Designation: E 509 – 97

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

This standard is issued under the fixed designation E 509; 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-cooled nuclear reactor vessel and demonstrating the effectiveness of the procedure. The purpose of this in-service 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 heat treatment 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 Practice E 185. 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 SI units are to be regarded separately as 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:*

E 184 Practice for Effects of High-Energy Neutron Radiation on the Mechanical Properties of Metallic Materials E 706 (1B)[3]

E 185 Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels E 706 (IF)[3]

E 636 Practice for Conducting Supplemental Surveillance Tests for Nuclear Power Reactor Vessels E 706 (IH)[3]

E 900 Guide for Predicting Neutron Radiation Damage to Reactor Vessel Materials E 706 (IIF)[3]

E 1253 Guide for Reconstitution of Irradiated Charpy Specimens[3]

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

[1] This guide is under the jurisdiction of ASTM Committee E-10 on Nuclear Technology and Applications and is the direct responsibility of Subcommittee E10.02 on Behavior and Use of Metallic Materials in Nuclear Systems.

Current edition approved June 10, 1997. Published May 1998.

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

[3] *Annual Book of ASTM Standards*, Vol 12.02.

[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.

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 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 such equipment as may be required to do the in-service annealing; and, to train personnel to perform the annealing treatment.

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 similar) materials to obtain the essential post-anneal and reirradiation RT_{NDT}, upper shelf energy level, or fracture toughness, or a combination thereof.

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

---

[6] Consideration can be given to the reevaluation of broken Charpy specimens from capsules withdrawn earlier which can be reconstituted using Guide E 1253 or from material obtained (sampled) from the actual pressure-vessel wall.

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 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 Practice E 185 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 Practice E 185. 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, 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 Practice E 185 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 E 900 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 E 185 should be compared to the data developed for 4.1.2.2 to ascertain whether the materials are performing in the manner 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 **(3-67)** in Appendix X1 provides references for data compilation. Data collected should include transition temperature shift 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 E 636 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.

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 heat treatment 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.

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

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 annealing treatment to establish the degree of material properties recovery for the specific materials in the beltline of the vessel (see Appendix X2). This program may use existing broken irradiated Charpy halves from the current surveillance program reconstituted following Guide E 1253 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 samples taken from the actual pressure vessel wall. 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 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 both a description of the annealing equipment and an outline of the operational requirements. 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.

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.

6. Annealing Surveillance and Verification

6.1 The effectiveness of the annealing treatment depends upon the degree of property recovery and the reembrittlement trend. The surveillance specimens, as described in Practice E 185, provide a means of assessing the degree of properties recovered from an annealing treatment.

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 inapplicable, an alternative program for demonstrating the effectiveness of the in-service annealing heat treatment and for monitoring the reirradiation response of the vessel materials should be established. Appendix X2 again contains guidelines that can be followed. The bibliography **(3-67)** 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 Practice E 185 and the applicable sections of Practice E 184.

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.

[8] Following 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.

7.1.5 Applicable ASME codes, ASTM standards and guides, NRC regulations and guides, and other technical references should be described. 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 cooled); 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 **(3-67)** 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 that only are typical of a few commercial vessels in operation today (see Refs **(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 Refs **(14-18)**.

X1.4 The 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)**.

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 similar, 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 methodology is termed "lateral shift" since the initial irradiation trend curve merely is translated laterally to project reirradiation behavior.

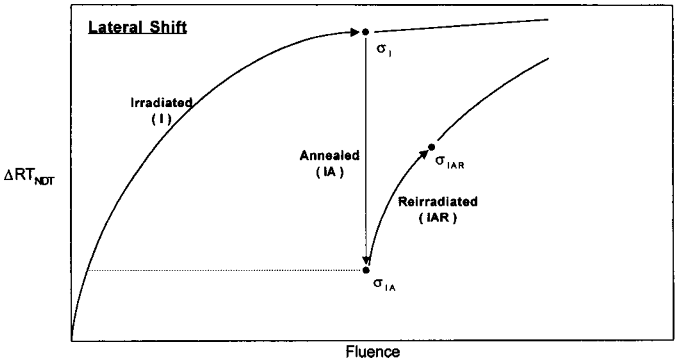

FIG. X2.1 Lateral Shift Method for Estimating Reirradiation Embrittlement

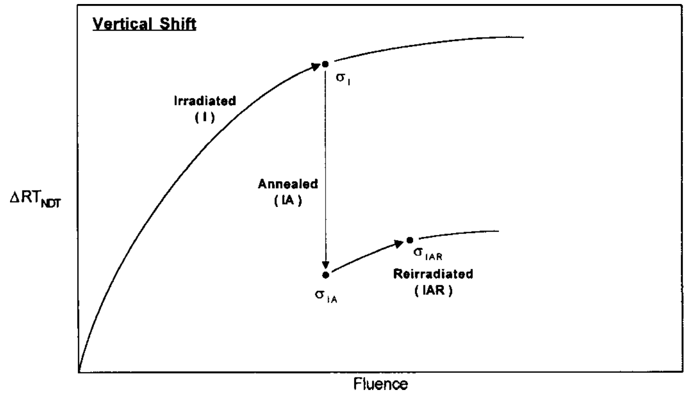

FIG. X2.2 Vertical Shift Method for Estimating Reirradiation Embrittlement

X2.5.1 The initial irradiation correlation must be established for the critical material(s). Suggested methods include using Guide E 900 or NRC Regulatory Guide 1.99, Revision 2. From these guides, a mean prediction curve for initial irradiation (*I*) damage is used with an approximate variance (σ_I^2). Existing surveillance data, or other appropriately justified data, can be used to adjust the mean curve, similar to the process allowed in NRC Regulatory Guide 1.99, Revision 2.

X2.5.2 The next step is the estimation of the annealing recovery for the irradiated-annealed (*IA*) condition. An approach, such as suggested in NRC Regulatory Guide 1.162 and documented in Ref **(49)**, may be used. This approach has been statistically analyzed, and a corresponding overall variance (σ_{IA}^2) is no greater than that from the original irradiation. The variance associated with the anneal, therefore, is encompassed in the irradiation variance: $\sigma_{IA}^2 = \sigma_I^2$. Certain limitations of this approach are acknowledged in Ref **(49)** relative to the range of applicable data and caution should be exercised when approaching these limiting conditions. The limited extent of data used to develop the predictive equations also should be considered.

X2.5.3 Next is a lateral shift of the initial irradiation embrittlement path to become the post-anneal reirradiation trend curve for the irradiated-annealed-reirradiated (*IAR*) condition. This step has some technical uncertainty, but appears to be a logical first approximation. A variance σ_{IAR}^2 (assumed equivalent to σ_I^2) for reembrittlement may be used to project a reirradiation trend curve and approximate statistical bound.

X2.5.4 The predicted trend curve should be checked by experimental results. This verification should be planned well before the actual annealing takes place. The suggested procedures that may be followed in evaluating the estimated trend curve are described in X2.7.

X2.6 The "vertical shift" trend curve approach is similar to that of the "lateral shift," except the portion of the initial irradiation trend, projected as reirradiation behavior, is translated down vertically as shown in Fig. X2.2. The use of this estimated trend curve should be justified with actual post-anneal reirradiation data since the vertical shift method predicts significantly lower changes in RT_{NDT} after thermal annealing. Limited data show that reembrittlement trends for anneals near 850°F for one week lie between the lateral and vertical shift approaches.

X2.7 The following procedures provide guidance for assessing recovery and reembrittlement prior to making the decision to anneal, as well as developing the post-vessel anneal surveillance program once the decision to anneal has been made. The new surveillance program will provide a check on the recovery and reembrittlement estimation methodology just described and provide actual data for making adjustments when appropriate.

X2.7.1 First, withdraw all, or nearly all, capsules from the reactor and follow the diagram in Fig. X2.3.[10] The entry point into the flow diagram is to answer the question in the top diamond-shaped box as to whether or not there is adequate material available to perform testing on the vessel materials or other representative materials.[11] If the answer is yes, then the following sequence of steps beginning in X2.7.2 should be followed. If the answer is no, the steps defined beginning in X2.7.3 should be followed. Note that the boxes in Fig. X2.3 are identified with the appropriate paragraph number as used throughout X2.7.

X2.7.2 *For Use When Adequate Vessel or Representative Material Exists*—The material can come from the existing surveillance program (tested or untested specimens,) samples taken from the vessel wall, archived material from the original vessel construction or the surveillance program, which can be irradiated, or from materials available from other sources that can be justified as representative.

X2.7.2.1 An evaluation of how much material is available should be made to determine the extent of testing that can be performed for *IA* and *IAR* condition assessment. If sufficient material is available to perform both pre-vessel and post-vessel annealing evaluations, proceed to X2.7.2.2. If there is very limited material available, then emphasis should be placed

---

[10] One capsule may be kept in place in case the decision to anneal is later reversed, or for contingency purposes.

[11] "Representative" materials should match the critical materials in the vessel with regard to ASTM specification, material heat, vintage, and chemistry (copper and nickel content) in that order for base materials and weld wire specification, material heat, weld flux, vintage, and chemistry (copper and nickel content) in that order for weld metals, to the extent practical. In some cases, representative materials also can be equated to bounding materials when shown that the expected embrittlement trends should be greater than the actual critical materials. Practice E 185 provides details on original surveillance program design which can yield guidance in developing a post-anneal surveillance program.