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Standard Guide for In-Service Annealing of Light-Water Moderated Nuclear Reactor Vessels[1]

This standard is issued under the fixed designation ~~E509;~~E509/E509M; the number immediately following the designation indicates the year of original adoption or, in the case of revision, the year of last revision. A number in parentheses indicates the year of last reapproval. A superscript epsilon (ε) indicates an editorial change since the last revision or reapproval.

1. Scope

1.1 This guide covers the general procedures ~~to be considered~~ for conducting an in-service thermal anneal of a light-water moderated nuclear reactor vessel and demonstrating the effectiveness of the procedure. The purpose of this in-service annealing (heat treatment) is to improve the mechanical properties, especially fracture toughness, of the reactor vessel materials previously degraded by neutron embrittlement. The improvement in mechanical properties generally is assessed using Charpy V-notch impact test results, or alternatively, fracture toughness test results or inferred toughness property changes from tensile, hardness, indentation, or other miniature specimen testing (**1**).[2]

1.2 This guide is designed to accommodate the variable response of reactor-vessel materials in post-irradiation annealing at various temperatures and different time periods. Certain inherent limiting factors must be considered in developing an annealing procedure. These factors include system-design limitations; physical constraints resulting from attached piping, support structures, and the primary system shielding; the mechanical and thermal stresses in the components and the system as a whole; and, material condition changes that may limit the annealing temperature.

1.3 This guide provides direction for development of the vessel annealing procedure and a post-annealing vessel radiation surveillance program. The development of a surveillance program to monitor the effects of subsequent irradiation of the annealed-vessel beltline materials should be based on the requirements and guidance described in Practices E185 and E2215. The primary factors to be considered in developing an effective annealing program include the determination of the feasibility of annealing the specific reactor vessel; the availability of the required information on vessel mechanical and fracture properties prior to annealing; evaluation of the particular vessel materials, design, and operation to determine the annealing time and temperature; and, the procedure to be used for verification of the degree of recovery and the trend for reembrittlement. Guidelines are provided to determine the post-anneal reference nil-ductility transition temperature (RT_{NDT}), the Charpy V-notch upper shelf energy level, fracture toughness properties, and the predicted reembrittlement trend for these properties for reactor vessel beltline materials. This guide emphasizes the need to plan well ahead in anticipation of annealing if an optimum amount of post-anneal reembrittlement data is to be available for use in assessing the ability of a nuclear reactor vessel to operate for the duration of its present license, or qualify for a license extension, or both.

1.4 The values stated in ~~inch-pound or~~ either SI units or inch-pound units are to be regarded separately as ~~the~~ standard. The values stated in each system may not be exact equivalents; therefore, each system shall be used independently of the other. Combining values from the two systems may result in non-conformance with the standard.

1.5 *This standard does not purport to address all of the safety concerns, if any, associated with its use. It is the responsibility of the user of this standard to establish appropriate safety and health practices and determine the applicability of regulatory limitations prior to use.*

2. Referenced Documents

2.1 *ASTM Standards:*[3]

E185 Practice for Design of Surveillance Programs for Light-Water Moderated Nuclear Power Reactor Vessels

E636 Guide for Conducting Supplemental Surveillance Tests for Nuclear Power Reactor Vessels, E 706 (IH)

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[1] This guide is under the jurisdiction of ASTM Committee E10 on Nuclear Technology and Applications and is the direct responsibility of Subcommittee E10.02 on Behavior and Use of Nuclear Structural Materials.

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[2] The boldface numbers in parentheses refer to the list of references at the end of this standard.

[3] For referenced ASTM standards, visit the ASTM website, www.astm.org, or contact ASTM Customer Service at service@astm.org. For *Annual Book of ASTM Standards* volume information, refer to the standard's Document Summary page on the ASTM website.

E900 Guide for Predicting Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials, E706 (IIF)
E1253 Guide for Reconstitution of Irradiated Charpy-Sized Specimens
E2215 Practice for Evaluation of Surveillance Capsules from Light-Water Moderated Nuclear Power Reactor Vessels
2.2 *ASME Standards:*
Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Power Plant Components[4]
Code Case N-557, In-Place Dry Annealing of a PWR Nuclear Reactor Vessel (Section XI, Division 1)[4]
2.3 *Nuclear Regulatory Commission Documents:*
NRC Regulatory Guide 1.99, Revision 2, Effects of Residual Elements on Predicted Radiation Damage on Reactor Vessel Materials[5]
NRC Regulatory Guide 1.162, Format and Content of Report for Thermal Annealing of Reactor Pressure Vessels[5]

3. Significance and Use

3.1 Reactor vessels made of ferritic steels are designed with the expectation of progressive changes in material properties resulting from in-service neutron exposure. In the operation of light-water-cooled nuclear power reactors, changes in pressure-temperature ($P - T$) limits are made periodically during service life to account for the effects of neutron radiation on the ductile-to-brittle transition temperature material properties. If the degree of neutron embrittlement becomes large, the restrictions on operation during normal heat-up and cool down may become severe. Additional consideration should be given to postulated events, such as pressurized thermal shock (PTS). A reduction in the upper shelf toughness also occurs from neutron exposure, and this decrease may reduce the margin of safety against ductile fracture. When it appears that these situations could develop, certain alternatives are available that reduce the problem or postpone the time at which plant restrictions must be considered. One of these alternatives is to thermally anneal the reactor vessel beltline region, that is, to heat the beltline region to a temperature sufficiently above the normal operating temperature to recover a significant portion of the original fracture toughness and other material properties that were ~~lost~~degraded as a result of neutron embrittlement.

3.2 Preparation and planning for an in-service anneal should begin early so that pertinent information can be obtained to guide the annealing operation. Sufficient time should be allocated to evaluate the expected benefits in operating life to be gained by annealing; to evaluate the annealing method to be employed; to perform the necessary system studies and stress evaluations; to evaluate the expected annealing recovery and reembrittlement behavior; to develop and functionally test such equipment as may be required to do the in-service annealing; and, to train personnel to perform the anneal.

3.3 Selection of the annealing temperature requires a balance of opposing conditions. Higher annealing temperatures, and longer annealing times, can produce greater recovery of fracture toughness and other material properties and thereby increase the post-anneal lifetime. The annealing temperature also can have an impact on the reembrittlement trend after the anneal. On the other hand, higher temperatures can create other undesirable property effects such as permanent creep deformation or temper embrittlement. These higher temperatures also can cause engineering difficulties, that is, core and coolant removal and storage, localized heating effects, etc., in preventing the annealing operation from distorting the vessel or damaging vessel supports, primary coolant piping, adjacent concrete, insulation, etc. See ASME Code Case N-557 for further guidance on annealing conditions and thermal-stress evaluations (**2**).

3.3.1 When a reactor vessel approaches a state of embrittlement such that annealing is considered, the major criterion is the number of years of additional service life that annealing of the vessel will provide. Two pieces of information are needed to answer the question: the post-anneal adjusted RT_{NDT} and upper shelf energy level, and their subsequent changes during future irradiation. Furthermore, if a vessel is annealed, the same information is needed as the basis for establishing pressure-temperature limits for the period immediately following the anneal and demonstrating compliance with other design requirements and the PTS screening criteria. The effects on upper shelf toughness similarly must be addressed. This guide primarily addresses RT_{NDT} changes. Handling of the upper shelf is possible using a similar approach as indicated in NRC Regulatory Guide 1.162.Appendix X1 provides a bibliography of existing literature for estimating annealing recovery and reembrittlement trends for these quantities as related to U.S. and other country pressure-vessel steels, with primary emphasis on U.S. steels.

3.3.2 A key source of test material for determining the post-anneal RT_{NDT}, upper shelf energy level, and the reembrittlement trend is the original surveillance program, provided it represents the critical materials in the reactor vessel.[6]Appendix X2 describes an approach to estimate changes in RT_{NDT} both due to the anneal and ~~after~~ reirradiation. The first purpose of Appendix X2 is to suggest ways to use available materials most efficiently to determine the post-anneal RT_{NDT} and to predict the reembrittlement trend, yet leave sufficient material for surveillance of the actual reembrittlement for the remaining service life. The second purpose is to describe alternative analysis approaches to be used to assess test results of archive (or representative) materials to obtain the essential post-anneal and reirradiation RT_{NDT}, upper shelf energy level, or fracture toughness, or a combination thereof.

[4] Available from the American Society of Mechanical Engineers, 345 E. 47th Street, New York, NY 10017.
[5] Available from Superintendent of Documents, U.S. Government Printing Office, Washington, DC 20402.
[6] Consideration can be given to the reevaluation of broken Charpy specimens from capsules withdrawn earlier which can be reconstituted using Guide E1253 or from material obtained (sampled) from the actual pressure-vessel wall.

3.3.3 An evaluation must be conducted of the engineering problems posed by annealing at the highest practical temperature. Factors required to be investigated to reduce the risk of distortion and damage caused by mechanical and thermal stresses at elevated temperatures to relevant system components, structures, and control instrumentation are described in 5.1.3 and 5.1.4.

3.4 Throughout the annealing operation, accurate measurement of the annealing temperature at key defined locations must be made and recorded for later engineering evaluation.

3.5 After the annealing operation has been carried out, several steps should be taken. The predicted improvement in fracture toughness properties must be verified, and it must be demonstrated that there is no damage to key components and structures.

3.6 Further action may be required to demonstrate that reactor vessel integrity is maintained within ASME Code requirements such as indicated in the referenced ASME Code Case N-557 (**2**). Such action is beyond the scope of this guide.

4. General Considerations

4.1 Successful use of in-service annealing requires a thorough knowledge of the irradiation behavior of the specific reactor-vessel materials, their annealing response and reirradiation embrittlement trend, the vessel design, fabrication history, and operating history. Some of these items may not be available for specific older vessels, and documented engineering judgment may be required to conservatively estimate the missing information.

4.1.1 To ascertain the design operating ~~life knowledge~~ life, knowledge of the following items is needed: reactor vessel material composition, mechanical properties, fabrication techniques, nondestructive test results, anticipated stress levels in the vessel, neutron fluence, neutron energy spectrum, operating temperature, and power history.

4.1.1.1 The initial RT_{NDT} as specified in subarticle NB-2300 of the ASME Boiler and Pressure Vessel Code, Section III, should be determined or estimated for those materials of concern in the high fluence regions of the reactor pressure vessel. Alternative methods for the determination of RT_{NDT} also may be used. Consideration should be given to the technical justification for alternate methodologies and the data, which form the basis for the RT_{NDT} determination. Initial RT_{NDT} values should be available or estimated for all materials located in these areas.

4.1.1.2 The initial Charpy upper shelf energy as defined by Practices E185 and E2215 should be determined for materials of concern in the beltline region of the reactor pressure vessel. Initial upper shelf energy levels should be available or estimated for all materials located in this area.

4.1.1.3 Unirradiated archive heats of reactor vessel beltline materials[7] should be maintained for preparation of additional surveillance samples as required by Practices E185 and E2215. Previously tested specimens should be retained as an additional source of material.

4.1.1.4 A record of the actual fabrication history, including heat treatment and welding procedure, of the materials in the beltline region of the vessel should be maintained.

4.1.1.5 The chemical composition should be determined for base metal(s) and deposited weld metal(s) and should include all elements potentially relevant to irradiation, annealing, and reirradiation behavior, for example, copper, nickel, phosphorus, manganese and sulfur. The variability in chemical composition should be determined when possible.

4.1.2 The anticipated remaining operating lifetime of the reactor vessel without annealing should be established using neutron embrittlement projections for the reactor vessel materials.

4.1.2.1 A surveillance program conducted in accordance with the requirements of Practices E185 and E2215 will provide information from which to evaluate vessel condition. Attention should be given to assuring that variations in the fluence-rate, neutron energy spectrum, and irradiation temperature for all different reactor neutron environments utilized are taken into account.

4.1.2.2 Transition temperature and upper-shelf Charpy energy ~~level~~ data have been compiled and used to develop correlations of ΔRT_{NDT} and upper shelf drop versus fluence, for example, Guide E900 or NRC Regulatory Guide 1.99, Revision 2. These approaches, or other class-specific correlations, should be used to estimate ΔRT_{NDT} and upper shelf energy drop for the specific heats of materials in the vessel beltline.

4.1.2.3 The results of surveillance specimen tests required by Practice E2215 should be compared to the data developed ~~for~~in 4.1.2.2 to ascertain whether the materials are performing ~~in the manner~~ as expected. If not, an evaluation should be made to establish the extent of the remaining service life before restoration of properties is necessary.

4.1.3 Available data should be compiled for the annealing and post-anneal reirradiation responses of each class of material, and if available, for the specific heats of materials in the vessel. The bibliography **(3-78)** in Appendix X1 provides references for data compilation. Data collected should include transition temperature ~~shift~~shifts and upper shelf Charpy energy changes. Actual fracture toughness data also should be compiled, as well as other supplemental information or data such as instrumented Charpy, indentation/hardness, tensile, and other miniature specimen test results (see Practice E636 for additional testing that can be utilized in assessing annealing behavior). The extent of the increased service life after annealing should be estimated using the guidance provided in Appendix X2.

4.1.4 Irradiated material from the vessel surveillance program should be retained as a source of material for future vessel condition assessments.

[7] Consideration should be given to the possibility of thermal embrittlement of beltline materials, including ~~heat-affected-zone,~~heat-affected-zones, as a result of the annealing heat-treatment.

5. Annealing Method

5.1 The annealing method selected should consider the magnitude of the recovery needed to extend the lifetime, the predicted annealing response, the reirradiation response, the accessibility of the reactor vessel to allow inspection and temperature monitoring, the constraints resulting from the design of the reactor, and the structural relationship of the reactor vessel to the primary system and supports. A detailed annealing procedure should be prepared, for example, see ASME Code Case N-577(**2**) and NRC Regulatory Guide 1.162. This written procedure should include all quality assurance measures and training to be conducted to assure an effective annealing operation.

5.1.1 The annealing method employed must not degrade the original design of the system. The parameters for a dry anneal may exceed the original design limits of the reactor vessel. In this case, the primary coolant water has been removed and a heating device is employed to raise the vessel temperature locally in the affected beltline region above the original design temperature. ASME Code Case N-557 (**2**) provides a framework for assuring design conformance for an in-service thermal anneal in air. A lower temperature wet anneal, in which the heating medium is the primary coolant water, should not exceed the original design pressure and temperature for the reactor vessel.

5.1.2 A review of all reactor components likely to be impacted by the anneal should be completed prior to the initiation of the anneal.

5.1.3 Consideration should be given to the effects of mechanical and thermal stresses and temperature on all system components, structures, and control instrumentation. Specific material properties should be justified by the analyst evaluating these effects. Examples of such effects are as follows:

5.1.3.1 Changes in the properties of friction reducing materials in sliding or articulating connections.

5.1.3.2 Reduction in neutron and gamma absorption capacity of supplementary shielding materials.

5.1.3.3 Effect of thermal growth on closely machined articulated or sliding interfaces.

5.1.3.4 Changes in mechanical and thermal properties of the reactor vessel insulation.

5.1.3.5 Effect of elevated temperatures on low melting point alloys, if applicable.

5.1.4 A detailed thermal and stress evaluation should be performed to demonstrate that localized temperatures, thermal stresses, and subsequent residual stresses are acceptable. This evaluation will help to establish the heating system design and heat-up/cool-down rates for the anneal procedure.

5.1.4.1 Vessel distortion should be considered both analytically and physically. Measurement of dimensions prior to and after annealing should be considered to assess dimensional stability.

5.1.4.2 Adequate analytical estimation and actual measurement of concrete temperatures in the region near the reactor vessel are needed to avoid concrete degradation. The properties of the concrete should be known or estimated[8] in order to demonstrate that no damage will occur during the annealing.

5.1.5 The annealing method selected must assure adequate recovery of the reactor vessel materials. An experimental program may be undertaken prior to the in-service anneal to establish the degree of material properties recovery for the specific materials in the beltline of the vessel (see Appendix X2). This program shall use materials that are representative of reactor vessel materials in accordance with the criteria set forth in Practice E185 for material selection and irradiation conditions. For example, the program may use existing broken irradiated Charpy halves from the current surveillance program that have been reconstituted following Guide ~~E 1253~~E1253 ~~or accelerated irradiations with specimens prepared from the available archive materials as described in 4.1.1.3~~, ~~from other sources of representative material, or reconstituted specimens of~~ or samples taken from the actual pressure ~~vessel wall.~~ vessel. Other miniature or small specimen testing techniques also can be considered if properly validated. The program also may assess the adequacy of selected heat treatment conditions for achieving the minimum required recovery. The results from the experimental program should be compared with the data compiled for 4.1.3. Data generated relative to the actual vessel neutron exposure should be reviewed in relation to ~~temperature~~ temperature, fluence and fluence-rate effects.

5.1.6 The annealing procedure employed should provide for adequate instrumentation to control and monitor the temperature of the vessel such that a complete temperature record is available throughout all phases of the annealing operation.[9] Special consideration should be given to axial, azimuthal, and through-wall thermal gradients in the beltline region and any regions anticipated to experience high stresses during the anneal, such as the nozzles.

5.1.7 The annealing procedure should include a description of the annealing equipment, an outline of the operational requirements, and integration of pre-annealing test of the heating equipment. Consideration should be given to storage of the core, internals, and coolant.

5.1.8 Special precautions to assure the protection of plant personnel and the general public from any release of radioactive materials should be provided. The annealing operation also should give adequate consideration to the radiation exposure of personnel, as well as any radioactive waste processing, radioactive-material decontamination, and radioactive-waste shipment.

[8] ~~Following~~Follow American Concrete Institute guidelines as appropriate. Additional guidance may be available from U.S. annealing demonstration programs.

[9] U.S. annealing demonstrations provide further insight into the degree of instrumentation needed to adequately monitor and control the annealing operation.

5.2 The annealing process must be carefully monitored to assure that the conditions outlined in the annealing procedure described in 5.1 are maintained. The temperature of the reactor vessel must be monitored to assure that the annealing operating conditions are maintained and to demonstrate that temperature gradients are consistent with the thermal and stress ~~analysis.~~analyses.

6. Annealing Surveillance and Verification

6.1 The effectiveness of the anneal depends upon the degree of property recovery and the reembrittlement trend. The surveillance specimens, as described in Practice E185, provide a means of assessing the degree of properties recovered from an anneal.

6.1.1 Guidelines for assessing annealing recovery from available materials are given in Appendix X2. A surveillance program must be established after the anneal to monitor reirradiation embrittlement. Appendix X2 also contains guidelines for such a surveillance program.

6.1.2 If sufficient materials are not available or if conditions dictate that the approach in 6.1.1 ~~in~~is inapplicable, an alternative program for demonstrating the effectiveness of the in-service anneal and for monitoring the reirradiation response of the vessel materials should be established. Appendix X2 again contains guidelines that can be followed. The bibliography (**3-78**) ~~of information~~ given in Appendix X1 also will be valuable in establishing an alternative program.

7. Documentation

7.1 A description and analysis of the annealing procedures, results, and supporting data should be prepared, for example, see ASME Code Case N-557 (**2**) and NRC Regulatory Guide 1.162. This documentation should include, but not be limited to, the following information and data:

7.1.1 A description should be provided of all data and analyses used to support the justification for performing the anneal. This should include all irradiation analyses or test program results, as well as all special calculations, related stress analyses, and heating evaluations.

7.1.2 A description of all materials used in the establishment of the annealing process and the monitoring of the actual annealing operation should be included. This section should include the reporting requirements of Practices E185 and E2215.

7.1.3 A detailed description of the proposed annealing procedure and a chronology of the proposed versus actual procedure for the annealing operation should be documented. Special emphasis is to be given to the location of temperature monitors and their records.

7.1.4 A detailed evaluation of the results of the annealing operation with appropriate technical justification should be reported. Any limitations regarding material property recovery or future plant operation should be described and documented.

7.1.5 Applicable ASME codes, ASTM standards and guides, NRC regulations and guides, and other technical references should be ~~described.~~provided. All appropriate regulations and standards should be addressed as to the extent to which they were met.

7.1.6 Specific details of the planned new surveillance program for monitoring the reembrittlement trend for the beltline materials should be described.

8. Keywords

8.1 fracture toughness; irradiation; nuclear reactor vessels (light-water moderated); radiation exposure; surveillance (of nuclear reactor vessels)

APPENDIXES

(Nonmandatory Information)

X1. BIBLIOGRAPHY OF MATERIAL PROPERTIES FOR PRESSURE VESSEL STEELS

X1.1 References containing existing material property information for pressure vessel materials are listed to cover annealing response, changes in RT_{NDT} and upper shelf recovery, and reirradiation embrittlement. Limited fracture toughness data also are available. These data are to be used in assessing the anticipated annealing recovery and reembrittlement for similar pressure vessel steels. These same data may be used to determine a generic response when relevant materials are not available for actual recovery demonstration and surveillance.

X1.2 The reference bibliography (**3-78**) of annealing information is not intended to be totally inclusive. Major emphasis is given to U.S. commercial pressure vessel steels and welds, particularly those with high copper concentrations that may be critical in older operating plants. Studies before 1974 ~~involved steels~~(see Refs ~~**that**~~(**3-12**)~~ only are typical of a few commercial vessels in operation today (see Refs~~) involved steels that only are typical of a few commercial vessels in operation ~~(~~today.~~3-12~~)~~)~~.

X1.3 The work performed on annealing in the 1970s at the Naval Research Laboratory is summarized in Ref (**13**). For other sources of information during the ~~1970s (see~~1970s, see Refs (**14-18**).

X1.4 ~~The data~~ Data and evaluations reported ~~beginning~~ in the 1980s can be found starting with Ref (**19**). This compilation includes data for European and Russian steels, for example, see Refs (**20-47**).

X1.5 ~~The most~~ More recent studies for pressure vessel steels, primarily focused on the WWER-440 steels, are contained in Refs (**68-78**).

X2. GUIDANCE FOR VERIFYING RECOVERY AND RE-IRRADIATION EMBRITTLEMENT

X2.1 The key elements with respect to continued operation of a reactor vessel after annealing are the degree of recovery and the reembrittlement trend. Ideally, both of these elements should be measured using existing surveillance capsules containing the limiting reactor beltline materials. Older vessels, however, which may be the first candidates for annealing, may not have enough surveillance capsules, or the limiting material may not have been included in the surveillance program. Even if there are capsules that can be used to assess annealing and the subsequent reembrittlement, different lead factors may make future assessments difficult to directly quantify unless a reembrittlement trend curve can be estimated. The purpose of this appendix is to provide guidance for defining the post-anneal reference temperature (RT_{NDT}) and to estimate and measure the reembrittlement trends for reactor beltline materials. This guide is general since it is impractical to give specific quantitative directions due to the variety of materials, irradiation conditions, and other considerations such as future operating plans.

X2.2 Quantification of annealing recovery has been studied in detail, primarily in test reactor environments, while subsequent reembrittlement trends have less supporting data, and therefore, less definition. Upper shelf Charpy energy changes can be addressed in a similar manner as the RT_{NDT} approach presented in this appendix.

X2.3 The approach presented here is to provide guidance in developing an approximate annealing/reembrittlement trend curve from the existing surveillance irradiation data and several correlations that can be checked with other available capsule results, post-anneal, and used to project future trends. Test reactor irradiations with archive, or representative, materials may be used in special cases to check the trend curve methodology, but uncertainties due to temperature and fluence-rate effects should be considered.

X2.4 Since the data base of annealing recovery and reembrittlement trend does not cover all materials and annealing conditions, several assumptions have been made in developing a trend curve approach, and these assumptions should be kept in mind ~~in~~ using the methodology. Mechanistic modeling of the irradiation, annealing, and reirradiation processes for plant specific materials may provide useful guidance and help reduce uncertainties in using this methodology.

X2.5 A conservative methodology of post-anneal reirradiation trend curve development is schematically shown in Fig. X2.1. This

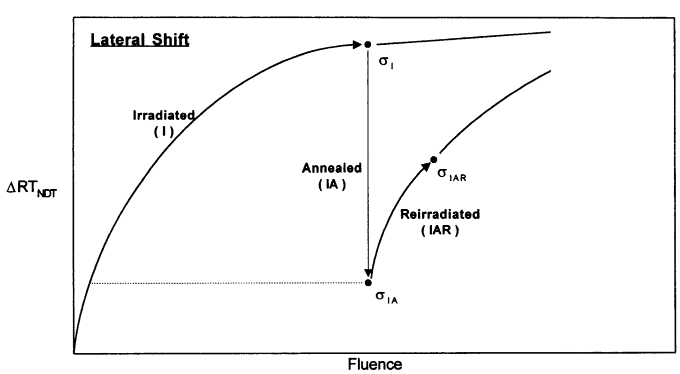

FIG. X2.1 Lateral Shift Method for Estimating Reirradiation Embrittlement