps

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Comments Related to Surveillance Data

COMMENT: Eliminate the requirement to assess surveillance data, including Table 5, of the proposed rule. There is little added value in the requirement to assess the surveillance data as a part of this rule because variability in data has already been accounted for in the derivation of the embrittlement correlation.

The commenters also stated that there is no viable methodology for adjusting the projected ΔT_{30} for the vessel based on the surveillance data. Any effort to make this adjustment is likely to introduce additional error into the prediction. Note that the embrittlement correlation described in the basis for the revised PTS rule (i.e., NUREG-1874) was derived using all of the currently available industry-wide surveillance data.

In the event that the surveillance data does not match the ΔT_{30} value predicted by the embrittlement correlation, the best estimate value for the pressure vessel material is derived using the embrittlement correlation. The likely source of the discrepancy is an error in the characterization of the surveillance material or of the irradiation environment. Therefore, unless the discrepancy can be resolved, obtaining the ΔT_{30} prediction based on the best estimate chemical composition for the heat of the material is more reliable than a prediction based on a single set of surveillance measurements. [PWROG-4, EPRI-4, NEI-2]

NRC RESPONSE: The NRC does not agree with the proposed change. The NRC believes that there is added value in the requirement to assess surveillance data. Although variability has been accounted for in the derivation of the embrittlement correlation, it is the NRC's view that the surveillance assessment required in § 50.61a(f)(6) is needed to determine if the embrittlement for a specific heat of material in a reactor vessel is consistent with the embrittlement predicted by the embrittlement correlation.

The commenters also assert that there is no viable methodology for adjusting the projected ΔT_{30} for the vessel based on the surveillance data, and that any adjustment is likely to introduce additional error into the prediction. The NRC believes that although there is no single methodology for adjusting the projected ΔT_{30} for the vessel based on the surveillance data, it is possible, on a case-specific basis, to justify adjustments to the generic ΔT_{30} prediction. For this reason the rule does not specify a method for adjusting the ΔT_{30} value based on surveillance data, but rather requires the licensee to propose a case-specific ΔT_{30} adjustment procedure for review and approval from the Director of NRR. Although the commenters assert that it is possible that error could be introduced, it is the NRC view that appropriate plant-specific adjustments based upon available surveillance data may be necessary to project reactor pressure vessel embrittlement for the purpose of this rule.

As the result of these public comments, the NRC has continued to work on statistical procedures to identify deviations from generic embrittlement trends, such as those described in § 50.61a(f)(6) of the proposed rule. Based on this work, the NRC further enhanced the procedure described in § 50.61a(f)(6) to, among other things, detect signs from the plant- and heat-specific surveillance data that may emerge at high fluences of embrittlement trends that are not reflected by Equations 5, 6, and 7. The empirical basis for the NRC's concern regarding the potential for un-modeled high fluence effects is described in documents located at ADAMS Accession Nos. ML081120253, ML081120289, ML081120365, ML081120380, and ML081120600. The technical basis for the enhanced surveillance assessment procedure is described in the document located at ADAMS Accession No. ML081290654. ML081120380, and ML081120600. The technical basis for the enhanced surveillance assessment procedure is described in the document located at ADAMS Accession No. ML081290654.

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

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

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

Comments Related to Flaw Limits and Flaw Density Determinations

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

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

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

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

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

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

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

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

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

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

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

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

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

The commenters requested that the NRC explain:

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

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

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

The commenters suggested that:

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

NRC RESPONSE: The NRC disagrees that Tables 2 and 3 fail to account for the three issues identified in the comment. Tables 2 and 3 were created to address concerns that the flaws in a plant-specific vessel might not be bounded consistent with the inputs and assumptions used in the FAVOR code calculations and that, in turn, might cause the TWCF to exceeded 1×10^{-6} per year if the RT_{MAX-X} limits in Table 1 are reached. The number and sizes of flaws in Tables 2 and 3 are to be compared to the number and sizes of flaws from the ASME Code volumetric examination of the reactor vessel welds.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

The NRC believes that guidance on whether the plate embedded flaw limits in Table 3 can be applied for forgings has already been provided. However, a footnote in § 50.61a(e)(1) has been added to further clarify that for forgings susceptible to underclad cracking the determination of the flaw density for that forging from the licensee's inspection shall exclude those indications identified as underclad cracks.

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

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

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

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

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

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

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

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

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

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

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

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

<u>NRC RESPONSE</u>: The NRC agrees that intended examination volume was not clearly described in the rule. Therefore, § 50.61a(e)(1) has been revised to indicate that the volumetric examination volume is the volume described in ASME Code, Section XI, Figures IWB-2500-01 and IWB-2500-02 and limited to a depth from the clad-to-base metal interface of 1-inch or 10 percent of the vessel thickness, whichever is greater.

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

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

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

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

Miscellaneous Comments

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

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

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

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

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

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

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

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

effectively zero, so any sequence with a non-zero conditional probability of a through-wall crack (about $1x10^{-11}$ per reactor year) makes a large contribution to the total value. Therefore, the scenario is a dominant contributor to TWCF at low embrittlement levels, but only because the TWCF is very small. However, at embrittlement levels close to the proposed 10 CFR 50.61a screening limit the sequence contribution (of about $1x10^{-8}$) to the total TWCF of $1x10^{-6}$ per year is insignificant because the relatively high conditional probability of a through-wall crack is more than offset by the very low frequency of the sequence.

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

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

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

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

NRC RESPONSE: The NRC disagrees that the mean values of TWCF should be used instead of the 95^{th} percentile upper bounds, whether the technical basis is redone or not. There are several differences between the risk metrics in RG 1.174 and the $1x10^{-6}$ per reactor year TWCF criteria. For example, the RG 1.174 acceptance guidelines identify increases in risk that would normally be acceptable but the TWCF is a total estimated frequency. The risk guidelines from RG 1.174 are discussed briefly in the TWCF technical basis in order to provide a reference to a quantitative frequency that is normally considered acceptably small for an undesired event. In addition to the difference in the estimated parameter, the PTS analysis is an evaluation to support a change to a regulation which is subject to Commission review and approval and to public comment. Consequently, the PTS work need only follow the RG guidance to the extent that the NRC determines that the RG guidance is applicable and appropriate. During the development of the PTS RT_{MAX-X} limits, the NRC noted that the highly skewed distribution resulted in mean values that lay in the higher percentile values and were always greater than the 75^{th} percentile. The NRC concluded that the 95^{th} percentile upper bound is preferable to the mean because more consistent RT_{MAX-X} limits could be derived by using the fixed percentile than by using the mean value whose percentile changes with the embrittlement level.

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

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

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

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

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

COMMENT: Add a statement that states that this rule is applicable to the current PWR fleet and not the new plant designs. The rule, as written, is only applicable to the existing fleet of PWRs. The characteristics of advanced PWR designs were not considered in the analysis. [PWROG-5, EPRI-5]

NRC RESPONSE: The NRC agrees with the comment that this rule is only applicable to the existing fleet of PWRs. The NRC cannot be assured that plants whose construction permit was issued after the effective date of the final rule and whose reactor vessel was designed and fabricated to ASME Code Editions later than the 1998 Edition will have material properties, operating characteristics, PTS event sequences and thermal-hydraulic responses consistent with the reactors that were evaluated as part of the technical basis for § 50.61a. Other factors, including materials of fabrication and welding methods, would also be consistent with the underlying technical basis of 10 CFR 50.61a. As a result of this comment, the NRC modified § 50.61a(b) and the statement of considerations of the rule to reflect this position to allow the use of the rule only to plants whose construction permit was issued before the effective date of the final rule and whose reactor vessel was designed and fabricated to the 1998 Edition or earlier of the ASME Code.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

COMMENTS RELATED TO THE SUPPLEMENTAL PROPOSED RULE

Comments Related to Adjustments of Volumetric Examination Data

COMMENT: Modify § 50.61a(e) to allow licensees to account for the effects of flaw sizing uncertainties and other uncertainties in meeting the requirements of Tables 2 and 3. The rule language should allow the use of applicable data from ASME qualification tests, vendor specific performance demonstration tests, and other current and future data that may be applicable for assessing these uncertainties. The rule language should permit flaw sizes to be adjusted to account for the sizing uncertainties and other uncertainties before comparing the estimated size and density distribution to the acceptable size and density distributions in Tables 2 and 3. [PWROG2-1 and EPRI2-1]

The industry will provide guidance to enable licensees to account for the effects of sizing uncertainties and other uncertainties in meeting the requirements of Tables 2 and 3 of the rule. Guidance to ensure that the risk associated with PTS is acceptable will be provided to the Director of NRR for review and approval when completed.

NRC RESPONSE: The NRC agrees that, in addition to the NDE sizing uncertainties, licensees should be allowed to consider other NDE uncertainties (e.g., probability of detection, flaw density and location) in meeting the requirements of the rule as these uncertainties may affect the ability of a licensee to demonstrate compliance with the rule. As a result, the language in § 50.61a(e) was modified to allow licensees to account for the effects of NDE-related uncertainties in meeting the flaw size and number requirements of Tables 2 and 3. This requirement would be accomplished by requiring licensees to base the methodology to account for the NDE uncertainties on statistical data collected from ASME Code inspector qualification tests and any other tests that measure the difference between the actual flaw size and the size determined from the ultrasonic examination. Collecting, evaluating, and using data from these tests will require extensive engineering judgment. Therefore, the methodology would have to be reviewed and approved by the Director of NRR.

Lastly, the commenters proposed to provide industry guidance to enable licensees to account for the effects of NDE uncertainties to the Director of NRR. The NRC determined that the rule language clearly states the information that must specifically be provided for NRC review and approval if licensees choose to account for NDE uncertainties. However, if industry guidance documents are developed, the NRC will consider them when submitted for review and approval.

Comments Related to Surveillance Data

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

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

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

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

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

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

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

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

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

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

The second required surveillance check is to address a slope change. The intent of this section appears to be to identify potential increases in the embrittlement rate at high fluence. The industry intends to move forward with an initiative to populate the power reactor vessel surveillance program database with higher neutron fluence surveillance data (i.e., extending to fluence values equivalent to 60-80 EFPY) that will adequately cover materials variables for the entire PWR fleet. This database should provide a more effective means of evaluating the potential for enhanced embrittlement rates at high fluence values than using an individual surveillance data set to modify the trend with fluence. Data from this initiative will be available in the next few years to assess the likelihood of enhanced embrittlement rates for the PWR fleet. [PRWOG2-3c and EPRI2-3c]

NRC RESPONSE: The NRC does not agree with the commenters' statement that the slope test (i.e., § 50.61a(f)(6)(iii)) has limited value and that it should be eliminated from the rule. The NRC believes that the slope test provides a method for determining whether high neutron fluence surveillance data is consistent with the ΔT_{30} model in the rule. Since there are currently only a few surveillance data points from commercial power reactors at high neutron fluences and the slope test will provide meaningful information, the NRC determines that the slope test should not be eliminated from the rule.

The NRC agrees with the industry initiative to obtain additional power reactor data at higher fluences. The NRC will review this data and the information available to evaluate the effects of high neutron fluence exposure when it becomes available. At that point, the NRC will determine if modifications to the embrittlement model and/or the surveillance data checks in § 50.61a should be made.

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