



Designation: E 509 – 97

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

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 Specimens³

2.2 ASME Standards:

Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Power Plant Components⁴

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 Materials⁵

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.