NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

RIN 3150–AD57

Fracture Toughness Requirements for Light Water Reactor Pressure Vessels

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule.

SUMMARY: The Nuclear Regulatory Commission (NRC) is amending 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 amendments will 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.

EFFECTIVE DATE: January 18, 1996.

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SUPPLEMENTARY INFORMATION: On October 4, 1994 (59 FR 50513), the NRC published in the Federal Register a proposed amendment to clarify several items related to fracture toughness requirements for reactor pressure vessels (RPV) and to add a new section on thermal annealing of a reactor vessel to 10 CFR Part 50.

Background

Maintaining the structural integrity of the reactor pressure vessel of light-water-cooled reactors is a critical concern related to the safe operation of nuclear power plants. To assure the structural integrity of RPVs, NRC regulations and regulatory guides have been developed to provide analysis and measurements methods and procedures to establish that each RPV has adequate safety margin for continued operation. Structural integrity of a RPV 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 the thermal annealing requirements in this rule and several regulatory guides, the regulatory documents contribute to a comprehensive set of regulations and regulatory guidance pertaining to RPV integrity.

This final rule adds requirements for thermal annealing of the RPV as a method for mitigating the effects of neutron irradiation (10 CFR 50.66) and amends the following:

1. The Pressurized Thermal Shock (PTS) rule (10 CFR 50.61).
2. Appendix G of 10 CFR Part 50, ''Fracture Toughness Requirements.''
3. Appendix H of 10 CFR Part 50, ''Reactor Vessel Material Surveillance Program Requirements.''

Overview of the Final Rule

PTS Rule (10 CFR 50.61)

This amendment to the PTS rule makes three changes:

1. The rule incorporates in total, and therefore makes binding by rule, the method for determining the reference temperature, RT_{NDT}, including treatment of the unirradiated RT_{NDT} value, the margin term, and the explicit definition of ''credible'' surveillance data, which is currently described in Regulatory Guide 1.99, Revision 2.
2. The section is restructured to improve clarity, with the requirements section giving only the requirements for the value for the reference temperature for end of life fluence, RT_{PTS}. The method for calculating RT_{PTS} is moved to a new paragraph of the rule.
3. Thermal annealing is identified as a method for mitigating the effects of neutron irradiation, thereby reducing RT_{PTS}.

Thermal Annealing Rule (10 CFR 50.66)

The thermal annealing rule, 10 CFR 50.66, provides a consistent set of requirements for the use of thermal annealing to mitigate the effects of neutron irradiation and replaces the requirements for annealing in the current Appendix G of 10 CFR Part 50.

The final rule requires, prior to initiation of thermal annealing, submittal of a Thermal Annealing Report containing: (1) A Thermal Annealing Operating Plan, (2) a Requalification Inspection and Test Program, (3) a Fracture Toughness Recovery and Reembrittlement Trend Assurance Program, and (4) Identification of Unreviewed Safety Questions and Technical Specifications Changes. The report must be submitted at least 3 years before the date at which the limiting fracture toughness criteria in 50.61 and Appendix G to Part 50 would be exceeded. This 3-year period is specified to provide the NRC staff with sufficient time to review the thermal annealing program. Under § 50.66(a), the NRC will, within three years of submission of a licensee's Thermal Annealing Report, document its views on the plan, including whether thermal annealing constitutes an unreviewed safety question.

In order to provide for public participation in the regulatory process, Section 50.66(f)(1) requires that the NRC hold a public meeting a minimum of 30 days before the licensee starts to thermal anneal the reactor vessel. The Commission will notify and solicit comments from cognizant local and state governments, and will publish a notice in the Federal Register and in a forum, such as local newspapers, which is readily accessible to individuals in the vicinity of the site, in order to solicit comments from the public.

The thermal annealing operating plan must include an evaluation of the effects of temperature, and of mechanical and thermal stresses on the reactor and associated equipment such as containment, the biological shield, and attached piping, to demonstrate that the operability of the reactor will not be detrimentally affected. The bounding conditions of 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 as discussed below in the next paragraph.

Upon completion of the thermal annealing, the licensee must confirm in writing to the Director, Office of Nuclear Reactor Regulation (NRR), that the thermal annealing was performed in accordance with the Thermal Annealing Operating Plan and the Requalification Inspection and Test Program. Within 15 days of the licensee's written confirmation that the thermal annealing was completed in accordance with the

Thermal Annealing Plan, and prior to restart, the NRC shall: (1) Briefly document whether the thermal annealing was performed in compliance with the licensee's Thermal Annealing Operating Plan and the Requalification Inspection and Test Program, with the documentation to be placed in the NRC public document room, and (2) hold a public meeting to: (1) permit the licensee to explain the results of the reactor vessel annealing to the NRC and the public, (2) allow the NRC to discuss its inspection of the reactor vessel annealing, and (3) provide an opportunity for the public to comment to the NRC on the thermal annealing. The licensee may restart its reactor after the meeting has been completed, unless the NRC orders otherwise. Within 45 days of the licensee's written confirmation that the thermal annealing was completed in accordance with the Thermal Annealing Operating plan and the Requalification Inspection and Test Program, the NRC staff shall complete full documentation of the NRC's inspection of the licensee's annealing process and place the documentation in the Public Document Room.

If the thermal annealing was completed but not performed in accordance with the Thermal Annealing Operating Plan and the Requalification Inspection and Test Program, including the bounding conditions of the temperature and times as discussed above, the licensee must submit a summary of lack of compliance and a justification for subsequent operations. The licensee must also identify any changes to the facility which are attributable to the noncompliances which constitute unreviewed safety questions and any changes to the technical specifications which are required for operation as a result of the noncompliances. This identification does not relieve the licensee from complying with applicable requirements of the Commission regulations and the operating license, and if, as a result of the annealing operation, these requirements cannot be met, the licensee must obtain the appropriate exemption per 10 CFR 50.12. If unreviewed safety questions or changes to technical specifications are not identified as necessary for resumed operation, the licensee may restart after the NRC staff places a summary of its inspection of the thermal annealing in the Public Document Room, and the NRC holds a public meeting on the thermal annealing. On the other hand, if unreviewed safety questions or changes to technical specifications are identified as necessary for resumed operation, the licensee may restart only after the Director of NRR authorizes restart, the summary of the NRC staff inspection is placed in the public document room, and a public meeting on the thermal annealing is held.

The final Thermal Annealing Rule also sets forth the requirements that a licensee must follow if the thermal annealing was terminated prior to completion. In general, the process and requirements for partial annealing are analogous to the situations where the thermal annealing was completed; *viz.*, where the partial annealing was otherwise performed in compliance with the Thermal Annealing Operating Plan and relevant portions of the Requalification Inspection and Test Program, the licensee submits written confirmation of such compliance and may restart following, *inter alia,* holding of a public meeting on the annealing. By contrast, where the partial annealing was not performed in accordance with the Thermal Annealing Operating Plan and relevant portions of the Requalification Inspection and Test Program, the licensee is required to submit a summary of lack of compliance and a justification for subsequent operations, and identify any changes to the facility which are attributable to the noncompliances which constitute unreviewed safety questions and changes to the technical specifications which are required for operation as a result of the noncompliances with the Thermal Annealing Operating Plan and relevant portions of the Requalification Inspection and Test Program. If Unreviewed Safety Questions and/or changes to technical specifications are identified as necessary for resumed operation, the licensee may restart only after the Director of NRR authorizes restart and the public meeting on the thermal annealing is held.

Every licensee that either completes a thermal annealing or terminates an annealing but elects to take full or partial credit for the annealing shall provide a Thermal Annealing Results Report detailing: (1) The time and temperature profile of the actual thermal anneal, (2) the post-anneal RT_{NDT} and Charpy upper shelf energy values of the reactor material to be used in subsequent operations, (3) the projected post-anneal reembrittlement trends for both RT_{NDT} and Charpy upper-shelf energy, and (4) the projected values of RT_{PTS} and Charpy upper-shelf energy at the end of the proposed period of operation addressed in the application. The report must be submitted within three months of completing the thermal anneal, unless an extension is authorized by the Director, NRR.

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

Two methods provided for evaluating annealing recovery are experimental methods to determine plant-specific annealing recovery, and a third method is a generic computational method. Experimental methods and the computational method are also provided 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. This final rule requires that surveillance specimens from ''credible'' surveillance programs must be used to develop plant-specific recovery data, if such specimens are available. This final rule does not require the removal of material from the RPV beltline to permit plant-specific evaluation of recovery.

As described previously, the computational method requires appropriate justification.

Post anneal reembrittlement trends of both the RT_{NDT} and the Charpy upper shelf energy must be estimated and monitored using a surveillance program described in the Thermal Annealing Report.

The reactor pressure vessel is perhaps the most important single component in the reactor coolant system. As such, ensuring its integrity is a fundamental element of plant safety. Thermal annealing is a positive action that could be taken to reduce the level of embrittlement in the pressure vessel beltline and, thereby, improve the ability of a pressure vessel to withstand accident loadings. While thermal annealing is a positive action, there are numerous complex technical questions regarding its application in the U.S. that are unanswered.

Thermal annealing of a commercial reactor pressure vessel has never been accomplished in the United States. Thermal annealing has been successfully employed in Eastern Europe and Russia on Russian-designed pressure vessels. However, there are significant differences between the U.S. and Russian designs in terms of the geometry of the pressure vessels, the attached piping, and the surrounding structures. The staff has observed one of these annealing operations. While informative, the East European and Russian experience does not provide answers to all of the potential questions related to annealing of U.S. designed pressure vessels.

Research analyses performed previously indicated the potential for plastic deformation of the main coolant piping for a typical U.S. plant design and anticipated annealing conditions. There are also questions regarding how thermal growth of the pressure vessel is treated, and the adequacy of the thermal and stress analyses used to predict response of the overall system under thermal annealing conditions. Additionally, there may be questions in other areas such as temperature limits for the concrete structures, and potential radiological hazards associated with removing and storing the reactor internals during the annealing process, and fire hazards associated with heating the vessel.

Recognition of the numerous complex technical questions related to thermal annealing, and of the potential benefits for operating nuclear power plants, has resulted in a cooperative effort, funded by the U.S. Department of Energy and the industry, to perform Annealing Demonstration Projects. Projects are planned to demonstrate two different annealing processes, evaluating heater designs and vessel designs. It is anticipated that the annealing demonstration projects will answer many of the generic questions regarding thermal annealing of U.S. pressure vessel and piping designs.

The thermal annealing report, required by the thermal annealing rule, is designed to facilitate a detailed review by the licensee of plant-specific questions and considerations in performing a thermal annealing. The proposed rule specifically discusses the potential for unreviewed safety questions and technical specification changes that may result from or be related to thermal annealing of the reactor pressure vessel. With completion of the demonstration projects and as the staff and industry gain experience with thermal annealing, many of the issues related to annealing will be better understood and related questions will be answered. However, until this experience is realized, the staff will critically review licensee determinations regarding unreviewed safety questions and the need for technical specification changes associated with each proposed thermal annealing.

The thermal annealing rule has been structured to provide time for the staff to thoroughly review the licensee's annealing plan and determination regarding unreviewed safety questions and the need for technical specification changes. If the staff identifies an unreviewed safety question or the need for a technical specification change, the licensee would be so notified and the existing NRC regulatory practices would be invoked to address the issues.

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 amendments to Appendix G are principally of a clarifying or a restructuring nature. Requirements for ''volumetric inspection'' and ''additional evidence of fracture toughness'' have been removed because they were 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 is incorporated in the ''equivalent margins'' analysis in Section IV.A.1 as a provisional method for developing fracture toughness data needed for that analysis.

The pressure-temperature and minimum permissible temperature requirements in Section IV have been restructured. The principal feature is 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 has been 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. An explicit statement has been added to require 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.

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, in the conduct of ASME, Section XI pressure and leak tests. The use of nuclear heat before the completion of such tests is not consistent with basic defense-in-depth nuclear safety principle 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 discussed in a letter to Messrs. Reynolds and Stenger of the Nuclear Utility Backfitting and Reform Group from James M. Taylor, Executive Director of Operations, dated February 2, 1990.

The current requirements in 10 CFR Part 50, Appendix G, Section V. D. with respect to reactor vessel thermal annealing are being replaced by a sentence which references the new Thermal Annealing rule, 10 CFR 50.66.

Appendix H of 10 CFR Part 50

Appendix H of 10 CFR Part 50, ''Reactor Vessel Material Surveillance Program Requirements'' provides the rules for monitoring the changes in the fracture toughness properties of the RPV beltline materials due to irradiation embrittlement using a surveillance program. 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.

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.'' This provision was deleted, although previous authorizations granted by the Director, Office of Nuclear Reactor Regulation, continue in effect.

A second 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 the subsequent testing and reporting of results from the surveillance capsules. Because 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.

Request for Public Comments

At the request of the Commission, the proposed rule contained a request for public comments on the following specific 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.

The supplementary information section of the proposed rule also discussed the issue of opportunity for public participation in regulating thermal annealing of pressure vessels.

The response to the request for public comments on these issues, along with other items, are summarized below.

Summary of Comments

The following includes a summary of the comments received on the proposed rule, on the five issues identified by the Commission, and on the options for public participation in thermal annealing.

Comments were received from nine separate sources. These sources consist of five utilities, the Nuclear Energy Institute (NEI), the Nuclear Utility Backfitting and Reform Group (NUBARG) represented by the firm Winston & Strawn, one public citizens group (Ohio Citizens for Responsible Energy (OCRE)), and one nuclear steam system supplier (NSSS).

NEI provided detailed comments on 10 CFR 50.61, 10 CFR 50.66, Appendix G to 10 CFR Part 50, and Appendix H to 10 CFR Part 50, responded to the request for comments on the five issues related to thermal annealing and included detailed comments on the opportunities for public participation. The five utilities and the NSSS endorsed the NEI comments. Three of the five utilities provided additional comments on 10 CFR 50.61; one of the five utilities provided additional comments on 10 CFR Part 50, Appendix G; two of the utilities provided additional comments on 10 CFR Part 50, Appendix H; and one of the five utilities disagreed with the NEI position on the opportunity for public participation and submitted a separate comment. OCRE provided comments on the opportunity for public participation. NUBARG provided comments on the backfitting aspects of the proposed rule and the staff's backfit justification.

NEI and one of the utilities included comments on the Draft Regulatory Guide DG–1027, ''Format and Content of Application for Approval for Thermal Annealing of Reactor Pressure Vessels,'' that was discussed in the proposed rule. These comments on Draft Regulatory Guide DG–1027 are being reviewed by the NRC staff and will be addressed separately in the resolution of comments on the regulatory guide.

The NRC reviewed the comments received on the proposed rule, the comments on the five questions related to thermal annealing and the issue of opportunities for public participation. The resolution of these comments is presented below.

PTS Rule (10 CFR 50.61)

Sixteen specific comments in the submittals from NEI and three utilities addressed 10 CFR 50.61. A general comment argued that both the existing 10 CFR 50.61 and the proposed modifications contained an excessive amount of prescriptive technical detail that limits licensee compliance flexibility. The commenters proposed that these prescriptive technical details be removed from the rule and placed in a regulatory guide. These commenters suggested that the rule not be issued until it has been written to contain only those requirements essential to regulate reactor pressure vessel embrittlement. A number of comments suggested changes that were clarifications to the proposed rule, including proposals to clarify the procedure for calculating the reference temperatures in the preservice condition, RT_{NDT}, and, at end of reactor life, RT_{PTS}. One comment noted that the proposed rule omitted part of the procedure in Regulatory Guide 1.99, presently being applied by the NRC, that permits adjustments for differences in chemistry between surveillance material and the vessel material when using credible surveillance data to calculate a best fit chemistry factor for transition temperature shifts due to irradiation. Several comments proposed changes in the criteria for establishing whether surveillance material data is credible that would result in a less restrictive basis for using surveillance data in determining the transition temperature shift. The comments argued that the proposed rule is ambiguous with respect to the use of information from other sources that contain limiting material for a specific plant and that the NRC must have the flexibility to approve use of such information on a case-by-case basis. Several comments proposed limiting the basis for making changes of RT_{PTS} subject to the approval of the Director, NRR.

The NRC recognizes that 10 CFR 50.61 contains an unusual amount of prescriptive material and that the comments proposing simplification have merit. Some changes to the rule have been made to provide flexibility, where appropriate. The NRC staff is evaluating subsequent changes that would be more performance based. However, the NRC staff believes that this rule, as written, is needed to ensure that plants apply the appropriate method for determining RT_{PTS} and that the appropriate reference to the thermal annealing rule be applied for the pressurized thermal shock situation.

A number of clarifications were made to the rule. The paragraphs dealing with the determination of RT_{PTS} were modified to make clear that RT_{PTS} is a unique, end of life, case of RT_{NDT} and to clarify the procedure for determining these values. As suggested, the adjustment procedure was added to the rule to permit accounting for differences in chemistry between surveillance materials and reactor vessel materials when calculating chemistry factors. With respect to the plant specific material surveillance data that is permitted to be used in a surveillance program, the rule was modified to make clear that such data includes results from other plant's surveillance programs and test reactors. Several clarifications were made to the criteria for determining credible material. The NRC determined that the requirements for approval by the Director, NRR, for changes in RT_{PTS} are appropriate and should not be modified.

Thermal Annealing Rule (10 CFR 50.66)

Twelve individual comments were received on the proposed Thermal Annealing Rule, 10 CFR 50.66. These comments included a number of suggestions for clarification of details of the proposed rule. Three of the comments addressed the requirements that, after the annealing operation, the reembrittlement rate of the reactor vessel due to neutron irradiation must be estimated and must be monitored using a surveillance program which conforms to Appendix H of 10 CFR 50, ''Reactor Vessel Materials Surveillance Program.'' The comments are summarized as follows:

(1) The supplementary information section for the proposed rule is silent on what is acceptable if limiting material is not available. The rule should provide appropriate requirements on the method for monitoring reembrittlement after annealing for those plants that do not have limiting material for their surveillance program and the monitoring plans should be consistent with the preannealing surveillance program approved by the NRC staff;

(2) Appendix H does not define an acceptable post-anneal surveillance program, the reference to Appendix H should be deleted, and the post-anneal surveillance program should be defined in the annealing plan that is approved by the staff; and

(3) The term reembrittlement rate is unclear as to the period of time to be used for its determination, and a wording change is proposed for the requirement that would relate change in toughness to fluence accumulated after the anneal.

Three of the comments addressed the requirements in the proposed rule that the Thermal Annealing Operation Plan include time-temperature profiles which represent the annealing conditions that may not be exceeded during the annealing operation and are to be used for determining the amount of recovery of the fracture toughness of the material due to annealing. The comments suggested that, instead of a single time-temperature profile, bounding time and temperature conditions be established for the maximum values that would be used for thermal and stress analysis and to verify the re-qualification inspection and test program, and the minimum values that would be used to establish the amount of recovery of fracture toughness and for reembrittlement rate estimates. The bounding values would be based upon the estimated uncertainties in the times and temperatures and the actual annealing conditions should fall within these bounds.

Two comments addressed the section on Certification of Annealing Effectiveness. One comment suggested deleting the requirement in the proposed rule for certification of the annealing effectiveness and instead adding a provision in the Thermal Annealing Operating Plan that approval prior to subsequent power operation be required only if the anneal was not performed in accordance with the approved plan. The comment also suggested that, if the licensee terminates the annealing before achieving the specified time but otherwise maintains the annealing envelop such that no concern exists for stress or thermal damage, no additional constraints be imposed on subsequent operations and no credit be given for annealing. The second comment suggested that (1) the staff's review of the annealing report (certification report) need not be completed prior to reinitiating power operation if the anneal was performed in accordance with the approved Thermal Annealing Operating Plan, (2) reporting and quantification of the actual recovery results need not be reported unless the vessel was at or above the PTS screening criteria when annealing was started, and (3) the Thermal Annealing Operating Plan should specify the minimum content and a schedule for reporting the annealing results. The commenter provided a proposed list of criteria, content, and schedule for reporting the annealing results.

One comment stated that no guidance was provided in the proposed rule on what constitutes components ''affected'' by the annealing operation that are required to be reported in the Thermal Annealing Operating Plan. The comment suggested alternative wording that components to be reported should be structures and components that are expected to experience significant temperature gradient or stress variations during the thermal annealing operation. One comment suggested qualifying the provision in the proposed rule that the effects of localized high temperatures must be evaluated for changes in thermal and mechanical properties of the reactor vessel insulation for those cases where such changes may be negligible at annealing conditions. One comment suggested that the use of applicable material data, such as data from integrated surveillance programs, be an optional part of the computational methods for determining fracture toughness recovery.

The NRC reviewed the comments received on the proposed rule in detail. After consideration, the NRC reached the conclusion that most of the comments are not inconsistent with the intent of the proposed rule and in some cases reflect a need for clarification of the rule. In these cases, alternative wording that clarified the intent of the rule was substituted in the text. With respect to the comments on the requirement that reembrittlement rate after annealing must be monitored using a surveillance program, the NRC is aware that some plants do not have limiting materials for their existing preannealing surveillance programs. For these situations the staff has approved alternative surveillance plans on a case-by-case basis. Clearly, these plants will not have limiting material for surveillance programs for use in determining reembrittlement rates after annealing.

The NRC recognizes that Appendix H of 10 CFR Part 50, which is referenced in this rule, does not specifically address the surveillance of an annealed reactor vessel. However, the requirements of Appendix H to 10 CFR Part 50 apply to all reactors including the specific case of an annealed reactor vessel. To clarify the surveillance requirements of an annealed plant, the final rule has been modified to include, as suggested, that the post-anneal reembrittlement is to be monitored using a surveillance program defined in the Thermal Annealing Report and that the surveillance program must conform to the intent of Appendix H to 10 CFR Part 50.

The term reembrittlement ''rate'' in the proposed rule was intended to mean the projected amount of reembrittlement over a specific fluence period. It is recognized that reembrittlement is not a straight line function of fluence. Determination of reembrittlement rate is discussed in more detail in Draft Regulatory Guide 1.162, ''Format and Content of Report for Thermal Annealing of Reactor Pressure Vessels.'' In Regulatory Guide 1.162, the approved method for estimating the reembrittlement rate, the lateral shift method, results in the same embrittlement trend as that used for the pre-anneal operating period. To avoid confusion the term ''rate'' has been changed to ''trend'' in the final rule and the regulatory guide.

The NRC agrees with the comments that the time and temperature profile required in the annealing operating plan should be bounding values. In this regard, Regulatory Guide DG–1027 calls for the thermal annealing operating plan to include identification of the limitations and permitted variations in temperature, time, heatup and cooldown rate. For clarification, the final rule has been modified to use the terms ''bounding conditions for times and temperatures and heatup and cooldown schedules'' to describe conditions that may not be exceeded during the annealing operation, and the lower limit time and temperature of the actual anneal is used for determining the projected recovery of fracture toughness by annealing.

The NRC considers that the intent of paragraphs (c), Completion or Termination of Thermal Annealing, and (d), Thermal Annealing Results Report, of the final rule to be consistent with the two comments on that subject. The final rule does not require that the NRC approve restart following the annealing operation if the Thermal Annealing Operating Plan and the Requalification Inspection and Test Program was complied with. The NRC accepts the suggestion that the rule should be more specific on the items the licensee should include in the report and has included the list in the final rule.

Finally, the NRC agrees with the suggestion to make clear that a report is not required if:

(1) The licensee terminates the anneal prior to completion;

(2) The partial anneal was otherwise in accordance with the Thermal Annealing Plan;

(3) The licensee does not elect to take credit for any recovery. A statement was added to the Final Rule to cover the early termination situation.

The NRC has accepted the suggested clarifications of what constitutes an ''affected'' component and the qualification on the requirement to evaluate changes in properties on reactor vessel insulation if these are negligible. The NRC considers it unnecessary to include a reference in the rule to data from integrated surveillance programs as an optional part of the computational methods to determine fracture toughness recovery. Generic computational methods for this purpose are provided in the Regulatory Guide 1.162. However, the final rule does not prohibit use of alternative methods if adequate justification is provided.

Appendix G to 10 CFR Part 50

Two comments were received on the Appendix G to 10 CFR Part 50 of the proposed rule. The NEI comment, which was endorsed by five utilities and one NSSS organization, included a table with six items on Appendix G. The other comment on Appendix G was received from one of the five utilities. Two of the comments identified typographical errors and suggested a change in organization to improve clarity. One of the comments suggested revising the rule to change the definition of reference temperature, RT_{NDT}, for cases where plants do not have data to comply with code procedures for determining RT_{NDT}. One comment suggested a change in the title of Table 1, ''Pressure and Temperature Requirements,'' by adding to the title ''For the Reactor Pressure Vessel'' to make clear that this table does not apply to other components in the reactor coolant pressure system and proposed adding a footnote to the table for the same purpose. One comment identified an error in the minimum temperature requirements for the hydrostatic and leak testing of the pressure vessel without fuel when the vessel pressure is equal or below 20 percent of the vessel design pressure. One of the comments suggested that two of the entries in the table were new requirements when the table was intended to provide clarification. The utility's comment disagreed with the proposed rule change to prohibit the use of nuclear heat for the performance of vessel leak and hydrostatic testing. The utility contended that using nuclear heat, by providing a significant temperature margin above the pressure and temperature limit curves, greatly reduces the probability of brittle fracture and should be allowed.

The NRC corrected the typographical errors and corrected the minimum temperature requirement for the hydrostatic and leak testing of the pressure vessel at low vessel pressures and without fuel. The title to Table 1 was changed, as suggested, for clarification.

The NRC does not agree with the proposal to change the definition of RT_{NDT}. The situation described in the comment, when data is not available to comply with code procedures, is presently handled on a case-by-case basis in accordance with MEB Branch position, MEB 5–2. The NRC staff does not agree with the comment that the two requirements cited are new requirements. Item 2.2.c. and Item 2.2.d of Table 1 are in the existing ASME code requirement and in Paragraph IV.A.3. in the rule. The NRC also does not agree with the utility's comment that using nuclear heat greatly reduces the probability of brittle fracture. The reasons for this are set forth in the February 2, 1990, letter to Messrs. Reynolds and Stenger of NUBARG from James M. Taylor, Executive Director for Operations.

Appendix H to 10 CFR Part 50

Three comments were received on Appendix H to 10 CFR 50. The comment from NEI was endorsed by the five utilities and the NSSS. Two of the five utilities submitted additional comments. NEI and one utility commented that the proposed change to Paragraph III.B.1, which establishes the applicable edition of ASTM standard E 185 for a reactor surveillance program, constituted a backfit that would require a substantial design change in the surveillance program for those plants fabricated to a code edition prior to 1973. The other two commenters suggested new changes to Appendix H to 10 CFR Part 50. One of the commenters noted that an existing provision in Appendix H to 10 CFR Part 50, not part of the proposed rule change, dealing with requirements for attaching capsule holders to the vessel wall is a reiteration of a requirement in the ASME Code and should be removed. The other commenter suggested a new change to Appendix H to 10 CFR Part 50 to add a statement to the criteria for approval of an integrated surveillance program that would permit the use of surveillance specimens for extension of license purposes. The commenter also suggested that there is an apparent conflict between Paragraph III.C.2. and Paragraph III.C.3. that address requirements for an integrated surveillance.

The provision in the proposed rule was changed and reference to ASTM E 185 73 was deleted to make clear that the surveillance programs must be designed to the edition of ASTM 185 that is current on the issue date of the ASME Code to which the reactor vessel was purchased or to a later edition through 1982. The Commission agrees with the industry comments that imposing the ASTM E 185 1973 edition is impractical because vessels purchased prior to 1973 could not necessarily comply with all of the surveillance requirements in the 1973 edition of the ASTM standard. The NRC staff believes that the provision in the present rule on requirements for attaching capsule holders to the reactor vessel wall is required for clarity and should not be deleted. The comments related to the requirements for an integrated surveillance program were not persuasive to the NRC staff. The existing provisions of the rule do not preclude the application of the integrated surveillance program for extension of license purposes. The two paragraphs purported to be in conflict address separate items; one addresses the number of materials to be irradiated, specimen types, and number of specimens per reactor; the other addresses amount of testing.

Request for Comments on Issues Related to Thermal Annealing

Comments were received from NEI on the five issues on thermal annealing that were included in the proposed rule at the Commission's direction. In addition, OCRE and one utility, Pacific Gas and Electric, submitted comments on Issue 4, concerning the preferred regulatory process (including opportunity for public participation). Public Comments on the five issues are summarized below:

Issue 1: The technical adequacy of the NRC staff's guidance.

Comment: The detailed comments submitted on 10 CFR 50.66 are summarized in the Summary of Comments section on the Thermal Annealing Rule. In addition, NEI suggested that draft Regulatory Guide, DG–1027, be revised to include acceptance criteria where an action is required, but the acceptance criteria was not defined. NEI further commented that the re-embrittlement rate equation (DG–1027, Equation 1) appeared to be very conservative and would result in a post-anneal operating life that is less than industry believes justified.

Response: The NRC is concurrently revising the noted draft regulatory guide and will address this comment in the resolution of comments for the guide.

Issue 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.

Comment: NEI noted that ''The reactor pressure vessel thermal annealing rule and guide address appropriate issues to assure public health and safety and that the annealed reactor pressure vessel may be safely operated. The prior NRC staff approval of the reactor vessel annealing plan assures a clear process and criteria to restart following the vessel anneal. The licensee needs only to attest to compliance with the approved plan prior to resuming operations. The resumption of operations should not be needlessly delayed while a report documenting performance of the vessel anneal and recovery of the embrittled material properties is confirmed, because the vessel anneal will only improve the material properties. The final report should be submitted on a schedule that considers when the vessel would have exceeded the RT_{PTS} or uppershelf energy (USE) screening criteria without an anneal. The material property recovery will document prior to the time when the vessel would have exceeded the screening criteria, thereby assuring that the vessel is safe to operate at restart and for the duration justified by the material embrittlement recovery.''

Response: NRC agrees with the NEI comment, except NRC believes it is necessary for the licensee to submit the final report within three months of completing or terminating the anneal, unless an extension is authorized by the Director, Office of Nuclear Reactor Regulation.

Issue 3: Whether health and safety concerns are best served by approval of the thermal annealing plan or of readiness for restart.

Comment: NEI noted that ''The performance of a reactor pressure vessel anneal in accordance with an approved annealing plan improves the public health and safety by reducing the probability of core melt frequency. This improvement occurs because of the increase in reactor vessel material ductility. The amount of recovery achieved by a thermal anneal will be documented prior to the original date when the reactor vessel would have exceeded the PTS or USE screening limit. Therefore, a demonstration for ''restart readiness'' is an extra burden that will not provide any further improvement of the public health and safety.''

Response: The NRC's determination as to the procedures for NRC review of the Thermal Annealing Operation Plan, Requalification Inspection and Test Program and justification for restart discussed below in further detail in the Opportunities for Public Participation section.

Issue 4: The preferred regulatory process (including opportunities for public participation) and the commenter's basis for recommending a particular process.

Comment: NEI noted that ''The industry recommends that a hearing opportunity be provided, but that it be a non-adjudicatory, 10 CFR Part 2, Subpart L type hearing on the docketed record. The essential features of the hearing process proposed are as follows. The NRC would at time of receiving the licensee proposed annealing plan issue a Federal Register announcement that staff is performing the review per 10 CFR 50.66. A Subpart L hearing could be held, if requested by an intervener, after the NRC staff has issued a safety evaluation report on the licensee annealing plan, but prior to commencement of the reactor vessel thermal annealing unless the NRC staff makes a ''no significant hazards determination.'' Enclosure 4 provides additional details that support this industry position.'' Additional detailed comments by NEI and the comments on this subject by OCRE are discussed under the Opportunities for Public Participation heading.

Response: The rule provides for public participation in the regulatory process by incorporating a public meeting on the Licensee's Thermal Annealing Report a minimum of 30 days before the start of thermal annealing, and a public meeting after the licensee completes the anneal but before the reactor is restarted. The opportunity for public hearings in thermal annealing should be limited to those cases where there is an unreviewed safety question or a change to the Technical Specifications or where the licensee did not comply with the Thermal Annealing Operating Plan and Requalification Inspection and Test Program. Expanded discussion on this issue is provided below under the Opportunities for Public Participation heading.

Issue 5: Whether there are health and safety issues concerning thermal annealing that cannot be addressed generically and would warrant plant-specific consideration.

Comment: NEI noted that ''Thermal annealing to reduce material irradiation embrittlement is a well understood metallurgical phenomenon. The supporting thermal and stress analysis used to demonstrate that the vessel is not damaged during the anneal are standard technologies used at nuclear plants. Because thermal annealing uses well understood technology, public health and safety is reasonably assured.''

Response: The NRC agrees with this comment.

Opportunities for Public Participation

The Supplementary Information section of the proposed rule discussed the four options the Commission considered for structuring the regulatory process related to public participation in the NRC's review and approval of a licensee's proposal for thermal annealing of a reactor vessel. The proposed rule, at the Commission's direction, requested comments on the preferred regulatory process (including opportunities for public participation). The four options included:

(1) No hearings under the rule as proposed;

(2) Discretionary opportunity for hearing under rule as proposed in which situation the Commission would decide on a case-by-case basis to determine whether a hearing should be held;

(3) Required opportunity for hearing under rule as proposed, but work could commence if the NRC were to make a ''no significant hazard determination'' on the proposed thermal annealing; and

(4) Modify the proposed rule to require suspension of license prior and during the thermal annealing at which time no hearing would be afforded and the license would only be reinstated if the licensee demonstrates that it has addressed the reactor embrittlement such that it is acceptable to operate the plant.

Three comments were submitted on the subject. OCRE and NEI addressed all of the alternatives in detail and they, as well as one utility, identified and discussed individual preferred alternatives.

NEI commented that each of the four alternatives has a sufficiently serious flaw to prevent adoption. With respect to the no hearing alternative, NEI agrees that annealing is presently subject to approval by the Director of NRR in accordance with Part 50 Appendix G rather than being the subject of a license amendment as an unreviewed safety question under § 50.59. However, NEI believes that annealing is an important process from a regulatory standpoint and that public participation, in the form of informal hearings, is appropriate. NEI objected to a discretionary opportunity for a hearing because it provides significant uncertainty in the process for licensees and members of the public. NEI's objection to requiring a hearing, as discussed in staff Option 3, is that it would allow those who object to the resumption of operation, on other than technical grounds, to use hearings to delay restart. Option 4 is objectionable to NEI because it does not provide the licensee with any stability or predictability since the licensee would be required to demonstrate compliance after the annealing was performed, and does not provide the public with any opportunity to express its views.

NEI further commented that a license amendment is not necessary to approve a thermal annealing plan because annealing will not change the reactor vessel or other components in a manner inconsistent with the facility technical specifications nor will it require changes in the FSAR, and further, that a licensee is not required to modify its procedures to address or accommodate the annealing process. NEI noted that, while there is an incentive for the licensee to obtain credit for its improved P/T curves, and could seek a licensee amendment to do so, the licensee's existing P/T curves could remain in force.

Despite the conclusion that a license amendment is not necessary for thermal annealing, NEI recommended that a hearing opportunity be provided, but that it be a non-adjudicatory, Subpart L type hearing on the record. NEI gave the following advantages for this approach: (1) The NRC would be provided with a clear understanding of the licensee's annealing process, and the NRC's hearing process; (2) a Subpart L hearing is held on the written record and typically does not include the discovery or live testimony associated with adjudicatory hearings, but allows the public to participate in a meaningful way without consuming the vast NRC, licensee, and public resources required for an adjudicatory hearing; and (3) it would provide predictability and stability by ensuring that all issues which could be subject to a hearing are addressed prior to restart. Any inspection or test performed in order to restart would be for the purpose of confirming compliance with the rule.

OCRE supported the proposed rule provided that the public hearing rights were preserved with regard to reactor pressure vessel annealing. It is OCRE's position on the request for public comment that, based on the Sholly decision, the NRC must offer the opportunity for a formal adjudicatory hearing on the application for annealing and on the licensee's justification for subsequent operation where the licensee cannot certify that the thermal annealing was performed in accordance with the approved application. OCRE commented that approval by the Director of NRR of the application for annealing and restart of the reactor, if the licensee cannot certify that annealing was performed in accordance with the approved application, will give the licensee the authority to operate in ways in which they otherwise could not, and is thus, a de facto license amendment. OCRE fully supported Option 3 which requires opportunity for hearing under the rule as proposed. OCRE suggested that the adequacy of the thermal annealing plan, as well as the vessel's ability to perform its safety function after annealing, could be raised in the hearing on the thermal annealing plan and that the licensee's implementation of the thermal annealing plan could not commence until any hearing is concluded or unless the NRC makes a ''no significant hazards determination'' with respect to thermal annealing.

With respect to Option 1, OCRE concluded that the informal hearings or public meetings proposed by the Commission for the initial thermal annealing are not a substitute for adjudicatory hearings required by the Atomic Energy Act (AEA) and do not give the interveners the same rights as they would have in a Section 189a hearing. OCRE found Option 2 preferable to having no hearing. However, OCRE contended that this option is flawed by the assumption that ''Section 189a of the AEA does not afford an interested member of the public a right to request a hearing.'' They contend that approval by the Director, NRR to anneal the reactor pressure vessel or to restart after annealing does constitute a de facto operating licensing amendment for which the opportunity for a hearing is required. OCRE found Options 1 and 4 unacceptable in that they do not provide the opportunity for a formal adjudicatory hearing.

The comment from the utility suggested that Option 1 is the appropriate approach as long as the annealing process to be implemented is approved in advance by the NRC staff and the utility certifies that they have complied with the approved annealing process during the annealing operation, as provided for in the proposed rule. The utility further commented that if Technical Specifications changes or amendments to the operating license are required in order to perform the annealing then the opportunity for hearings would be required due to the normal license amendment process and if the final safety analysis report (FSAR) were required to be updated to reflect the thermal annealing process, the provisions of 10 CFR 50.59 would apply. The utility suggested that if those changes did not constitute an ''unreviewed safety question,'' no amendment would be needed and the license amendment process should not be invoked and that if a member of the public is concerned about a licensee's compliance with the NRC approved thermal annealing plan, those concerns could be addressed pursuant to the 10 CFR 2.206 petition process. The utility commented that, under its proposal, existing regulatory provisions for public participation would apply as appropriate and no new prescriptive requirements would be necessary.

The Commission has considered the public comments and has modified the proposed rule as follows. A licensee that seeks to utilize thermal annealing to mitigate the effects of neutron irradiation of the nuclear reactor vessel must, at least three years prior to the date at which the limiting fracture toughness criteria in § 50.61 or Appendix G to Part 50 would be exceeded, submit a Thermal Annealing Report to the NRC staff for review. The report shall contain four sections: (i) Thermal Annealing Operating Plan, (ii) Requalification Inspection and Test Program, (iii) Program for determining Fracture Toughness Recovery and Reembrittlement Trend, and (iv) a section identifying any changes to the description of the facility as described in the updated final safety analysis report (FSAR) which constitute unreviewed safety questions (USQs) under § 50.59, and changes to the facility's technical specifications, which are necessary either to perform the thermal annealing, or to operate following completion of the annealing. Section 50.66(a) provides that the NRC will, within three years of submission of a licensee's annealing report, document its views on whether the plan for conducting thermal annealing constitutes an unreviewed safety question or otherwise requires a change to the plant's technical specifications. Such a determination is the threshold determination for whether NRC approval is required before undertaking the activity. In the event the NRC were to conclude, contrary to the licensee, that an unreviewed safety question is present or a change to the technical specifications is necessary, the NRC would, as a discretionary enforcement matter, issue an appropriate order to the licensee prohibiting annealing prior to issuance of a license amendment. An opportunity for formal adjudicatory hearing would be provided in connection with the license amendment; however, if the NRC makes a finding that the proposed change to the FSAR description or technical specification constitutes a ''no significant hazards consideration'' pursuant to Section 189.(a)(2)(A), the licensee may conduct the thermal annealing prior to completion of any hearing. In any event, at least 30 days before the licensee starts to thermal anneal and before the NRC completes its review, the NRC will hold a public meeting on the licensee's proposed Thermal Annealing Plan and Requalification Inspection and Test Program.

Following the completion of the annealing operation, the licensee must confirm in writing to the Director, Office of Nuclear Reactor Regulation, that the thermal annealing was performed in accordance with the Thermal Annealing Operating Plan and the Requalification and Inspection Test Program. In support of this confirmation, the licensee must submit a report, within three months of completion or termination of the anneal, that presents the results of the annealing operation. Within two weeks of the licensee's written confirmation that the thermal annealing was completed in accordance with the Thermal Annealing Plan, and prior to restart, the NRC shall: (1) Place in its public document room a summary of the NRC staff's inspection of the licensee's thermal annealing process to confirm that the thermal annealing was completed in accordance with the Thermal Annealing Operating Plan and the Requalification Inspection and Test Program, and (2) hold a public meeting with the licensee to permit the licensee to explain the results of the reactor vessel annealing to the NRC and the public, for the NRC to discuss its inspection of the reactor vessel annealing process, and to provide an opportunity for the public to comment to the NRC on the annealing operation and the results of the Staff's inspection.

Within 45 days of the licensee's written confirmation that the thermal annealing was completed, the NRC shall complete full documentation of the NRC's inspection of the licensee's annealing process to confirm that the annealing was completed in accordance with the Thermal Annealing Operating Plan and the Requalification Inspection and Test Program.

The licensee may resume operation if: (1) The licensee concludes that the thermal annealing operation was performed in compliance with the Thermal Annealing Operating Plan, the Requalification Inspection and Test Program, and the provisions of Section 50.66(b), (2) a summary of the NRC's inspection of the thermal annealing is placed in the NRC public document room as required by Section 50.66(c) (2) and (3) the NRC holds the public meeting required by Section 50.66(f)(2), unless the staff takes action against the licensee. Since NRC approval to resume operation is not necessary, an opportunity for hearing would not be provided in this situation. If, however, the licensee cannot conclude that the thermal annealing was performed in compliance with the Thermal Annealing Operating Plan or the Requalification Inspection and Test Program, the licensee must submit a justification for continued operation to the Director. If the noncompliance presents an unreviewed safety question, as determined by the licensee or directed by the NRC following its review of the report, then the plant may not restart until the Director has approved restart. Those failures to comply with the Thermal Annealing Operating Plan and the Requalification Inspection and Test Program, which either (1) Are considered to be ''unreviewed safety questions'' or (2) require changes to the technical specifications as a result of the noncompliances, would also be subject to an opportunity for a formal adjudicatory hearing in accordance with the Commission's regulations governing license amendments. However, the licensee may restart prior to completion of the hearing if the Director makes a finding that such restart constitutes a ''no significant hazards consideration,'' as provided under Section 189.(a)(2)(A) of the Atomic Energy Act of 1954, as amended.

The regulatory process for thermal annealing and the associated hearing opportunities are consistent with long-standing NRC regulatory practices defining those matters which present sufficient potential effect on public health and safety (e.g., are unreviewed safety questions) to justify both prior NRC review of the change, and an opportunity for hearings (with the associated time and resource impacts on both the licensee and the NRC). With respect to the thermal annealing review process, the Commission reassessed the regulatory requirements and processes for assuring safety. The Commission determined that the most important safety matters are normally addressed in license conditions, technical specifications, and the FSAR. The regulatory process for NRC consideration of licensee-initiated changes concerning these matters, and the associated opportunities for hearings is in 10 CFR 50.59. In view of this well-established regulatory process for important safety information, the Commission determined that a regulatory process requiring NRC approval of a thermal annealing plan is not necessary, because the licensee is already required to comply with its license conditions, technical specifications, and FSAR. Important changes to license conditions, technical specifications, and FSAR from a safety standpoint are subject to both prior NRC review and approval and an opportunity for hearing. With respect to restart following completion of the annealing, the 15-day delay period should be sufficient time for review of the licensee's input given the NRC staff's understanding of the annealing operation plan prior to implementation, ongoing resident inspections and headquarters inspections of the implementation of thermal annealing operating plan. The Commission did not adopt NEI's suggestion for informal hearings where the Director must approve restart if the Thermal Annealing Operating Plan and Requalification Inspection and Test Program were not complied with, because the Commission does not see any distinction (in terms of safety implications) between the subject matter of hearings under this rule, as compared with other actions under Part 50 which would require formal hearings.

As discussed earlier in the supplementary information, previously performed research analyses indicated the potential for plastic deformation of the main coolant piping for a typical U.S. plant design and anticipated annealing conditions. There are also questions regarding how thermal growth of the pressure vessel is treated, and the adequacy of the thermal and stress analyses used to predict response of the overall system under thermal annealing conditions. Additionally, there may be questions in other areas such as temperature limits for the concrete structures, and potential radiological hazards associated with removing and storing the reactor internals during the annealing process, and fire hazards associated with heating the vessel.

Recognition of the numerous complex technical questions related to 4 thermal annealing and of the potential benefits for operating nuclear power plants has resulted in a cooperative effort, funded by the U.S. Department of Energy and the industry, to perform Annealing Demonstration Projects. Projects are planned to demonstrate two different annealing processes, evaluating heater designs and vessel designs. It is anticipated that the annealing demonstration projects will answer many of the generic questions regarding thermal annealing of U.S. pressure vessel and piping designs.

The Thermal Annealing Report, required by the thermal annealing rule, is designed to facilitate a detailed review by the licensee of plant-specific questions and considerations in performing a thermal annealing. The proposed rule specifically discusses the potential for unreviewed safety questions and technical specification changes that may result from or be related to thermal annealing of the reactor pressure vessel. With completion of the demonstration projects and as the staff and industry gain experience with thermal annealing, many of the issues related to annealing will be better understood and related questions will be answered. However, until this experience is realized, the staff will critically review licensee determinations regarding unreviewed safety questions and the need for technical specification changes associated with each proposed thermal annealing. The level of staff effort is expected to be significantly greater during its review of the initial proposed vessel annealings than that which will be required after experience is gained.

The thermal annealing rule has been structured to provide time for the staff to thoroughly review the licensee's annealing plan and determination regarding unreviewed safety questions and the need for technical specification changes. If the staff identifies an unreviewed safety question or the need for a technical specification change, the licensee would be so notified and the existing NRC regulatory practices would be invoked to address the issues.

Backfitting Issues

Comments were received on backfitting issues from the Nuclear Utility Backfitting and Reform Group (NUBARG). NUBARG commented that they do not object to the new NRC position in Appendix G to 10 CFR Part 50 which prohibits core criticality before completion of hydrostatic pressure and leak tests as a conservative measure to enhance safety. However, they are concerned that amending Appendix G on the basis of a compliance exception may set a bad precedent for avoiding backfitting analyses. NUBARG stated that ''The logic of the proposed rule would seem to allow the NRC to avoid a backfitting analysis by (1) invoking the intent of one requirement to override the explicit provisions of another, (2) using the compliance exception when the practice being eliminated seems specifically contemplated by and specified in the pertinent regulation, and (3) overlooking the fact that the NRC has apparently accepted this position in practice by some licensees * * *'' In NUBARG's view, this proposed amendment should be supported by a backfit analysis. The Commission has reviewed this comment and has concluded that use of the compliance exception under § 50.109 for the changes in Appendix G to 10 CFR Part 50 is appropriate. The Backfit Analysis section contains further discussion on this subject. The issue of explicitly prohibiting core criticality before completing pressure and leak tests has been addressed previously (letter from J. M. Taylor, EDO, to N. S. Reynolds and D. F. Stenger, NUBARG, dated February 2, 1990) and the NUBARG comment did not provide new information. 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).

NUBARG also commented on the amendment to Appendix H to 10 CFR Part 50 regarding surveillance that would preclude reducing the amount of testing if the initial test results agreed with predicted results. Although NUBARG recognizes the change would be prospective, it believes that NRC should provide flexibility to allow continued relief for any licensee who lacks such an authorization but has relied on the provision. The Commission believes that sufficient flexibility already exists in that licensees who do not have an authorization may seek an exemption under 10 CFR Part 50.12.

Another aspect of the backfitting concern raised by NUBARG addresses the proposed amendment to § 50.61 which, based on the adequate protection exception, would impose a uniform methodology for calculating the reference temperature. NUBARG contends that to rely on the adequate protection exception is arguably erroneous because the change in methodology is not likely an adequate protection issue (i.e., for most plants, the screening criteria will not be approached for many years). As discussed further under Backfit Analysis, the Commission believes that a new backfit analysis is not required for this conforming change, which corrects an inadvertent omission from the previous rulemaking. Therefore, the Commission concludes that the adequate protection basis for the backfit continues to apply from the previous rulemaking (56 FR 22300; May 15, 1991) to § 50.61.

Criminal Penalties

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

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 is not a major Federal action significantly affecting the quality of human environment and, therefore, an environmental impact statement is not required.

The individual actions covered in this final rule 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 serves 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 has 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 thermal annealing rule (10 CFR 50.66) permits and provides 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 final rule only applies when a licensee elects to use it. The final rule provides an alternative for assuring compliance with the requirements in 10 CFR 50.61 and Appendix G of 10 CFR Part 50.

The application of thermal annealing to a reactor vessel improves the condition of the reactor vessel material. In addition, this rule establishes requirements to avoid damaging the reactor system and to protect against accidents during the annealing operation.

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

Appendix G to 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 potential environmental impact of plant operation.

The restructuring of Sections IV and V of Appendix G is clarifying or corrective in nature, and 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 because 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 because Appendix G currently permits the use of thermal annealing to reduce fracture toughness loss of the RPV materials due to irradiation embrittlement.

Appendix H to 10 CFR Part 50

Concerning the amendments to Appendix H to 10 CFR Part 50 in the final rule, the requirement that all irradiation surveillance tests be made (i.e., no reduction in testing is permitted) will 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 final rule amends information collection requirements that are subject to the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 et seq.). These requirements were approved by the Office of Management and Budget, approval number 3150–0011.

The public reporting burden for this collection of information is estimated to average 6,000 hours per response, 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, DC 20555–0001, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB–10202, (3150–0011), Office of Management and Budget, Washington, DC 20503.

Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless it displays a currently valid OMB control number.

Regulatory Analysis

The NRC staff has prepared a regulatory analysis for the amendments to 10 CFR 50.61, Appendix G of 10 CFR Part 50, and Appendix H of 10 CFR Part 50 that describes the factors and alternatives considered by the Commission in deciding to issue 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 20555–0001. Single copies of the analysis may be obtained from Alfred Taboada, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, 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 this final rule will not have a significant economic impact on a substantial number of small entities. The rules which are affected by the amendments will: (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, the Small Business Size Standards in regulations issued by the Small Business Administration at 13 CFR Part 121, or the size standards established by the NRC at 10 CFR 2.810 (60 FR 18344; April 11, 1995).

Backfit Analysis

PTS Rule (10 CFR 50.61)

The revision to § 50.61 requires licensees to calculate RT_{PTS} using the same methodology specified in Regulatory Guide 1.99, Revision 2, for determining RT_{NDT}. This change was logically a requisite part of the previous rulemaking (56 FR 22300; May 15, 1991) to § 50.61 that set forth a unified method for calculating radiation embrittlement of the reactor beltline materials in Part 50. However, the Commission, at that time, inadvertently failed to make the conforming change to § 50.61. The Commission believes that the backfit statement for the previous amendment, which determined that the backfit was necessary to ensure that the facility continues to provide adequate protection to the public health and safety, is applicable to this conforming change to § 50.61.

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

The inclusion of thermal annealing in § 50.61 does not constitute a backfit as defined in 10 CFR 50.109(a)(1) because the decision to perform annealing is voluntary, no annealing has been conducted in this country, and there are no staff positions or Commission requirements relied upon by licensees that are being changed.

Thermal Annealing Rule (10 CFR 50.66)

The final thermal annealing rule establishes requirements with respect to applications for thermal annealing. However, the Commission has determined that the rule does not impose a ''backfit'' as defined in 10 CFR 50.109(a)(1). The thermal annealing rule does 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 Appendix G of CFR Part 50. However, these rules provide an alternative means for meeting these screening limits (e.g., 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. The rule does not reflect any new or different NRC staff position which conflicts with a prior NRC staff position or Commission rule. Thus, the final rule will have a purely prospective effect on future applications for thermal annealing. The Commission has stated in other rulemakings establishing prospective requirements (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. Accordingly, the Commission concludes that the rule does not impose backfits and a backfit analysis need not be prepared for the final thermal annealing rule.

Appendix G to 10 CFR Part 50

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 are of a clarifying nature.

10 CFR Part 50, Appendix G, Paragraph IV.2.d. of the final rule explicitly prohibits core criticality before completion of ASME Code hydrostatic pressure and leak tests. This is intended to make clear that licensees may not use nuclear heat in order to perform ASME Code hydrostatic tests. This amendment can be construed as a backfit, inasmuch as the prior version of 10 CFR Part 50, Appendix G, Paragraph IV.A.5 could be read to permit core criticality during ASME hydrostatic tests and Section XI of the ASME Code does not explicitly prohibit core criticality prior to completion of these tests. However, the Commission never intended the disputed language in Paragraph IV.A.5 of Appendix G to permit core criticality before successful completion of the required ASME hydrostatic tests. The scope of Appendix G is ''fracture toughness requirements'' only; that scope is stated clearly in the title of Appendix G, and Appendix G was not intended to specify system operational requirements. It is not correct, therefore, to interpret paragraph IV.A.5. as permitting nuclear hydrotesting. The final phrase in IV.A.5, ''depending on whether the core is critical during the test,'' was included in the rule for the sake of completeness, to specify appropriate fracture toughness requirements in the event that a licensee for some reason wanted to have the core critical during hydrotest, and was given approval to do so (e.g., as in the case of the Hatch units, where nuclear hydrotesting was allowed one last time as an approved exception.) The ASME Code's hydrostatic testing provisions for the reactor coolant pressure boundary (RCPB) provides the necessary assurance that GDC–14 is met. GDC–14 *inter alia* requires RCPB testing in order to provide an extremely low probability of RCPB failure, in terms of abnormal leakage, rapidly propagating failure, and gross rupture. Using heat produced by a critical reactor core to perform such testing essentially undercuts the basic safety principle embodied in GDC–14 that testing should be completed prior to nuclear reactor operation. It makes little sense to allow core criticality—thereby allowing the reactor to be in an operational condition where a loss of coolant could have significant consequences—prior to successful completion of tests that are intended to ensure that the probability of such coolant losses during such an operational condition are extremely low.[1] The ASME Code, Section XI, requires that the System Leakage Test be performed prior to plant startup following each refueling outage (Table-2500–1, Examination Category B–P, Note 2). The only way to interpret the ASME Code as permitting core criticality prior to completion of the hydrostatic tests is to read the term, ''plant startup'' as referring to something other than reactor criticality. This is neither the normal industry practice, nor has it been the NRC staff's longstanding interpretation of this provision of the ASME code. Indeed, it does not appear that the NRC staff has construed either Appendix G, Paragraph IV.A.5 nor Section XI of the ASME Code as permitting core criticality prior to successful completion of ASME Code hydrostatic tests. Moreover, the vast majority of nuclear utility licensees do not use nuclear heat to perform ASME code hydrostatic tests. This suggests that most licensees hold the same interpretation of Appendix G and Section XI of the ASME Code as the Commission. In sum, the Commission believes Section XI of the ASME Code, which is endorsed by 10 CFR 50.55a, implicitly prohibits core criticality prior to successful completion of hydrostatic testing. Therefore, the Commission concludes that the change in the language of Appendix G, Paragraph IV.2.d. is necessary to assure compliance with 10 CFR 50.55a and the ASME Code.

[1] The Commission is aware that NUBARG has presented an argument to the NRC that performance of ASME Code hydrostatic tests are more effective at the higher temperatures achieved when using nuclear heat, as compared with the heat sources normally employed by utilities in performing the hydrostatic tests. However, for the reasons set forth in the 1990 letter from James M. Taylor, EDO to N. S. Reynolds and D.F. Stenger, NUBARG, the Commission rejects this argument.

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).

Appendix H to 10 CFR Part 50

The amendments to Appendix H to 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).

List of Subjects in 10 CFR Part 50

Antitrust, Classified information, Criminal penalties, Fire protection, Intergovernmental relations, Nuclear power plants and reactors, Radiation protection, Reactor siting criteria, Reporting and record keeping 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. 552 and 553; the NRC is adopting the following amendments to 10 CFR Part 50.

PART 50—DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

1. The general authority citation for Part 50 is corrected to read as set forth below, and the section-specific authority citations continue 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. 1444, 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. In § 50.8, paragraph (b) is revised to read as follows:

§ 50.8 Information collection requirements: OMB approval.

* * * * *

(b) The approved information collection requirements contained in this part appear in §§ 50.30, 50.33, 50.33a, 50.34, 50.34a, 50.35, 50.36, 50.36a, 50.48, 50.49, 50.54, 50.55, 50.55a, 50.59, 50.60, 50.61, 50.63, 50.64, 50.65, 50.66, 70.71, 50.72, 50.73, 50.75, 50.80, 50.82, 50.90, 50.91, 50.120, and Appendices A, B, E, G, H, I, J, K, M, N, O, Q, and R, to this part.

* * * * *

3. 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.

(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 or other methods approved by the Director, Office of Nuclear Reactor Regulation.

(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.

(7) 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 calculation procedures given in paragraph (c)(1) of this section, except as provided in paragraphs (c)(2) and (c)(3) of this section. The assessment must specify the bases for the projected value of RT_{PTS} for each vessel beltline material, including the assumptions regarding core loading patterns, and must specify the copper and nickel contents and the fluence value used in the calculation for each beltline material. This assessment must be updated whenever there is a significant[2] change in projected values of RT_{PTS}, or upon 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 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

[2] 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.

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 calculated for each vessel beltline material using a fluence value, f, which is the EOL fluence for the material. 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) of this section.

(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) If a measured value of $RT_{NDT(U)}$ is not available, a generic mean value for the class [3] of material may be used if there are sufficient test results to establish a mean and a standard deviation for the class.

(ii) For generic values of weld metal, 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.

(iii) *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: $$M = 2\sqrt{\sigma_U^2 + \sigma_\Delta^2}$$

(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) of this section 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 need not exceed one-half of ΔRT_{NDT}.

(iv) 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) *CF* (°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 [4] 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 this paragraph, the EOL fluence for the vessel beltline material is used in calculating KRT_{PTS}.

(v) Equation 4 must be used for determining RT_{PTS} using equation 3 with EOL fluence values for determining ΔRT_{PTS}.

Equation 4: $RT_{PTS}=RT_{NDT(U)}+M+\Delta RT_{PTS}$

(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

[3] 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.

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

embrittlement. This information includes but is not limited to the reactor vessel operating temperature and any related surveillance program[5] 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 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, if present, must fall within the scatter band of the data base for the material.

(ii)(A) Surveillance data deemed credible according to the criteria of paragraph (c)(2)(i) of this section 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 5.

$$\text{Equation 5:} \quad CF = \frac{\sum_{i=1}^{n}\left[A_i \times f_i^{\left(0.28-0.10 \log f_i\right)}\right]}{\sum_{i=1}^{n}\left[f_i^{\left(0.56-0.20 \log f_i\right)}\right]}$$

(B) In Equation 5, ''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. If there is clear evidence that the copper and nickel content of the surveillance weld differs from the vessel weld, i.e. differs from the average for the weld wire heat number associated with the vessel weld and the surveillance weld, the measured values of ΔRT_{NDT} must be adjusted for differences in copper and nickel content by multiplying them by the ratio of the chemistry factor for the vessel material to that for the surveillance weld.

(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} need not exceed one-half of DRT_{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) of this section 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

[5] Surveillance program results means any data that demonstrates the embrittlement trends for the limiting beltline material, including but not limited to data from test reactors or from surveillance programs at other plants with or without surveillance program integrated per 10 CFR Part 50, Appendix H.

TABLE 1.—CHEMISTRY FACTOR FOR WELD METALS, °F—Continued

Copper, wt-%	Nickel, wt-%						
	0	0.20	0.40	0.60	0.80	1.00	1.20
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	208	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

4. A new § 50.66 is added under the center heading ''Issuance, Limitations, and Conditions of Licenses and Construction Permits'' 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 may be applied to the reactor vessel to recover the fracture toughness of the material. The use of a thermal annealing treatment is subject to the requirements in this section. A report describing the licensee's plan for conducting the thermal annealing must be submitted in accordance with § 50.4 at least three years prior to the date at which the limiting fracture toughness criteria in § 50.61 or Appendix G to Part 50 would be exceeded. Within three years of the submittal of the Thermal Annealing Report and at least thirty days prior to the start of the thermal annealing, the NRC will review the Thermal Annealing Report and place the results of its evaluation in its Public Document Room. The licensee may begin the thermal anneal after:

(1) Submitting the Thermal Annealing Report required by paragraph (b) of this section;

(2) the NRC places the results of its evaluation of the Thermal Annealing Report in the Public Document Room; and

(3) the requirements of paragraph (f)(1) of this section have been satisfied.

(b) Thermal Annealing Report. The Thermal Annealing Report must include: a Thermal Annealing Operating Plan; a Requalification Inspection and Test Program; a Fracture Toughness Recovery and Reembrittlement Trend Assurance Program; and Identification of Unreviewed Safety Questions and Technical Specification Changes.

(1) Thermal Annealing Operating Plan.

The thermal annealing operating plan must include:

(i) A detailed description of the pressure vessel and all structures and components that are expected to experience significant thermal or stress effects during the thermal annealing operation;

(ii) An evaluation of the effects of mechanical and thermal stresses and temperatures on the vessel, containment, biological shield, attached piping and appurtenances, and adjacent equipment and components to demonstrate that operability of the reactor will not be detrimentally affected. This evaluation must include:

(A) Detailed thermal and structural analyses 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.

(B) The effects of localized high temperatures on degradation of the concrete adjacent to the vessel and changes in thermal and mechanical properties, if any, of the reactor vessel insulation, and on detrimental effects, if any, on containment and the biological shield. If the design temperature limitations for the adjacent concrete structure are to be exceeded during the thermal annealing operation, an acceptable maximum temperature for the concrete must be established for the annealing operation using appropriate test data.

(iii) The methods, including heat source, instrumentation and procedures proposed for performing the thermal annealing. This shall include 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.

(iv) The proposed thermal annealing operating parameters, including bounding conditions for temperatures and times, and heatup and cooldown schedules.

(A) The thermal annealing time and temperature parameters selected must be based on projecting sufficient recovery of fracture toughness, using the procedures of paragraph (e) of this section, to satisfy the requirements of § 50.60 and § 50.61 for the proposed period of operation addressed in the application.

(B) The time and temperature parameters evaluated as part of the thermal annealing operating plan, and supported by the evaluation results of paragraph (b)(1)(ii) of this section, represent the bounding times and temperatures for the thermal annealing operation. If these bounding conditions for times and temperatures are violated during the thermal annealing operation, then the annealing operation is considered not in accordance with the Thermal Annealing Operating Plan, as required by paragraph (c)(1) of this section, and the licensee must comply with paragraph (c)(2) of this section.

(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 thermal annealing conditions of paragraph (b)(1)(iv) of this section have not been exceeded, and to determine the thermal annealing time and temperature to be used in quantifying the fracture toughness recovery. The requalification inspection and test program must demonstrate that the thermal 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 Trend Assurance Program. The percent recovery of RT_{NDT} and Charpy upper-shelf energy due to the thermal annealing treatment must be determined based on the time and temperature of the actual vessel thermal anneal. 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 trend determined in accordance with paragraph (b)(3)(ii) of this section. Recovery and reembrittlement evaluations shall include:

(i) Recovery Evaluations.

(A) The percent recovery of both RT_{NDT} and Charpy upper-shelf energy must be determined by one of the procedures described in paragraph (e) of this section, using the proposed lower bound thermal annealing time and temperature conditions described in the operating plan.

(B) 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 thermal annealing parameters used in the test program are equal to or bounded by those used in the vessel annealing operation.

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

(ii) Reembrittlement Evaluations.

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

(B) The post-anneal reembrittlement trend of both RT_{NDT} and Charpy upper-shelf energy must be estimated, and must be monitored using a surveillance program defined in the Thermal Annealing Report and which conforms to the intent of Appendix H of this part, ''Reactor Vessel Material Surveillance Program Requirements.''

(4) Identification of Unreviewed Safety Questions and Technical Specification Changes. Any changes to the facility as described in the updated final safety analysis report constituting unreviewed safety questions, and any changes to the technical specifications, which are necessary to either conduct the thermal annealing or operate the nuclear power reactor following the annealing, must be identified. The section shall demonstrate that the Commission's requirements continue to be complied with, and that there is reasonable assurance of adequate protection to the public health and safety following the changes.

(c) Completion or Termination of Thermal Annealing.

(1) If the thermal annealing was completed in accordance with the Thermal Annealing Operating Plan and the Requalification Inspection and Test Program, the licensee shall so confirm in writing to the Director, Office of Nuclear Reactor Regulation. The licensee may restart its reactor after the requirements of paragraph (f)(2) of this section have been met.

(2) If the thermal annealing was completed but the annealing was not performed in accordance with the Thermal Annealing Operating Plan and the Requalification Inspection and Test Program, the licensee shall submit a summary of lack of compliance with the Thermal Annealing Operating Plan and the Requalification Inspection and Test Program and a justification for subsequent operation to the Director, Office of Nuclear Reactor Regulation. Any changes to the facility as described in the updated final safety analysis report which are attributable to the noncompliances and constitute unreviewed safety questions, and any changes to the technical specifications which are required as a result of the noncompliances, shall also be identified.

(i) If no unreviewed safety questions or changes to technical specifications are identified, the licensee may restart its reactor after the requirements of paragraph (f)(2) of this section have been met.

(ii) If any unreviewed safety questions or changes to technical specifications are identified, the licensee may not restart its reactor until approval is obtained from the Director, Office of Nuclear Reactor Regulation and the requirements of paragraph (f)(2) of this section have been met.

(3) If the thermal annealing was terminated prior to completion, the licensee shall immediately notify the NRC of the premature termination of the thermal anneal.

(i) If the partial annealing was otherwise performed in accordance with the Thermal Annealing Operating Plan and relevant portions of the Requalification Inspection and Test Program, and the licensee does not elect to take credit for any recovery, the licensee need not submit the Thermal Annealing Results Report required by paragraph (d) of this section but instead shall confirm in writing to the Director, Office of Nuclear Reactor Regulation that the partial annealing was otherwise performed in accordance with the Thermal Annealing Operating Plan and relevant portions of the Requalification Inspection and Test Program. The licensee may restart its reactor after the requirements of paragraph (f)(2) of this section have been met.

(ii) If the partial annealing was otherwise performed in accordance with the Thermal Annealing Operating Plan and relevant portions of the Requalification Inspection and Test Program, and the licensee elects to take full or partial credit for the partial annealing, the licensee shall confirm in writing to the Director, Office of Nuclear Reactor Regulation that the partial annealing was otherwise performed in compliance with the Thermal Annealing Operating Plan and relevant portions of the Requalification Inspection and Test Program. The licensee may restart its reactor after the requirements of paragraph (f)(2) of this section have been met.

(iii) If the partial annealing was not performed in accordance with the Thermal Annealing Operating Plan and relevant portions of the Requalification Inspection and Test Program, the licensee shall submit a summary of lack of compliance with the Thermal Annealing Operating Plan and the Requalification Inspection and Test Program and a justification for subsequent operation to the Director, Office of Nuclear Reactor Regulation. Any changes to the facility as described in the updated final safety analysis report which are attributable to the noncompliances and constitute unreviewed safety questions, and any changes to the technical specifications which are required as a result of the noncompliances, shall also be identified.

(A) If no unreviewed safety questions or changes to technical specifications are identified, the licensee may restart its reactor after the requirements of paragraph (f)(2) of this section have been met.

(B) If any unreviewed safety questions or changes to technical specifications are identified, the licensee may not restart its reactor until approval is obtained from the Director, Office of Nuclear Reactor Regulation and the requirements of paragraph (f)(2) of this section have been met.

(d) Thermal Annealing Results Report. Every licensee that either completes a thermal annealing, or that terminates an annealing but elects to take full or partial credit for the annealing, shall provide the following information within three months of completing the thermal anneal, unless an extension is authorized by the Director, Office of Nuclear Reactor Regulation:

(1) The time and temperature profiles of the actual thermal annealing;

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

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

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

(e) 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 thermal 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 thermal 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[6] 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.

(f) Public information and participation.

(1) Upon receipt of a Thermal Annealing Report, and a minimum of 30 days before the licensee starts thermal annealing, the Commission shall:

(i) Notify and solicit comments from local and State governments in the vicinity of the site where the thermal annealing will take place and any Indian Nation or other indigenous people that have treaty or statutory rights that could be affected by the thermal annealing,

(ii) Publish a notice of a public meeting in the Federal Register and in a forum, such as local newspapers, which is readily accessible to individuals in the vicinity of the site, to solicit comments from the public, and

(iii) Hold a public meeting on the licensee's Thermal Annealing Report.

(2) Within 15 days after the NRC's receipt of the licensee submissions required by paragraphs (c)(1), (c)(2) and (c)(3)(i)–(iii) of this section, the NRC staff shall place in the NRC Public Document Room a summary of its inspection of the licensee's thermal annealing, and the Commission shall hold a public meeting:

(i) For the licensee to explain to NRC and the public the results of the reactor pressure vessel annealing,

(ii) for the NRC to discuss its inspection of the reactor vessel annealing, and

(iii) for the NRC to receive public comments on the annealing.

(3) Within 45 days of NRC's receipt of the licensee submissions required by paragraphs (c)(1), (c)(2) and (c)(3)(i)–(iii) of this section, the NRC staff shall complete full documentation of its inspection of the licensee's annealing process and place this documentation in the NRC Public Document Room.

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

Appendix G to Part 50—Fracture Toughness Requirements

I. Introduction and scope.
II. Definitions.
III. Fracture toughness tests.
IV. Fracture toughness requirements.

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 Street, New York, NY 10017, and are available for inspection at the NRC Library, 11545 Rockville Pike, Two White Flint North, Rockville, MD 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. RT_{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

[6] 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.

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.

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 3, 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 the core is critical. In Table 3, 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 3 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 3 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 3, 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 FOR THE REACTOR PRESSURE VESSEL

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])
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.

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

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

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

I. Introduction
II. Definitions
III. Surveillance Program Criteria
IV. Report of Test Results

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 Part 50.

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 Street, Philadelphia, PA 19103 and are available for inspection at the NRC Library, 11545 Rockville Pike, Two White Flint North, Rockville, MD 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 III.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 the edition of ASTM E 185 that is current on the issue date of the ASME Code to which the reactor vessel was purchased. 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 12th day of December, 1995.

For the Nuclear Regulatory Commission.

John C. Hoyle,

Secretary of the Commission.

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