



Standard Guide for In-Service Annealing of Light-Water Moderated Nuclear Reactor Vessels¹

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1. Scope

1.1 This guide covers the general procedures 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 either SI units or inch-pound units are to be regarded separately as 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)
- 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

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.