[7590-01-P]

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

RIN: 3150 - AD57

Fracture Toughness Requirements For Light Water Reactor Pressure Vessels

AGENCY: Nuclear Regulatory Commission.

ACTION: Proposed rule.

SUMMARY: The Nuclear Regulatory Commission (NRC) is proposing to amend its regulations for light-water-cooled nuclear power plants to clarify several items related to the fracture toughness requirements for reactor pressure vessels (RPV). The proposed amendments would clarify the pressurized thermal shock (PTS) requirements, make changes to the Fracture Toughness Requirements and the Reactor Vessel Material Surveillance Program Requirements, and provide new requirements for thermal annealing of a reactor pressure vessel.

DATE: The comment period expires (90 days from publication date). Comments received after this date will be considered if it is practical to do so, but the Commission is able to assure consideration only for comments received on or before this date.

ADDRESSES: Mail comments to: The Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Service Branch.

Comments may be delivered to: One White Flint North, 11555 Rockville Pike, Rockville, MD 20852, between 7:30 am and 4:15 pm on Federal workdays. Comments received on the proposed rules may be examined at the NRC Public Document Room, 2120 L Street NW. (Lower Level), Washington, DC.

A free single copy of draft regulatory guides DG-1023, DG-1025, and DG-1027 may be requested by those considering public comment by writing to the U.S. Nuclear Regulatory Commission, ATTN: Distribution and Mail Services Section, Room P-130A, Washington, DC 20555. A copy is also available for inspection and/or copying in the NRC Public Document Room, 2120 L Street NW. (Lower Level), Washington, DC.

FOR FURTHER INFORMATION CONTACT: Alfred Taboada, Division of Engineering, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone: (301) 415-6014.

SUPPLEMENTARY INFORMATION:

Background

Maintaining the structural integrity of the reactor pressure vessel (RPV) of light-water-cooled reactors is a critical concern related to the safe operation of nuclear power plants. To assure the structural integrity of RPVs, NRC regulations and regulatory guides have been developed to provide analysis and measurement methods and procedures to assure that each RPV has adequate safety margins for continued operation. Structural integrity of a reactor pressure vessel is generally assured through a fracture mechanics evaluation, including measurement or estimation of the fracture toughness of the materials which compose the RPV. However, the fracture toughness of the RPV materials varies with time. As the plant operates, neutrons escaping from the reactor core impact the vessel beltline materials (e.g. the materials that

surround the reactor core), causing embrittlement of those materials. The NRC's regulations and regulatory guides related to RPV integrity provide the criteria and methods needed to estimate the extent of the embrittlement, to evaluate the consequences of the embrittlement in terms of the structural integrity of the RPV, and to provide methods to mitigate the deleterious effects of the embrittlement.

The NRC has several regulations and regulatory guides that establish criteria and procedures for assuring the structural integrity of RPVs. With the addition of a proposed rule on thermal annealing and several draft regulatory guides, the existing and proposed regulatory documents contribute to a comprehensive set of regulations and regulatory guidance pertaining to RPV integrity, including:

1. The fracture toughness criteria that each RPV must satisfy (10 CFR 50.60, 10 CFR 50.61, and 10 CFR Part 50, Appendix G).
2. Irradiation embrittlement surveillance requirements (10 CFR Part 50, Appendix H).
3. Guidance for estimating the fracture toughness of the RPV (Regulatory Guide 1.99 and a draft regulatory guide on dosimetry).
4. Guidance for cases in which the RPV is estimated to exceed specified screening criteria (Regulatory Guide 1.154 and a draft regulatory guide on evaluating RPVs with Charpy upper-shelf energy less than 50 ft-lb).
5. Requirements and guidance for using thermal annealing of the RPV as a method for mitigating the effects of neutron irradiation (proposed 10 CFR 50.66 and a draft regulatory guide).

This notice proposes changes to the following requirements:

a. The Pressurized Thermal Shock (PTS) rule (10 CFR 50.61).

b. Appendix G of 10 CFR Part 50, "Fracture Toughness Requirements."

c. Appendix H of 10 CFR Part 50, "Reactor Vessel Material Surveillance Program Requirements."

This notice also proposes a thermal annealing requirement, 10 CFR 50.66.

In addition to the proposed amendments addressed in this document, the NRC is publishing for public comment a draft regulatory guide on thermal annealing, DG-1027 (see the "Notices" section of this Federal Register for a document announcing the availability of this draft regulatory guide for comment).

Two related draft regulatory guides have been published for public comment (58 FR 51392; October 1, 1993). These draft regulatory guides are:

1. A draft regulatory guide that addresses evaluation of RPVs with Charpy upper-shelf energy levels less than 50 ft-lb (DG-1023).
2. A draft regulatory guide on dosimetry (DG-1025).

Other regulatory guides pertaining to RPV integrity, Regulatory Guides 1.99 and 1.154, are under evaluation. Revisions to these guides, if any, will be addressed in the future.

Other regulatory issues related to reactor pressure vessel integrity, such as low temperature over-pressure protection system setpoints, are being addressed as part of a broader scope evaluation of the pressure vessel regulations and are not part of this proposed amendment.

Reasons for the Proposed Changes

The needs for these proposed amendments to the fracture toughness regulations have been identified from three sources:

1. The 1989 Nuclear Utility Backfitting and Reform Group (NUBARG) appeal concerning use of nuclear heat to warm the RPV for system leakage and hydrostatic pressure tests;
2. The 1990 review of the RPV integrity of the Yankee Nuclear Power Station; and
3. A comprehensive review of the regulations by NRC staff, resulting in the identification of the need for clarifications, corrections, and improved guidance in certain areas.

The recognition in 1986 by NRC staff that certain boiling water reactor (BWR) units were using nuclear heat to warm the system prior to performing leakage and pressure tests led to an NRC staff initiative to amend Appendix G to 10 CFR Part 50. During the NRC staff review to determine if this use of nuclear heating was permissible under either the ASME Code or the NRC regulations, the Nuclear Utility Backfitting and Reform Group (NUBARG) filed a backfitting claim, and later an appeal of the determination that a backfit was not involved. Stemming from this claim and appeal process, the Committee to Review Generic Requirements recommended to the Executive Director for Operations that the affected portions of Appendix G be revised to clearly indicate that all required leakage and pressure tests of the reactor pressure vessel must be performed when the core is not critical.

In 1990, the NRC began a review of the integrity of the Yankee Nuclear Power Station (YNPS) RPV. That review, along with stated plans by the licensee to consider thermal annealing of the RPV, highlighted the need for the NRC to amend its regulations and guidance pertaining to RPV integrity. The NRC staff proposed a plan to revise and clarify the pertinent regulations

in SECY-91-333 (October 22, 1991) and SECY-92-283 (August 14, 1992), including schedules and general descriptions of the changes contemplated. The proposed changes included clarifications and corrections planned prior to the YNPS review. However, the YNPS review identified the need to clarify the requirements in Sections IV and V of Appendix G to 10 CFR Part 50, and the need to provide more complete requirements and guidance for thermal annealing.

The PTS rule, 10 CFR 50.61, was amended on May 15, 1991 (56 FR 22300) to make the method for evaluating irradiation embrittlement consistent with the recommended procedures of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." Subsequent inquiries to the Commission concerning the appropriate margin terms and use of surveillance data indicated that the PTS rule required clarification. A recent review of the rule by NRC staff concluded that the PTS rule should also be modified to bring the procedures for evaluating RT_{PTS} into complete agreement with the recommended procedures in Regulatory Guide 1.99, Revision 2.

Overview of the Proposed Changes

PTS Rule (10 CFR 50.61)

The pressurized thermal shock rule, 10 CFR 50.61, was initially published in final form on July 23, 1985 (50 FR 29937) and amended on May 15, 1991 (56 FR 22300). This rule provides a screening criterion for irradiation embrittlement of RPV beltline materials, above which the plant cannot continue to be operated without justification. Historically, a value of reference temperature has been determined for each vessel beltline material for comparison to the PTS screening criteria. These values of reference temperature are termed RT_{PTS} values. However, the method for evaluating RT_{PTS} values has not been consistent with the embrittlement estimates used for other

purposes, such as pressure-temperature limit calculations. The May 15, 1991, amendment was a step towards unifying the embrittlement estimate methodology. The amendment included the procedures given in Regulatory Guide 1.99, Revision 2, for the evaluation of irradiation embrittlement of the RPV beltline materials. The 1991 amendment left two differences between the rule and Regulatory Guide 1.99, Revision 2. These two differences are:

1. Values of unirradiated RT_{NDT} are specified for general classes of material in the PTS rule, while greater flexibility in determining unirradiated values is permitted in Regulatory Guide 1.99, Revision 2; and
2. The margin terms used in the PTS rule are based on assumptions which are not consistent with the method used in Regulatory Guide 1.99, Revision 2 for calculating the margin term.

This proposed amendment is intended to make the evaluation of RT_{PTS} consistent with the recommended methods of Regulatory Guide 1.99, Revision 2, which are used to evaluate RT_{NDT}. In this case, the RT_{PTS} value for each vessel beltline material is simply the RT_{NDT} value estimated for the projected end of license fluence.

This proposed amendment to the PTS rule would make three changes:

1. The Regulatory Guide 1.99, Revision 2, method for determining RT_{NDT}, of which RT_{PTS} is a unique value determined for the end of license fluence, would be incorporated in total, including treatment of the unirradiated RT_{NDT} value, the margin term, and the explicit definition of "credible" surveillance data.
2. The section would be restructured to improve clarity, with the requirements section giving only the requirements for the RT_{PTS} value. The method for calculating RT_{PTS} would be moved to a new paragraph of the rule.

3. Thermal annealing would be introduced as a method for mitigating the effects of neutron irradiation, thereby reducing RT_{PTS}.

Additionally, it should be noted that evaluations of current surveillance data have indicated that the standard deviation of the differences between predicted and measured shifts in RT_{NDT}, termed the residual, are higher than the margin values used in the PTS rule and in Regulatory Guide 1.99, Revision 2, particularly for plate materials. However, the mean embrittlement estimation equations in the rule and in Regulatory Guide 1.99, Revision 2, overestimate the actual surveillance data by less than 10°F on average. The NRC staff considered amending the PTS rule to incorporate the revised margin terms and the overestimation bias, but decided against such an amendment due to the small number of plants that would be affected by such a change, and a related NRC research program that addresses the overall issue of irradiation embrittlement correlations. The number of plants which would have their RT_{PTS} values change "significantly" by such a change to the margin terms is not large; the impact of the revised margin terms on those plants is being addressed through other regulatory mechanisms. The effect of the revised margin on pressure-temperature limits is being handled in a similar manner.

As noted, this proposed amendment to 10 CFR 50.61 introduces thermal annealing of the reactor pressure vessel beltline as a method for mitigating the effects of neutron irradiation and reducing RT_{PTS} to levels below the screening criteria. As specified in § 50.61(b)(7) of this proposed rule, the use of thermal annealing would be subject to the requirements of the proposed new section on thermal annealing (10 CFR 50.66).

Thermal Annealing Rule (10 CFR 50.66)

The proposed thermal annealing rule, 10 CFR 50.66, would provide a consistent set of requirements for the use of thermal annealing to mitigate the effects of neutron irradiation. The proposed rule would replace the requirements for annealing in the current Appendix G of 10 CFR Part 50 with the proposed consistent set of requirements in this proposed rule. Also, the PTS rule would be amended to add a new paragraph (b)(7) which would reference the proposed thermal annealing rule as a method for mitigating the effects of neutron irradiation, thereby reducing RT_{PTS}. Therefore, the intent of the thermal annealing rule, and related changes in 10 CFR 50.61 and Appendix G of 10 CFR Part 50, is to provide requirements for use of thermal annealing to mitigate the deleterious effects of neutron irradiation on reactor vessel material properties.

Consistent with guidance in Section V.D of the current Appendix G, the proposed thermal annealing rule would specify that thermal annealing would be subject to the approval of the Director, Office of Nuclear Reactor Regulation (NRR). Section 50.66(a) of the proposed thermal annealing rule would require submittal of an application containing three sections: a thermal annealing operating plan, a requalification inspection and test program, and a fracture toughness recovery and reembrittlement rate assurance program. This application would be required to be submitted at least three years before the proposed date of the annealing operation. This three-year period is specified only to provide NRC staff with sufficient time to review the thermal annealing application. The licensee may initiate the thermal annealing program as soon as NRC approval is given, even if this approval is given before three years from the date of the application.

The thermal annealing operating plan also must include an evaluation of the effects of temperature, and of mechanical and thermal stresses on the

reactor and associated equipment to demonstrate that the operability of the reactor will not be detrimentally affected. The temperatures and times used in this analysis define the proposed annealing conditions. If these conditions are exceeded during the vessel annealing, then the evaluation would no longer be valid, and the acceptability of the actual vessel annealing would have to be demonstrated.

Upon completion of the thermal annealing and before subsequent operation of the plant, the licensee would be required to certify that the thermal annealing was performed in accordance with the approved application, that the approved criteria were satisfied, and that the proposed annealing conditions were not exceeded. However, in the event that the licensee cannot make this certification, a justification for subsequent operation would have to be submitted for approval by the Director, NRR. However, this provision does not relieve the licensee from obtaining 10 CFR 50.12 exemptions from any other requirements of this part that cannot be satisfied.

Two items of particular importance to the overall annealing are the recovery of fracture toughness and the rate of reembrittlement of the RPV beltline materials. This proposed rule provides three alternative methods for determining these values, ranging from assessments using plant-specific materials to an assessment using a generic computation.

Two methods for evaluating annealing recovery are experimental methods to determine plant-specific annealing recovery, and the third method is a generic computational method. The first method would be required for those plants with "credible" surveillance programs and where surveillance specimens are available. The second method would be an optional method, in which the licensee may remove material from the beltline of the RPV to evaluate annealing recovery. This method should provide the most accurate evaluation of annealing recovery. Presumably, it would be selected for those plants

without credible surveillance programs or when surveillance specimens are not available. However, for this method to be acceptable, the vessel must be sufficiently thick so that the stress limits in Section III of the ASME Code can be satisfied, considering the effects of fatigue and corrosion.

The third method would use generic computational methods, for which appropriate justification would be required.

Paragraph (d) of §50.66 describes the experimental methods and the computational method for estimating recovery of RT_{NDT} and Charpy upper-shelf energy of the beltline materials. The experimental methods for estimating recovery of RT_{NDT} and the Charpy upper-shelf energy utilize either surveillance program specimens or material removed from the vessel beltline. The experimental methods provide a plant-specific estimate of recovery, rather than the generic value evaluated from the computational method. It is the intent of this proposed rule to require that surveillance specimens from "credible" surveillance programs must be used to develop plant-specific recovery data, if such specimens are available. It is not the intent of this rule to require the removal of material from the RPV beltline to permit plant-specific evaluation of recovery.

As described previously, the computational method would require appropriate justification.

Reembrittlement trends are estimated, and monitored by continued surveillance in accordance with Appendix H of 10 CFR Part 50.

Paragraph (b)(3)(ii) provides that the reembrittlement rate must be monitored using a surveillance program which conforms to Appendix H of this part. Some older plants conform to Appendix H by applying issues of ASTM Standard E 185 that do not require the use of the vessel "limiting materials" in the surveillance program. Within this context, the term "limiting materials" refers to the materials predicted to have the highest RT_{NDT} or the

lowest Charpy upper shelf energy during the operational lifetime of the plant. It is the intent of this rule that, as required by later issues of ASTM Standard E 185, the vessel "limiting materials" should be used to monitor reembrittlement if the materials are available.

Appendix G of 10 CFR Part 50

Appendix G of 10 CFR Part 50 specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light-water-cooled nuclear power reactors. These requirements provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests. The proposed amendments to Appendix G are principally of a clarifying or a restructuring nature. These amendments include:

1. Sections IV and V of Appendix G which would be combined and rewritten to clarify the Charpy upper-shelf energy requirements, and the pressure-temperature and minimum permissible temperature requirements.
2. An explicit statement that would be added to Section IV requiring that pressure and leak tests of the reactor pressure vessel required by Section XI of the American Society of Mechanical Engineers Boiler & Pressure Vessel (B&PV) Code (ASME Code) must be completed before the core is critical.
3. The proposed thermal annealing rule, 10 CFR 50.66, that would be referenced in lieu of the details on thermal annealing previously given in Section V.D.
4. The reference to the ASME Code that would be changed from Appendix G of Section III to Appendix G of Section XI of the ASME Code.

5. The "design to permit annealing" requirement (Section IV.B), which would be deleted.

The restructuring of Sections IV and V is intended to promote clarity of the requirements in these sections. The procedures required for cases in which the Charpy upper-shelf energy of a RPV beltline material falls below 50 ft-lb also would be clarified by consolidating the requirements previously addressed in parts of Sections IV and V.

The provisions in Section V.C concerning requirements for "volumetric inspection" and "additional evidence of fracture toughness" would be removed. The volumetric examination requirement would be removed because it was unnecessary, given the inspection and performance demonstration programs currently required under 10 CFR 50.55a. The "additional evidence of fracture toughness" requirement in Section V.C.2 would be incorporated in the "equivalent margins" analysis in Section IV.A.1, as a provisional method for developing fracture toughness data needed for that analysis. At the present time there is an adequate generic fracture toughness data base available to perform these analyses, with appropriate bounding considerations. The modification would permit a licensee to develop plant-specific data. Generally, plant-specific data would result in a reduction in the margin applied to the fracture toughness data, to reflect the reduction in uncertainties due to the availability of plant-specific data. However, this must be evaluated on a case-by-case basis.

The pressure-temperature and minimum permissible temperature requirements in Section IV would be restructured, with the principal feature being the addition of a table which summarizes the pressure-temperature limit requirements and minimum temperature requirements as a function of the plant operating condition, the vessel pressure, whether fuel is in the vessel, and whether the core is critical. In addition, Section IV would be reworded to

clarify the minimum permissible temperature requirement by indicating the criteria for use in determining the location in the component or material which must satisfy the minimum temperature requirement. This minimum temperature is defined in Section IV as the metal temperature of the controlling material in the region which has the least favorable combination of stress and temperature for the appropriate plant condition.

The requirement that all pressure and leak tests of the RPV required by Section XI of the ASME Code must be completed before the core is critical is intended to prohibit the use of nuclear heat, i.e., core criticality, before the completion of these tests. The use of nuclear heat before the completion of such tests is considered unsafe for several reasons, including the hindrance of finding leaks with the vessel at such a high temperature and the potential for exacerbating the consequences of a vessel rupture (in the extremely unlikely event that it should occur) by having the core critical. The explicit prohibition of nuclear heat in these cases was recommended to the Executive Director for Operations by the Committee to Review Generic Requirements in a memorandum dated June 7, 1990.

The requirements on thermal annealing contained in the current Appendix G (Section V.D) would be replaced by a reference to the proposed Thermal Annealing rule, 10 CFR 50.66.

Changing the reference to Appendix G of the ASME Code from Section III to Section XI means that the requirements for operating plants will no longer come from the construction code (Section III of the ASME Code) but instead will come from Section XI, the in-service inspection code. Appendix G to Section XI and Appendix G to Section III are identical, so this amendment would not result in a change in technical requirements.

Section IV.B of Appendix G requires that:

"Reactor vessels for which the predicted value of upper-shelf energy at end of life is below 50 ft-lb or for which the predicted value of adjusted reference temperature at end of life exceeds 200°F (93°C) must be designed to permit a thermal annealing treatment..."

This proposed rule would delete that requirement. This deletion conforms with Commission direction from 1985 and public comments to delete this section. An additional consideration to delete this requirement is that there should be no requirement to ensure the feasibility of a (future) voluntary activity.

During the Commission review of the revision of Appendix G published final on May 27, 1983 (48 FR 24009), the requirement to "design to permit annealing" was criticized because licensee response to the requirement was perfunctory and staff review of the responses was cursory, as detailed in SECY-83-254 (June 27, 1983). Further, there were no criteria to assess whether a design would permit annealing. An additional problem cited with the requirement was that it was misinterpreted to mean that plant operation with an RT_{NDT} greater than 200°F or a Charpy upper-shelf energy below 50 ft-lb was unsafe. The Commission indicated that it would seek public comments on the proposed deletion of the requirement, and this was done concurrently with the publication of the proposed PTS rule on February 4, 1984 (49 FR 4498). All sixteen of the commenters on this item recommended deletion of the paragraph, although eight of them urged that the deletion should not in any way imply that annealing is no longer an option to increase safety margins. In the notice of final rulemaking for the PTS rule published on July 23, 1985 (50 FR 29944), the "Supplementary Information" noted that the Commission planned a separate rulemaking action to delete Section IV.B. That planned

deletion was delayed so that it could be combined with other amendments to Appendix G.

Appendix H of 10 CFR Part 50

Changes in the fracture toughness properties of the RPV beltline materials due to irradiation embrittlement are monitored using a surveillance program, as required in Appendix H of 10 CFR Part 50, "Reactor Vessel Material Surveillance Program Requirements". Appendix H references American Society for Testing and Materials (ASTM) standard E 185 ("Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels") for many of the detailed requirements of surveillance programs, and permits the use of integrated surveillance programs, wherein surveillance program capsules for one reactor are irradiated in another reactor. This proposed amendment would make the following changes:

1. End the provision for "reducing the amount of testing" for integrated surveillance programs,
2. Restructure the section on requirements for integrated surveillance programs (Section II.C), and
3. Clarify the version of ASTM Standard E 185 that applies to the surveillance program.

Integrated surveillance programs are permitted under Section II.C of Appendix H of 10 CFR Part 50. One provision of this section is that "the amount of testing may be reduced if the initial results agree with predictions." It is proposed to discontinue this provision as of the effective date of the Appendix, although previous authorizations granted by the Director, Office of Nuclear Reactor Regulation, would continue in effect.

A second proposed change to Appendix H restructures Section II.C to clarify the requirements for integrated surveillance programs.

The other principal change to Appendix H clarifies the version of ASTM Standard E 185 that applies to the various portions of the surveillance programs. Appendix H recognizes the need to separate surveillance programs into two essential parts, specifically the design of the program, and subsequent testing and reporting of results from the surveillance capsules. Since the design of the surveillance program cannot be changed once the program is in place, the requirements for design of the surveillance program are static for each plant. However, the testing and reporting requirements are updated along with technical improvements made to ASTM standard E 185. The clarification proposed in this revision indicates that the design of the program and the withdrawal schedule must meet the requirements of ASTM E 185-73, or the edition of ASTM E 185 that is current on the issue date of the ASME Code to which the reactor vessel was purchased, whichever is latest. Licensees could choose to comply with later editions of ASTM E 185, up through the 1982 edition. Further, specimen test procedures and reporting requirements must meet the requirements of ASTM E 185-82 "to the extent practicable for the configuration of the specimens in the capsule."

The NRC staff intended that this proposed amendment to Appendix H would incorporate by reference a version of ASTM standard E 185 updated from the currently available 1982 version. However, that standardization process has not been completed, and it was decided to proceed with this proposed amendment. A subsequent amendment to Appendix H will be considered after the NRC staff has reviewed the updated ASTM standard.

Request for Public Comments

On June 13, 1994 (SECY-94-163) the staff requested Commission approval to publish for public comment these proposed revisions and provided a

discussion of options for public participation related to thermal annealing. The Commission approved issuance of the proposed revisions but directed that the staff to (1) include with the proposed rule package a discussion of options the staff has considered for structuring of the regulatory process for the proposed thermal annealing rule (10 CFR 50.66), which is included in the following section, and (2) request comments on the following issues related to the proposed regulation on thermal annealing:

1. The technical adequacy of the staff's guidance;
2. The sufficiency of the guidance and criteria to support a certification that if satisfied, a plant with an annealed vessel can safely resume operation;
3. Whether health and safety concerns are best served by approval of the thermal annealing plan or of readiness for restart;
4. The preferred regulatory process (including opportunities for public participation) and the commenter's basis for recommending a particular process; and
5. Whether there are health and safety issues concerning thermal annealing that cannot be addressed generically and would warrant plant-specific consideration.

Options the Staff has considered for Structuring of the Regulatory Process Related to Public Participation in Thermal Annealing

A significant issue with respect to thermal annealing, identified in SECY-92-283 (August 14, 1992), is the nature and timing of public participation related to the NRC's review and approval of a licensee's proposal for thermal annealing. The proposed rule does not address public participation per se, but instead provides the performance requirements that a

licensee would have to meet to gain NRC approval of a thermal annealing application and to permit subsequent operation. Under the proposed rule, there are three circumstances that arguably require an opportunity for hearing pursuant to Section 189 of the Atomic Energy Act of 1954 (AEA) in connection with NRC review and approval of thermal annealing. First, a licensee seeking to anneal its reactor vessel must obtain NRC approval of the content of the thermal annealing plan prior to implementing the plan (see Section 50.66(a) of the proposed rule). Second, and apart from the NRC approval required under Section 50.66(a), the thermal annealing process as described in the licensee's plan may necessitate license amendments (including technical specification changes). License amendments may be required if the licensee's final safety analysis report (FSAR) needs to be revised to reflect the thermal annealing process, and the licensee is unable to conclude that such FSAR changes do not constitute "unreviewed safety questions" under 10 CFR 50.59. Implementation of the thermal annealing plan may also violate existing technical specifications, necessitating requests for changes to technical specifications. Any license amendment and technical specification change must be approved by the NRC before the licensee may implement the thermal annealing plan. Finally, after the licensee implements the annealing plan, if he determines that he cannot meet the criteria specified in Section 50.66(c)(1) of the proposed rule, then NRC approval is needed in order for the licensee to resume operation (see Section 50.66(c)(2) of the proposed rule).

It is clear that any license amendments and technical specification changes necessitated by the thermal annealing plan would require an opportunity for hearing, in accordance with Section 189 of the AEA. However, the scope of such a hearing would normally be limited to consideration of whether the proposed license amendment and technical specification changes are in accordance with the Commission's rules, and therefore provide reasonable

assurance of adequate protection to the public health and safety. Issues related to the more general matter of the acceptability of the thermal annealing plan proposed by the licensee would not fall within the scope of any hearing for license amendment or technical specification change, except as they fall directly in the scope of the requested amendment or technical specification change.

However, there is some question whether the AEA <u>requires</u> an opportunity for hearing in connection with the NRC approval of the thermal annealing plan or the NRC decision approving resumption of operation under the proposed rule. There are four primary alternatives with respect to providing an opportunity for hearing in connection with thermal annealing. These alternatives are discussed in greater detail below:

Alternative 1. No Opportunity for Hearing Under Rule as Proposed

Under this alternative, the contention is that Section 189 of the AEA does not afford an interested member of the public a right to request a hearing in connection with NRC approvals of thermal annealing plans and resumption of operation under Section 50.66(c)(2). This alternative is consistent with other provisions in 10 CFR Part 50 where approval by the Director of NRR is required and hearings are not routinely offered.

Notwithstanding the lack of a requirement for a public hearing, the staff anticipates that, with respect to the initial or the first several applications for thermal annealing, several informal hearings or public meetings would be held by the staff to permit discussion of both the thermal annealing plan proposed by the licensee and the technical issues related to annealing. These hearings or meetings would ensure that all of the pertinent

technical issues have been addressed by the licensee in its thermal annealing plan and by the staff in its review of the plan. These hearings or meetings would be noticed in the Federal Register.

Alternative 2. Discretionary Opportunity for Hearing Under Rule as Proposed

Under this alternative, the contention is that Section 189 of the AEA does not afford an interested member of the public a right to request a hearing. However, as a matter of discretion, the Commission would determine on a case-by-case basis whether an opportunity for hearing will be provided in connection with the Director of NRR's determination on a thermal annealing application under Section 50.66(b) of the proposed rule. In the hearing, the Commission would consider issues related to the adequacy of the thermal annealing plan, as well as the vessel's ability to perform its safety function after being annealed.

A case-by-case determination would also be made by the NRC with respect to providing an opportunity for hearing on the Director of NRR's determination on the licensee's justification for subsequent operation under the proposed Section 50.66(c)(2).

In both cases, the Commission would publish a notice in the Federal Register announcing the NRC's approval of the licensee's thermal annealing plan or approval of resumed operation under Section 50.66(c)(2). Neither implementation of the thermal annealing plan nor resumption of operation, once approved by the NRC, would be contingent upon completion of any hearing; i.e., the Commission does not believe that it is required to make a Section 189 "no significant hazards determination" ("Sholly finding") when it provides a discretionary hearing.

Alternative 3. Required Opportunity for Hearing Under Rule as Proposed

Under this alternative, the contention is that a hearing is required by Section 189 of the AEA for both NRC's approval of the thermal annealing plan and any NRC approval of resumed operation following annealing. The adequacy of the thermal annealing plan, as well as the vessel's ability to perform its safety function after being annealed, could be raised in the hearing associated with approval of the thermal annealing plan. Licensee implementation of the thermal annealing plan could not commence until any hearing is concluded unless the NRC makes a "no significant hazards determination" with respect to the thermal annealing.

Alternative 4. Modify Proposed Rule to Require Suspension of License Prior to Thermal Annealing

Under this alternative, the proposed rule's regulatory approach for thermal annealing would be modified to include a suspension of the operating license during thermal annealing. The suspension would be automatic under the rule, without the need for a suspension order, although a letter confirming the licensee's status under the annealing rule would be prepared. The rule itself, as is currently drafted, would specify the conditions for lifting of the suspension (Section 50.66(b)). The licensee would anneal its reactor vessel without prior NRC approval of its program for conducting the annealing. Upon completion, the suspension would be lifted only if the licensee demonstrated that the thermal annealing has addressed the reactor enbrittlement such that it is acceptable to operate the plant. There would be no opportunity for hearing associated with the lifting of the suspension, and

since there would be no prior NRC approval of the annealing program, a hearing opportunity under Section 189 would not be implicated by any such approval.

Submission of Comments in Electronic Format

Commenters are encouraged to submit, in addition to the original paper copy, a copy of the letter in electronic format on a DOS-formatted (IBM compatible) 5.25 or 3.5 inch computer diskette. Text files should be provided in WordPerfect format or unformatted ASCII code. [illegible] format and version should be identified on the diskette's external label.

Finding of No Significant Environmental Impact:

The Commission has determined under the National Environmental Policy Act of 1969, as amended, and the Commission's regulations in Subpart A of 10 CFR Part 51, that this rule, if adopted, would not be a major Federal action significantly affecting the quality of human environment and, therefore, an environmental impact statement is not required.

As discussed below, the individual actions covered in this proposed rulemaking would either serve to enhance safety of the reactor pressure vessel, thereby decreasing the environmental impact of plant operation, or have no impact on the environment. Therefore, in all cases these individual actions will not have an adverse impact on the environment.

PTS Rule (10 CFR 50.61)

The inclusion of thermal annealing as an option for mitigating the effects of neutron irradiation would serve to decrease the environmental impact of plant operation by enhancing the safety of the reactor pressure vessel.

The incorporation of the Regulatory Guide 1.99, Revision 2, method for determining RT_{NDT} into the PTS rule would have no impact on the environment because this change will result in values of RT_{PTS} which are consistent with those currently used in plant operation.

The restructuring of the PTS rule is the type of action described in categorical exclusion 10 CFR 51.22(c)(2). Therefore, an environmental assessment is not necessary for this change.

Thermal Annealing Rule (10 CFR 50.66)

The proposed thermal annealing rule (10 CFR 50.66) would permit and provide requirements for the thermal annealing of a reactor vessel to restore fracture properties of the reactor vessel material which have been degraded by neutron irradiation. This rule does not affect all plants but provides an alternative for assuring compliance with the requirements in 10 CFR 50.61 and Appendix G of 10 CFR Part 50, and would only apply when a licensee elects to use it.

The application of thermal annealing to a reactor vessel would improve the condition of the reactor vessel material. In addition, this rule would establish requirements to avoid damaging the reactor system and to protect against accidents during the annealing operation, with attendant environmental consequences.

This rule is one of several regulatory requirements that will function to ensure reactor vessel integrity. In that sense, this rule would have a positive impact on the environment by reducing the potential for vessel failure. For these reasons, the Commission has determined that there would be no significant impact and, therefore, an environmental statement is not required.

Appendix G of 10 CFR Part 50

Concerning the amendments proposed to Appendix G of 10 CFR Part 50, the prohibition of core criticality before completion of the required pressure and leak tests will serve to reduce the potential for vessel failure, and thereby decrease the environmental impact of plant operation.

The restructuring of Sections IV and V of Appendix G is clarifying or corrective in nature, and hence is the type of action described in categorical exclusion 10 CFR 51.22(c)(2). Therefore, an environmental assessment is not necessary for this change.

The changing of the reference from Appendix G of Section III of the ASME Code to Appendix G of Section XI of the ASME Code has no impact on the environment since the requirements in the Appendices are identical. Therefore, there is no adverse impact on the environment from this change.

The referencing of the thermal annealing rule results in no adverse impact on the environment since Appendix G currently permits the use of thermal annealing to reduce fracture toughness loss of the RPV materials due to irradiation embrittlement.

The deletion of the "design to permit annealing" requirement has no adverse impact on the environment. This assessment is based on the fact that annealing is a voluntary activity.

Appendix H of 10 CFR Part 50

Concerning the amendments proposed to Appendix H of 10 CFR Part 50, the requirement that all irradiation surveillance tests be made (i.e., no reduction in testing is permitted) would have a positive impact on the environment in helping to assure the integrity of the reactor pressure vessel.

The restructuring of Section II.C is the type of action described in categorical exclusion 10 CFR 51.22(c)(2). Therefore, an environmental assessment is not necessary for this change.

The clarification of the applicable version of ASTM Standard E 185 will result in no adverse impact to the environment since there will be no change to current surveillance programs. Changes to future surveillance programs will make the programs more effective in assessing irradiation embrittlement effects to the RPV materials, thereby helping to assure the integrity of the reactor pressure vessel.

Paperwork Reduction Act Statement

This proposed rule amends information collection requirements that are subject to the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 et seq.). This rule has been submitted to the Office of Management and Budget for review and approval of the paperwork requirements.

The public reporting burden for this collection of information is estimated to average 6,000 hours per respondent, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the collection of information. Send comments regarding the burden estimate or any other aspect of this collection of information, including suggestions for reducing the burden to the Information and Records Management Branch (T-6 F33), U.S. Nuclear Regulatory Commission, Washington, D.C. 20555; and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-3019, (3150-0011), Office of Management and Budget, Washington, D.C. 20503.

Regulatory Analysis

The NRC staff has prepared a regulatory analysis for the proposed amendments to 10 CFR 50.61, 10 CFR Part 50, Appendix G and 10 CFR Part 50 Appendix H, which describes the factors and alternatives considered by the Commission in deciding to propose these amendments. A copy of the regulatory analysis is available for inspection and copying for a fee at the NRC Public Document Room, 2120 L Street NW. (Lower Level), Washington, DC 20555. Single copies of the analysis may be obtained from Alfred Taboada, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone, (301) 415-6014.

Single copies of the regulatory analysis prepared for 10 CFR 50.66 may be obtained from Alfred Taboada, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone, (301) 415-6014.

Regulatory Flexibility Act Certification

As required by the Regulatory Flexibility Act, 5 U.S.C. 605(b), the Commission certifies that, if adopted, the proposed amendments would not have a significant economic impact on a substantial number of small entities. The rules which would be affected by the proposed amendments: (1) preclude brittle fracture of embrittled vessels during PTS events, (2) provide the general fracture toughness requirements for RPVs, including ductile fracture toughness requirements and pressure-temperature limits, (3) provide the requirements for surveillance programs to monitor irradiation embrittlement of RPV beltline materials, and (4) provide for a method for restoring the fracture toughness of RPV beltline materials used in nuclear facilities

licensed under the provision of 10 CFR 50.21(b) and 10 CFR 50.22. The companies that own these facilities do not fall within the scope of the definition of "small entities" as set forth in the Regulatory Flexibility Act or the Small Business Size Standards in regulations issued by the Small Business Administration at 13 CFR Part 121.

Backfit Analysis

PTS Rule (10 CFR 50.61)

The proposed revision to Section 50.61 would require licensees to calculate RT_{PTS} using the same methodology specified in Regulatory Guide 1.99, Revision 2 for determining RT_{NDT}. This proposal is logically a requisite part of the 1991 revisions to 50.61, which addressed a unified method for calculating radiation enbrittlement of the reactor beltline materials. However, the Commission inadvertently failed to make the conforming change to 50.61. Therefore, the Commission believes that the backfit statement for the 1991 amendments, which determined that the backfits were necessary to ensure that the facility provides adequate protection to the public health and safety, are applicable to this conforming change to Section 50.61.

The restructuring of the PTS rule does not impose any backfits as defined in 10 CFR 50.109(a)(1), since there is no change in requirements due to this restructuring.

The inclusion of thermal annealing does not impose any backfits as defined in 10 CFR 50.109(a)(1), for the reasons set forth below in "Thermal Annealing Rule (10 CFR 50.66)."

Thermal Annealing Rule (10 CFR 50.66)

The proposed thermal annealing rule would establish new requirements with respect to applications for thermal annealing. However, the Commission has determined that the proposed rule would not impose a "backfit" as defined in 10 CFR 50.109(a)(1). The proposed thermal annealing rule would not require any licensee to perform thermal annealing. Under existing requirements, all licensees are required to evaluate whether they exceed the PTS screening limits in 10 CFR 50.61 and the Charpy upper shelf screening limits in 10 CFR Part 50, Appendix G. However, these rules provide an alternative means to meet these screening limits, viz., performing thermal annealing. No licensee currently has pending before the NRC an application for thermal annealing, nor has any current licensee been granted permission to conduct thermal annealing. In addition, the proposed rule does not reflect any new or different Staff position which conflicts with a prior Staff position or Commission rule. Thus, the proposed rule would have a purely prospective effect on future applications for thermal annealing. The Commission has stated in other rulemakings establishing prospective requirements, e.g., 10 CFR Part 52 and the License Renewal Rule, 10 CFR Part 54, that the Backfit Rule was not intended to protect the future applicant from current changes in Commission requirements when there are no prior NRC positions upon which the "substantial increase in overall protection" can be measured. Accordingly, the Commission concludes that the proposed rule does not impose backfits and a backfit analysis need not be prepared for the proposed thermal annealing rule.

10 CFR Part 50 Appendix G

The restructuring of Sections IV and V of this appendix, referencing of the thermal annealing rule, changing the reference from Appendix G of Section III of the ASME Code to Appendix G of Section XI of the ASME Code, and deleting the "design to permit annealing" requirement do not impose any backfits as defined in 10 CFR 50.109(a)(1), because they are either prospective in nature or of a clarifying nature.

The explicit prohibition on core criticality before the completion of pressure and leak tests can be construed as a backfit, although NRC staff intent was never to permit such a procedure (letter from J. M. Taylor, NRC, to N. S. Reynolds and D. F. Stenger, NUBARG, dated February 2, 1990). The Commission has concluded that any backfit requirements in this amendment are necessary to bring the facilities into compliance with licenses, or the rules and orders of the Commission, or into conformance with written commitments by the licensees. Therefore, a backfit analysis is not required pursuant to 10 CFR 50.109(a)(4)(i). This amendment underscores the prior intent of the Commission to prohibit the use of nuclear heat before the completion of leak and pressure tests that is implicit in 10 CFR 50.55a and Section XI of the ASME Code. The Commission's intent in this regard is demonstrated by the fact that only a very small minority of licensees actually used nuclear heat to conduct pressure and leak tests required by the ASME Code.

10 CFR Part 50 Appendix H

The amendments to Appendix H of 10 CFR Part 50 are either prospective in nature or of a clarifying nature, and hence do not involve any provisions which would impose backfits as defined in 10 CFR 50.109(a)(1).

Criminal Penalties

For purposes of Section 223 of the Atomic Energy Act (AEA), the Commission proposes to issue the proposed rule under one or more of Sections 161b, 161i or 161o of the AEA. Willful violations of the rule would be subject to criminal enforcement.

List of Subjects

10 CFR Part 50

Antitrust, Classified information, Criminal penalties, Fire protection, Incorporation by reference, Intergovernmental relations, Nuclear power plants and reactors, Radiation protection, Reactor siting criteria, Reporting and recordkeeping requirements.

For the reasons set out in the preamble and under the authority of the Atomic Energy Act of 1954, as amended; the Energy Reorganization Act of 1974, as amended; and 5 U.S.C. 553; the NRC is proposing to adopt the following amendments to 10 CFR Part 50.

PART 50 -- DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

1. The authority citation for Part 50 is revised to read as follows:

AUTHORITY: Secs. 102, 103, 104, 105, 161, 182, 183, 186, 189, 68 Stat. 936, 937, 938, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 1244, as amended (42 U.S.C. 2132, 2133, 2134, 2135, 2201, 2232, 2233, 2236, 2239, 2282); secs. 201, as amended, 202, 206, 88 Stat. 1242, as amended 1244, 1246, (42 U.S.C. 5841, 5842, 5846).

Section 50.7 also issued under Pub. L. 95-601, sec. 10, 92 Stat. 2951 (42 U.S.C. 5851). Section 50.10 also issued under secs. 101, 185, 68 Stat. 955 as amended (42 U.S.C. 2131, 2235), sec. 102, Pub. L. 91-190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.13, and 50.54(dd), and 50.103 also issued under sec. 108, 68 Stat. 939, as amended (42 U.S.C. 2138). Sections 50.23, 50.35, 50.55, and 50.56 also issued under sec. 185, 68 Stat. 955 (42 U.S.C. 2235). Sections 50.33a, 50.55a and Appendix Q also issued under sec. 102, Pub. L. 91-190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.34 and 50.54 also issued under sec. 204, 88 Stat. 1245 (42 U.S.C. 5844). Sections 50.58, 50.91, and 50.92 also issued under Pub. L. 97-415, 96 Stat. 2073 (42 U.S.C. 2239). Section 50.78 also issued under sec. 122, 68 Stat. 939 (42 U.S.C. 2152). Sections 50.80 - 50.81 also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C. 2234). Appendix F also issued under sec. 187, 68 Stat 955 (42 U.S.C. 2237).

2. Section 50.61 is revised to read as follows:

§ 50.61 Fracture toughness requirements for protection against pressurized thermal shock events.

(a) Definitions. For the purposes of this section:

(1) ASME Code means the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Division I, "Rules for the Construction of Nuclear Power Plant Components," edition and addenda and any limitations and modifications thereof as specified in § 50.55a of this part.

(2) Pressurized Thermal Shock Event means an event or transient in pressurized water reactors (PWRs) causing severe overcooling (thermal shock) concurrent with or followed by significant pressure in the reactor vessel.

(3) Reactor Vessel Beltline means the region of the reactor vessel (shell material including welds, heat affected zones and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage.

(4) RT_{NDT} means the reference temperature for a reactor vessel material, under any conditions. For the reactor vessel beltline materials, RT_{NDT} must account for the effects of neutron radiation.

(5) $RT_{NDT(U)}$ means the reference temperature for a reactor vessel material in the pre-service or unirradiated condition, evaluated according to the procedures in the ASME Code, Paragraph NB-2331.

(6) EOL Fluence means the best-estimate neutron fluence projected for a specific vessel beltline material at the clad-base-metal interface on the inside surface of the vessel at the location where the material receives the highest fluence on the expiration date of the operating license, the proposed

expiration date if a change in the term of the operating license has been requested, or the end of a renewal term if an application for a renewed license under 10 CFR Part 54 has been submitted.

(7) $\underline{RT_{PTS}}$ means the reference temperature, RT_{NDT}, evaluated for the EOL Fluence for each of the vessel beltline materials, using the procedures of paragraph (c) of this section.

(8) PTS Screening Criterion means the value of RT_{PTS} for the vessel beltline material above which the plant cannot continue to operate without justification.

(b) Requirements.

(1) For each pressurized water nuclear power reactor for which an operating license has been issued, the licensee shall have projected values of RT_{PTS}, accepted by the NRC, for each reactor vessel beltline material for the EOL fluence of the material. The assessment of RT_{PTS} must use the calculative procedures given in paragraph (c)(1) of this section, except as provided in paragraphs (c)(2) and (c)(3) of this section. The assessment must specify the bases for the projected value of RT_{PTS} for each vessel beltline material, including the assumptions regarding core loading patterns, and must specify the copper and nickel contents and the fluence value used in the calculation for each beltline material. This assessment must be updated whenever there is a significant[1] change in projected values of RT_{PTS}, or upon a request for a change in the expiration date for operation of the facility.

(2) The pressurized thermal shock (PTS) screening criterion is 270°F for plates, forgings, and axial weld materials, and 300°F for circumferential weld

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[1] Changes to RT_{PTS} values are considered significant if either the previous value or the current value, or both values, exceed the screening criterion prior to the expiration of the operating license, including any renewed term, if applicable, for the plant.

materials. For the purpose of comparison with this criterion, the value of RT_{PTS} for the reactor vessel must be evaluated according to the procedures of paragraph (c) of this section, for each weld and plate, or forging, in the reactor vessel beltline. RT_{PTS} must be determined for each vessel beltline material using the EOL fluence for that material.

(3) For each pressurized water nuclear power reactor for which the value of RT_{PTS} for any material in the beltline is projected to exceed the PTS screening criterion using the EOL fluence, the licensee shall implement those flux reduction programs that are reasonably practicable to avoid exceeding the PTS screening criterion set forth in paragraph (b)(2) of this section. The schedule for implementation of flux reduction measures may take into account the schedule for submittal and anticipated approval by the Director, Office of Nuclear Reactor Regulation, of detailed plant-specific analyses, submitted to demonstrate acceptable risk with RT_{PTS} above the screening limit due to plant modifications, new information or new analysis techniques.

(4) For each pressurized water nuclear power reactor for which the analysis required by paragraph (b)(3) of this section indicates that no reasonably practicable flux reduction program will prevent RT_{PTS} from exceeding the PTS screening criterion using the EOL fluence, the licensee shall submit a safety analysis to determine what, if any, modifications to equipment, systems, and operation are necessary to prevent potential failure of the reactor vessel as a result of postulated PTS events if continued operation beyond the screening criterion is allowed. In the analysis, the licensee may determine the properties of the reactor vessel materials based on available information, research results, and plant surveillance data, and may use probabilistic fracture mechanics techniques. This analysis must be submitted at least three years before RT_{PTS} is projected to exceed the PTS screening criterion.

(5) After consideration of the licensee's analyses, including effects of proposed corrective actions, if any, submitted in accordance with paragraphs (b)(3) and (b)(4) of this section, the Director, Office of Nuclear Reactor Regulation, may, on a case-by-case basis, approve operation of the facility with RT_{PTS} in excess of the PTS screening criterion. The Director, Office of Nuclear Reactor Regulation, will consider factors significantly affecting the potential for failure of the reactor vessel in reaching a decision.

(6) If the Director, Office of Nuclear Reactor Regulation, concludes, pursuant to paragraph (b)(5) of this section, that operation of the facility with RT_{PTS} in excess of the PTS screening criterion cannot be approved on the basis of the licensee's analyses submitted in accordance with paragraphs (b)(3) and (b)(4) of this section, the licensee shall request and receive approval by the Director, Office of Nuclear Reactor Regulation, prior to any operation beyond the criterion. The request must be based upon modifications to equipment, systems, and operation of the facility in addition to those previously proposed in the submitted analyses that would reduce the potential for failure of the reactor vessel due to PTS events, or upon further analyses based upon new information or improved methodology.

(7) If the limiting RT_{PTS} value of the plant is projected to exceed the screening criteria in paragraph (b)(2), or the criteria in paragraphs (b)(3) through (b)(6) of this section cannot be satisfied, the reactor vessel beltline may be given a thermal annealing treatment to recover the fracture toughness of the material, subject to the requirements of § 50.66. The reactor vessel may continue to be operated only for that service period within which the predicted fracture toughness of the vessel beltline materials satisfy the requirements of paragraphs (b)(2) through (b)(6) of this section, with RT_{PTS} accounting for the effects of annealing and subsequent irradiation.

(c) Calculation of RT_{PTS}. RT_{PTS} must be evaluated using the same procedures used to calculate RT_{NDT}, as indicated in paragraph (c)(1) of this section, and as provided in paragraphs (c)(2) and (c)(3). RT_{PTS} must be calculated for each vessel beltline material using a fluence value, f, which is the EOL fluence for the material.

(1) Equation 1 must be used to calculate values of RT_{NDT} for each weld and plate, or forging, in the reactor vessel beltline.

Equation 1: $$RT_{NDT} = RT_{NDT(U)} + M + \Delta RT_{NDT}$$

(i) $\underline{RT_{NDT(U)}}$ is the reference temperature, RT_{NDT}, of the material in the pre-service or unirradiated condition, evaluated according to the procedures in the ASME Code, Paragraph NB-2331.

(A) If a measured value of $RT_{NDT(U)}$ is not available, a generic mean value for the class[2] of material may be used if there are sufficient test results to establish a mean and a standard deviation for the class.

(B) For weld metals, the following generic mean values must be used, unless justification for different values is provided: 0°F for welds made with Linde 80 flux, and -56°F for welds made with Linde 0091, 1092 and 124 and ARCOS B-5 weld fluxes.

(ii) $\underline{M}$ means the margin to be added to account for uncertainties in the values of $RT_{NDT(U)}$, copper and nickel contents, fluence and the calculational procedures. M is evaluated from Equation 2.

Equation 2: $$M = 2\sqrt{\sigma_U^2 + \sigma_\Delta^2}$$

[2] The class of material for estimating $RT_{NDT(U)}$ is generally determined for welds by the type of welding flux (Linde 80, or other), and for base metal by the material specification.

(A) In Equation 2, σ_U is the standard deviation for $RT_{NDT(U)}$. If a measured value of $RT_{NDT(U)}$ is used, then σ_U is determined from the precision of the test method. If a measured value of $RT_{NDT(U)}$ is not available and a generic mean value for that class of materials is used, then σ_U is the standard deviation obtained from the set of data used to establish the mean. If a generic mean value given in paragraph (c)(1)(i)(B) for welds is used, then σ_U is 17°F.

(B) In Equation 2, σ_Δ is the standard deviation for ΔRT_{NDT}. The value of σ_Δ to be used is 28°F for welds and 17°F for base metal; the value of σ_Δ shall not exceed one-half of ΔRT_{NDT}.

(iii) ΔRT_{NDT} is the mean value of the transition temperature shift, or change in RT_{NDT}, due to irradiation, and must be calculated using Equation 3.

Equation 3: $$\Delta RT_{NDT} = (CF) f^{(0.28-0.10 \log f)}$$

(A) <u>CF</u> (°F) is the chemistry factor, which is a function of copper and nickel content. CF is given in Table 1 for welds and in Table 2 for base metal (plates and forgings). Linear interpolation is permitted. In Tables 1 and 2, "Wt-% copper" and "Wt-% nickel" are the best-estimate values for the material, which will normally be the mean of the measured values for a plate or forging. For a weld, the best estimate values will normally be the mean of the measured values for a weld deposit made using the same weld wire heat number as the critical vessel weld. If these values are not available, the upper limiting values given in the material specifications to which the vessel material was fabricated may be used. If not available, conservative estimates

(mean plus one standard deviation) based on generic data[3] may be used if justification is provided. If none of these alternatives are available, 0.35% copper and 1.0% nickel must be assumed.

(B) _f_ is the best estimate neutron fluence, in units of 10^{19} n/cm^2 (E greater than 1 MeV), at the clad-base-metal interface on the inside surface of the vessel at the location where the material in question receives the highest fluence for the period of service in question. As specified in paragraph (c), the EOL fluence for the vessel beltline material is used in calculating RT_{PTS}.

(2) To verify that RT_{NDT} for each vessel beltline material is a bounding value for the specific reactor vessel, licensees shall consider plant-specific information that could affect the level of embrittlement. This information includes but is not limited to the reactor vessel operating temperature and surveillance program results.

(i) Results from the plant-specific surveillance program must be integrated into the RT_{NDT} estimate if the plant-specific surveillance data has been deemed credible as judged by the following criteria:

(A) The materials in the surveillance capsules must be those which are the controlling materials with regard to radiation embrittlement.

(B) Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions must be small enough to permit the determination of the 30-foot-pound temperature unambiguously.

(C) Where there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values must be less than 28°F for welds and

[3] Data from reactor vessels fabricated to the same material specification in the same shop as the vessel in question and in the same time period is an example of "generic data."

17°F for base metal. Even if the range in the capsule fluences is large (two or more orders of magnitude), the scatter may not exceed twice those values.

(D) The irradiation temperature of the Charpy specimens in the capsule must equal the vessel wall temperature at the cladding/base metal interface within ±25°F.

(E) The surveillance data for the correlation monitor material in the capsule must fall within the scatter band of the data base for the material.

(ii) Surveillance data deemed credible according to the criteria of paragraph (c)(2)(i) must be used to determine a material-specific value of CF for use in Equation 3. A material-specific value of CF is determined from Equation 4.

Equation 4:

$$CF = \frac{\sum_{i=1}^{n}\left[A_i \times f_i^{(0.28 - 0.10 \log f_i)}\right]}{\sum_{i=1}^{n}\left[f_i^{(0.56 - 0.20 \log f_i)}\right]}$$

In Equation 4, "n" is the number of surveillance data points, "A_i" is the measured value of ΔRT_{NDT} and "f_i" is the fluence for each surveillance data point.

(iii) For cases in which the results from a credible plant-specific surveillance program are used, the value of σ_Δ to be used is 14°F for welds and 8.5°F for base metal; the value of σ_Δ may not exceed one-half of ΔRT_{NDT}.

(iv) The use of results from the plant-specific surveillance program may result in an RT_{NDT} that is higher or lower than those determined in paragraph (c)(1).

(3) Any information that is believed to improve the accuracy of the RT_{PTS} value significantly must be reported to the Director, Office of Nuclear

Reactor Regulation. Any value of RT_{PTS} that has been modified using the procedures of paragraph (c)(2) is subject to the approval of the Director, Office of Nuclear Reactor Regulation when used as provided in this section.

TABLE 1
CHEMISTRY FACTOR FOR WELD METALS, °F

Copper, Wt-%	Nickel, Wt-% 0	0.20	0.40	0.60	0.80	1.00	1.20
0	20	20	20	20	20	20	20
0.01	20	20	20	20	20	20	20
0.02	21	26	27	27	27	27	27
0.03	22	35	41	41	41	41	41
0.04	24	43	54	54	54	54	54
0.05	26	49	67	68	68	68	68
0.06	29	52	77	82	82	82	82
0.07	32	55	85	95	95	95	95
0.08	36	58	90	106	108	108	108
0.09	40	61	94	115	122	122	122
0.10	44	65	97	122	133	135	135
0.11	49	68	101	130	144	148	148
0.12	52	72	103	135	153	161	161
0.13	58	76	106	139	162	172	176
0.14	61	79	109	142	168	182	188
0.15	66	84	112	146	175	191	200
0.16	70	88	115	149	178	199	211
0.17	75	92	119	151	184	207	221
0.18	79	95	122	154	187	214	230
0.19	83	100	126	157	191	220	238
0.20	88	104	129	160	194	223	245
0.21	92	108	133	164	197	229	252
0.22	97	112	137	167	200	232	257
0.23	101	117	140	169	203	236	263
0.24	105	121	144	173	206	239	268
0.25	110	126	148	176	209	243	272
0.26	113	130	151	180	212	246	276
0.27	119	134	155	184	216	249	280
0.28	122	138	160	187	218	251	284
0.29	128	142	164	191	222	254	287
0.30	131	146	167	194	225	257	290
0.31	136	151	172	198	228	260	293
0.32	140	155	175	202	231	263	296
0.33	144	160	180	205	234	266	299
0.34	149	164	184	209	238	269	302
0.35	153	168	187	212	241	272	305
0.36	158	172	191	216	245	275	308
0.37	162	177	196	220	248	278	311
0.38	166	182	200	223	250	281	314
0.39	171	185	203	227	254	285	317
0.40	175	189	207	231	257	288	320

TABLE 2
CHEMISTRY FACTOR FOR BASE METALS, °F

Copper, Wt-%	Nickel, Wt-% 0	0.20	0.40	0.60	0.80	1.00	1.20
0	20	20	20	20	20	20	20
0.01	20	20	20	20	20	20	20
0.02	20	20	20	20	20	20	20
0.03	20	20	20	20	20	20	20
0.04	22	26	26	26	26	26	26
0.05	25	31	31	31	31	31	31
0.06	28	37	37	37	37	37	37
0.07	31	43	44	44	44	44	44
0.08	34	48	51	51	51	51	51
0.09	37	53	58	58	58	58	58
0.10	41	58	65	65	67	67	67
0.11	45	62	72	74	77	77	77
0.12	49	67	79	83	86	86	86
0.13	53	71	85	91	96	96	96
0.14	57	75	91	100	105	106	106
0.15	61	80	99	110	115	117	117
0.16	65	84	104	118	123	125	125
0.17	69	88	110	127	132	135	135
0.18	73	92	115	134	141	144	144
0.19	78	97	120	142	150	154	154
0.20	82	102	125	149	159	164	165
0.21	86	107	129	155	167	172	174
0.22	91	112	134	161	176	181	184
0.23	95	117	138	167	184	190	194
0.24	100	121	143	172	191	199	204
0.25	104	126	148	176	199	[illegible]	214
0.26	109	130	151	180	205	216	221
0.27	114	134	155	184	211	225	230
0.28	119	138	160	187	216	233	239
0.29	124	142	164	191	221	241	248
0.30	129	146	167	194	225	249	257
0.31	134	151	172	198	228	255	266
0.32	139	155	175	202	231	260	274
0.33	144	160	180	205	234	264	282
0.34	149	164	184	209	238	268	290
0.35	153	168	187	212	241	272	298
0.36	158	173	191	216	245	275	303
0.37	162	177	196	220	248	278	308
0.38	166	182	200	223	250	281	313
0.39	171	185	203	227	254	285	317
0.40	175	189	207	231	257	288	320

3. A new § 50.66 is added to read as follows:

§ 50.66 Requirements for thermal annealing of the reactor pressure vessel.

(a) For those light water nuclear power reactors where neutron radiation has reduced the fracture toughness of the reactor vessel materials, a thermal annealing treatment may be applied to the reactor vessel to restore the fracture toughness to acceptable levels. The use of a thermal annealing treatment is subject to the approval of the Director, Office of Nuclear Reactor Regulation, and to the requirements in this section. The application for the Director's approval must be submitted in accordance with § 50.4, and at least three years prior to the proposed date of the annealing operation.

(b) Thermal Annealing Application. The content of the application for approval by the Director, Office of Nuclear Reactor Regulation, for thermal annealing of the reactor vessel must include: a thermal annealing operating plan that includes an evaluation of the effects of mechanical and thermal stresses and temperatures, an inspection and test program to requalify the annealed reactor vessel, and a program for demonstrating that the recovery of fracture toughness and the re-embrittlement rate are adequate to permit subsequent safe operation of the reactor vessel for the period specified in the application.

(1) Thermal Annealing Operating Plan.

(i) The thermal annealing operating plan must include:

(A) A detailed description of the pressure vessel and all structures and components that will be affected by the thermal annealing operation;

(B) The methods, instrumentation and procedures proposed for performing the thermal annealing;

(C) A description of the heat source to be used; and

(D) The proposed thermal annealing operating parameters, including temperatures, times, and heatup and cooldown schedules.

(ii) The annealing time and temperature parameters selected must be based on projecting sufficient recovery of fracture toughness, using the procedures of paragraph (d) of this section, to satisfy the requirements of § 50.60 and § 50.61 for the proposed period of operation addressed in the application. In addition, the operating plan must describe any special precautions necessary to minimize occupational exposure, in accordance with the As Low As Reasonably Achievable (ALARA) principle and the provisions of § 20.1206.

(iii) An evaluation of the effects of mechanical and thermal stresses and temperatures on the vessel, attached piping and appurtenances, and adjacent equipment and components must demonstrate that operability of the reactor will not be detrimentally affected. A detailed thermal and structural analysis must be performed to establish the time and temperature profile of the annealing operation. These analyses must include heatup and cooldown rates, and must demonstrate that localized temperatures, thermal stress gradients, and subsequent residual stresses will not result in unacceptable dimensional changes or distortions in the vessel, attached piping and appurtenances, and that the thermal annealing cycle will not result in unacceptable degradation of the fatigue life of these components. The effects of localized high temperatures must be evaluated for degradation of the concrete adjacent to the vessel and changes in thermal and mechanical properties of the reactor vessel insulation. If the design temperature limitations for the adjacent concrete structure are to be exceeded during the annealing

operation, an acceptable maximum temperature for the concrete must be established for the annealing operation using appropriate test data.

(iv) The time and temperature profile evaluated as part of the annealing operating plan, and supported by the evaluation results of paragraph (b)(1)(iii) of this section, represents the proposed annealing conditions that may not be exceeded during the annealing operation. If these conditions are exceeded, then the licensee cannot certify that the annealing operation was performed in accordance with the approved application, as required by paragraph (c)(1) of this section, and must comply with paragraph (c)(2) of this section.

(v) The projected percent recovery of both RT_{NDT} and Charpy upper-shelf energy must be determined by the procedures described in paragraph (d) of this section, using the proposed annealing time and temperature described in the operating plan. The projected post-anneal RT_{NDT} and Charpy upper-shelf energy must be determined from the projected percent recovery.

(vi) The projected rate of reembrittlement of RT_{NDT} must be calculated using the procedures in § 50.61(c), or must be the same rate as that used for the pre-anneal operating period. The projected rate of reembrittlement for Charpy upper-shelf energy must be the same rate as that used for the pre-anneal operating period.

(2) Requalification Inspection and Test Program. The inspection and test program to requalify the annealed reactor vessel must include the detailed monitoring, inspections, and tests proposed to demonstrate that the limitations on temperatures, times and temperature profiles, and stresses evaluated for the proposed annealing conditions of paragraph (b)(1)(iv) of this section have not been exceeded, and to determine the annealing time and

temperature to be used in quantifying the fracture toughness recovery. In addition, the program must demonstrate that the annealing operation has not degraded the reactor vessel, attached piping or appurtenances, or the adjacent concrete structures to a degree that could affect the safe operation of the reactor.

(3) Fracture Toughness Recovery and Reembrittlement Rate Assurance Program. The percent recovery of RT_{NDT} and Charpy upper-shelf energy obtained by the thermal annealing treatment must be determined from the time and temperature of the actual vessel annealing. The recovery of RT_{NDT} and Charpy upper-shelf energy provide the basis for establishing the post-anneal RT_{NDT} and Charpy upper-shelf energy for each vessel material. Changes in the RT_{NDT} and Charpy upper-shelf energy with subsequent plant operation must be determined using the post-anneal values of these parameters in conjunction with the projected reembrittlement rate determined in accordance with paragraph (b)(3)(ii) of this section.

(i) The recovery of RT_{NDT} and Charpy upper-shelf energy must be established using the procedures in paragraph (d) of this section, using the time and temperature of the actual vessel annealing.

(A) If the percent recovery is determined from testing surveillance specimens or from testing materials removed from the reactor vessel, then it shall be demonstrated that the proposed annealing parameters used in the test program are equal to or bounded by those used in the vessel annealing operation.

(B) If generic computational methods are used, appropriate justification must be submitted as a part of the application.

(ii) The reembrittlement rate of both RT_{NDT} and Charpy upper-shelf energy must be estimated, and must be monitored using a surveillance program which conforms to Appendix H of this part, "Reactor Vessel Material Surveillance Program Requirements."

(c) Certification of the Annealing Effectiveness.

(1) Upon completion of the anneal and prior to re-start of the nuclear power plant, the licensee shall certify to the Director, Office of Nuclear Reactor Regulation, that the thermal annealing was performed in accordance with the approved application required by paragraph (a) of this section, and meets the provisions of paragraph (b) of this section. In this certification, the licensee shall establish the period for which the reactor vessel will satisfy the requirements of § 50.60 and § 50.61, and shall provide:

(i) The post-anneal RT_{NDT} and Charpy upper-shelf energy values of the reactor vessel materials for use in subsequent reactor operation;

(ii) The projected reembrittlement trends for both RT_{NDT} and Charpy upper-shelf energy; and

(iii) The projected values of RT_{PTS} and Charpy upper-shelf energy at the end of the proposed period of operation addressed in the application.

(2) If the licensee cannot certify that the thermal annealing was performed in accordance with the approved application and the provisions of paragraph (b) of this section, the licensee shall submit a justification for subsequent operation for approval by the Director, Office of Nuclear Reactor Regulation.

(d) Procedures for Determining the Recovery of Fracture Toughness. The procedures of this paragraph must be used to determine the percent recovery of

ΔRT_{NDT}, R_t, and percent recovery of Charpy upper-shelf energy, R_u. In all cases, R_t and R_u may not exceed 100.

(1) For those reactors with surveillance programs which have developed credible surveillance data as defined in § 50.61, percent recovery due to annealing (R_t and R_u) must be evaluated by testing surveillance specimens that have been withdrawn from the surveillance program and that have been annealed under the same time and temperature conditions as those given the beltline material.

(2) Alternatively, the percent recovery due to annealing (R_t and R_u) may be determined from the results of a verification test program employing materials removed from the beltline region of the reactor vessel[1] and that have been annealed under the same time and temperature conditions as those given the beltline material.

(3) Generic computational methods may be used to determine recovery if adequate justification is provided.

4. In 10 CFR Part 50, Appendix G is revised to read as follows:

Appendix G to Part 50 -- Fracture Toughness Requirements

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[1] For those cases where materials are removed from the beltline of the pressure vessel, the stress limits of the applicable portions of the ASME Code Section III must be satisfied, including consideration of fatigue and corrosion, regardless of the Code of record for the vessel design.

Table of Contents

I. Introduction and Scope

This appendix specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime.

The ASME Code forms the basis for the requirements of this appendix. "ASME Code" means the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. If no section is specified, the reference is to Section III, Division 1, "Rules for Construction of Nuclear Power Plant Components." "Section XI" means Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components." If no edition or addenda are specified, the ASME Code edition and addenda and any limitations and modifications thereof, which are specified in § 50.55a, are applicable.

The sections, editions and addenda of the ASME Boiler and Pressure Vessel Code specified in § 50.55a have been approved for incorporation by reference by the Director of the Federal Register. A notice of any changes

made to the material incorporated by reference will be published in the Federal Register. Copies of the ASME Boiler and Pressure Vessel Code may be purchased from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th St., New York, NY 10017 and are available for inspection at the NRC Library, 11545 Rockville Pike, Two White Flint North, Rockville, Maryland 20852-2738.

The requirements of this appendix apply to the following materials:

A. Carbon and low-alloy ferritic steel plate, forgings, castings, and pipe with specified minimum yield strengths not over 50,000 psi (345 MPa), and to those with specified minimum yield strengths greater than 50,000 psi (345 MPa) but not over 90,000 psi (621 MPa) if qualified by using methods equivalent to those described in paragraph G-2110 of Appendix G of Section XI of the latest edition and addenda of the ASME Code incorporated by reference into § 50.55a(b)(2).

B. Welds and weld heat-affected zones in the materials specified in paragraph I.A. of this appendix.

C. Materials for bolting and other types of fasteners with specified minimum yield strengths not over 130,000 psi (896 MPa).

Note: The adequacy of the fracture toughness of other ferritic materials not covered in this section must be demonstrated to the Director, Office of Nuclear Reactor Regulation, on an individual case basis.

II. Definitions

A. Ferritic material means carbon and low-alloy steels, higher alloy steels including all stainless alloys of the 4xx series, and maraging and

precipitation hardening steels with a predominantly body-centered cubic crystal structure.

B. System hydrostatic tests means all preoperational system leakage and hydrostatic pressure tests and all system leakage and hydrostatic pressure tests performed during the service life of the pressure boundary in compliance with the ASME Code, Section XI.

C. Specified minimum yield strength means the minimum yield strength (in the unirradiated condition) of a material specified in the construction code under which the component is built under § 50.55a.

D. RT_{NDT} means the reference temperature of the material, for all conditions.

(i) For the pre-service or unirradiated condition, RT_{NDT} is evaluated according to the procedures in the ASME Code, Paragraph NB-2331.

(ii) For the reactor vessel beltline materials, RT_{NDT} must account for the effects of neutron radiation.

E. ΔRT_{NDT} means the transition temperature shift, or change in RT_{NDT}, due to neutron radiation effects, which is evaluated as the difference in the 30 ft-lb (41 J) index temperatures from the average Charpy curves measured before and after irradiation.

F. Beltline or Beltline region of reactor vessel means the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage.

III. Fracture Toughness Tests

A. To demonstrate compliance with the fracture toughness requirements of Section IV of this appendix, ferritic materials must be tested in accordance with the ASME Code and, for the beltline materials, the test requirements of Appendix H of this part. For a reactor vessel that was constructed to an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition (under § 50.55a), the fracture toughness data and data analyses must be supplemented in a manner approved by the Director, Office of Nuclear Reactor Regulation, to demonstrate equivalence with the fracture toughness requirements of this appendix.

B. Test methods for supplemental fracture toughness tests described in paragraph IV.A.1.b of this appendix must be submitted to and approved by the Director, Office of Nuclear Reactor Regulation, prior to testing.

C. All fracture toughness test programs conducted in accordance with paragraphs III.A and III.B must comply with ASME Code requirements for calibration of test equipment, qualification of test personnel, and retention of records of these functions and of the test data.

IV. Fracture Toughness Requirements

A. The pressure-retaining components of the reactor coolant pressure boundary that are made of ferritic materials must meet the requirements of the ASME Code, supplemented by the additional requirements set forth below, for fracture toughness during system hydrostatic tests and any condition of normal operation, including anticipated operational occurrences. Reactor vessels may

continue to be operated only for that service period within which the requirements of this section are satisfied. For the reactor vessel beltline materials, including welds, plates and forgings, the values of RT_{NDT} and Charpy upper-shelf energy must account for the effects of neutron radiation, including the results of the surveillance program of Appendix H of this part. The effects of neutron radiation must consider the radiation conditions (i.e., the fluence) at the deepest point on the crack front of the flaw assumed in the analysis.

1. Reactor Vessel Charpy Upper-Shelf Energy Requirements

a. Reactor vessel beltline materials must have Charpy upper-shelf energy[1], in the transverse direction for base material and along the weld for weld material according to the ASME Code, of no less than 75 ft-lb (102 J) initially and must maintain Charpy upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb (68 J), unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of Charpy upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code. This analysis must use the latest edition and addenda of the ASME Code incorporated by reference into § 50.55a(b)(2) at the time the analysis is submitted.

b. Additional evidence of the fracture toughness of the beltline materials after exposure to neutron irradiation may be obtained from results of supplemental fracture toughness tests, for use in the analysis specified in section IV.A.1.a.

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[1] Defined in ASTM E 185-79 and -82 which are incorporated by reference in Appendix H to Part 50.

c. The analysis for satisfying the requirements of section IV.A.1 of this appendix must be submitted, as specified in § 50.4, for review and approval on an individual case basis at least three years prior to the date when the predicted Charpy upper-shelf energy will no longer satisfy the requirements of section IV.A.1 of this appendix, or on a schedule approved by the Director, Office of Nuclear Reactor Regulation.

2. Pressure-Temperature Limits and Minimum Temperature Requirements

a. Pressure-temperature limits and minimum temperature requirements for the reactor vessel are given in Table 1, and are defined by the operating condition (i.e., hydrostatic pressure and leak tests, or normal operation including anticipated operational occurrences), the vessel pressure, whether or not fuel is in the vessel, and whether or not the core is critical. In Table 1, the vessel pressure is defined as a percentage of the preservice system hydrostatic test pressure. The appropriate requirements on both the pressure-temperature limits and the minimum permissible temperature must be met for all conditions.

b. The pressure-temperature limits identified as "ASME Appendix G limits" in Table 1 require that the limits must be at least as conservative as limits obtained by following the methods of analysis and the margins of safety of Appendix G of Section XI of the ASME Code.

c. The minimum temperature requirements given in Table 1 pertain to the controlling material, which is either the material in the closure flange or the material in the beltline region with the highest reference temperature. As specified in Table 1, the minimum temperature requirements and the controlling material depend on the operating condition (i.e., hydrostatic pressure and leak tests, or normal operation including anticipated operational

occurrences), the vessel pressure, whether fuel is in the vessel, and whether the core is critical. The metal temperature of the controlling material, in the region of the controlling material which has the least favorable combination of stress and temperature, must exceed the appropriate minimum temperature requirement for the condition and pressure of the vessel specified in Table 1.

d. Pressure tests and leak tests of the reactor vessel that are required by Section XI of the ASME Code must be completed before the core is critical.

B. If the procedures of Section IV.A. of this appendix do not indicate the existence of an equivalent safety margin, the reactor vessel beltline may be given a thermal annealing treatment to recover the fracture toughness of the material, subject to the requirements of § 50.66. The reactor vessel may continue to be operated only for that service period within which the predicted fracture toughness of the beltline region materials satisfies the requirements of Section IV.A. of this appendix using the values of RT_{NDT} and Charpy upper-shelf energy that include the effects of annealing and subsequent irradiation.

TABLE 1 PRESSURE AND TEMPERATURE REQUIREMENTS

OPERATING CONDITION	VESSEL PRESSURE (1)	REQUIREMENTS FOR PRESSURE-TEMPERATURE LIMITS	MINIMUM TEMPERATURE REQUIREMENTS
1. Hydrostatic pressure and leak tests (core is not critical):			
1.a Fuel in the vessel	≤ 20%	ASME Appendix G Limits	(2) + 60°F
1.b Fuel in the vessel	> 20%	ASME Appendix G Limits	(2) + 90°F (6)
1.c No fuel in the vessel (Preservice Hydrotest Only)	ALL	(Not Applicable)	(3) + 60°F
2. Normal operation (incl. heat-up and cool-down), including anticipated operational occurrences:			
2.a Core not critical	≤ 20%	ASME Appendix G Limits	(2)
2.b Core not critical	> 20%	ASME Appendix G Limits	(2) + 120°F (6)
2.c Core critical	≤ 20%	ASME Appendix G Limits + 40°F	larger of [(4)] or [(2) + 40°F]
2.d Core critical	> 20%	ASME Appendix G Limits + 40°F	larger of [(4)] or [(2) + 160°F]
2.e Core critical for BWR (5)	≤ 20%	ASME Appendix G Limits + 40°F	(2) + 60°F

(1) Percent of the preservice system hydrostatic test pressure.

(2) The highest reference temperature of the material in the closure flange region that is highly stressed by the bolt preload.

(3) The highest reference temperature of the vessel.

(4) The minimum permissible temperature for the inservice system hydrostatic pressure test.

(5) For boiling water reactors (BWR) with water level within the normal range for power operation.

(6) Lower temperatures are permissible if they can be justified by showing that the margins of safety of the controlling region are equivalent to those required for the beltline when it is controlling.

5. In 10 CFR Part 50, Appendix H is revised to read as follows:

Appendix H to Part 50 -- Reactor Vessel Material Surveillance Program Requirements

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I. Introduction

The purpose of the material surveillance program required by this appendix is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors which result from exposure of these materials to neutron irradiation and the thermal environment. Under the program, fracture toughness test data are obtained from material specimens exposed in surveillance capsules, which are withdrawn periodically from the reactor vessel. These data will be used as described in Section IV of Appendix G to this part.

ASTM E 185-73, -79, and -82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," which are referenced in the following paragraphs, have been approved for incorporation by reference by the Director of the Federal Register. Copies of ASTM E 185-73, -79, and -82, may

be purchased from the American Society for Testing and Materials, 1916 Race St., Philadelphia, PA 19103 and are available for inspection at the NRC Library, 11545 Rockville Pike, Two White Flint North, Rockville, Maryland 20852-2738.

II. Definitions

All terms used in this Appendix have the same meaning as in Appendix G.

III. Surveillance Program Criteria

A. No material surveillance program is required for reactor vessels for which it can be conservatively demonstrated by analytical methods applied to experimental data and tests performed on comparable vessels, making appropriate allowances for all uncertainties in the measurements, that the peak neutron fluence at the end of the design life of the vessel will not exceed 10^{17} n/cm^2 (E > 1 MeV).

B. Reactor vessels that do not meet the conditions of paragraph II.A of this appendix must have their beltline materials monitored by a surveillance program complying with ASTM E 185, as modified by this appendix.

1. The design of the surveillance program and the withdrawal schedule must meet the requirements of ASTM E 185-73 or the edition of ASTM E 185 that is current on the issue date of the ASME Code to which the reactor vessel was purchased, whichever is later. Later editions of ASTM E 185 may be used, but including only those editions through 1982. For each capsule withdrawal, the test procedures and reporting requirements must meet the requirements of ASTM E 185-82 to the extent practicable for the configuration of the specimens in the capsule.

2. Surveillance specimen capsules must be located near the inside vessel wall in the beltline region so that the specimen irradiation history duplicates, to the extent practicable within the physical constraints of the system, the neutron spectrum, temperature history, and maximum neutron fluence experienced by the reactor vessel inner surface. If the capsule holders are attached to the vessel wall or to the vessel cladding, construction and inservice inspection of the attachments and attachment welds must be done according to the requirements for permanent structural attachments to reactor vessels given in Sections III and XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). The design and location of the capsule holders must permit insertion of replacement capsules. Accelerated irradiation capsules may be used in addition to the required number of surveillance capsules.

3. A proposed withdrawal schedule must be submitted with a technical justification as specified in § 50.4. The proposed schedule must be approved prior to implementation.

C. Requirements for an Integrated Surveillance Program

1. In an integrated surveillance program, the representative materials chosen for surveillance for a reactor are irradiated in one or more other reactors that have similar design and operating features. Integrated surveillance programs must be approved by the Director, Office of Nuclear Reactor Regulation, on a case-by-case basis. Criteria for approval include the following:

a. The reactor in which the materials will be irradiated and the reactor for which the materials are being irradiated must have sufficiently similar design and operating features to permit accurate comparisons of the predicted amount of radiation damage.

b. Each reactor must have an adequate dosimetry program.

c. There must be adequate arrangement for data sharing between plants.

d. There must be a contingency plan to assure that the surveillance program for each reactor will not be jeopardized by operation at reduced power level or by an extended outage of another reactor from which data are expected.

e. There must be substantial advantages to be gained, such as reduced power outages or reduced personnel exposure to radiation, as a direct result of not requiring surveillance capsules in all reactors in the set.

2. No reduction in the requirements for number of materials to be irradiated, specimen types, or number of specimens per reactor is permitted.

3. After (the effective date of this section), no reduction in the amount of testing is permitted unless previously authorized by the Director, Office of Nuclear Reactor Regulation.

IV. Report of Test Results

A. Each capsule withdrawal and the test results must be the subject of a summary technical report to be submitted, as specified in § 50.4, within one year of the date of capsule withdrawal, unless an extension is granted by the Director, Office of Nuclear Reactor Regulation.

B. The report must include the data required by ASTM E 185, as specified in paragraph III.B.1 of this appendix, and the results of all fracture toughness tests conducted on the beltline materials in the irradiated and unirradiated conditions.

C. If a change in the Technical Specifications is required, either in the pressure-temperature limits or in the operating procedures required to meet the

limits, the expected date for submittal of the revised Technical Specifications must be provided with the report.

Dated at Rockville MD, this 26th day of September, 1994.

For the Nuclear Regulatory Commission.

John C. Hoyle

John C. Hoyle,
Acting Secretary of the Commission.

CONGRESSIONAL CORRESPONDENCE SYSTEM
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8. COMMENTS