



Designation: E185 – 21

Standard Practice for Design of Surveillance Programs for Light-Water Moderated Nuclear Power Reactor Vessels¹

This standard is issued under the fixed designation E185; 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 practice covers procedures for designing a surveillance program for monitoring the radiation-induced changes in the mechanical properties of ferritic materials in light-water moderated nuclear power reactor vessels. New advanced light-water small modular reactor designs with a nominal design output of 300 MWe or less have not been specifically considered in this practice. This practice includes the minimum requirements for the design of a surveillance program, selection of vessel material to be included, and the initial schedule for evaluation of materials.

1.2 This practice was developed for all light-water moderated nuclear power reactor vessels for which the predicted maximum fast neutron fluence ($E > 1$ MeV) exceeds 1×10^{21} neutrons/m² (1×10^{17} n/cm²) at the inside surface of the ferritic steel reactor vessel.

1.3 This practice does not provide specific procedures for monitoring the radiation induced changes in properties beyond the design life. Practice E2215 addresses changes to the withdrawal schedule during and beyond the design life.

1.4 The values stated in SI units are to be regarded as the standard. The values given in parentheses are for information only.

NOTE 1—The increased complexity of the requirements for a light-water moderated nuclear power reactor vessel surveillance program has necessitated the separation of the requirements into three related standards. Practice E185 describes the minimum requirements for design of a surveillance program. Practice E2215 describes the procedures for testing and evaluation of surveillance capsules removed from a reactor vessel. Guide E636 provides guidance for conducting additional mechanical tests. A summary of the many major revisions to Practice E185 since its original issuance is contained in Appendix X2.

NOTE 2—This practice applies only to the planning and design of surveillance programs for reactor vessels designed and built after the effective date of this practice. Previous versions of Practice E185 apply to earlier reactor vessels. See Appendix X2.

1.5 *This international standard was developed in accordance with internationally recognized principles on standardization established in the Decision on Principles for the Development of International Standards, Guides and Recommendations issued by the World Trade Organization Technical Barriers to Trade (TBT) Committee.*

2. Referenced Documents

2.1 ASTM Standards:²

- A370 Test Methods and Definitions for Mechanical Testing of Steel Products
- A751 Test Methods and Practices for Chemical Analysis of Steel Products
- E8/E8M Test Methods for Tension Testing of Metallic Materials
- E21 Test Methods for Elevated Temperature Tension Tests of Metallic Materials
- E23 Test Methods for Notched Bar Impact Testing of Metallic Materials
- E170 Terminology Relating to Radiation Measurements and Dosimetry
- E208 Test Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels
- E482 Guide for Application of Neutron Transport Methods for Reactor Vessel Surveillance
- E636 Guide for Conducting Supplemental Surveillance Tests for Nuclear Power Reactor Vessels
- E844 Guide for Sensor Set Design and Irradiation for Reactor Surveillance
- E853 Practice for Analysis and Interpretation of Light-Water Reactor Surveillance Neutron Exposure Results
- E900 Guide for Predicting Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials
- E1214 Guide for Use of Melt Wire Temperature Monitors for Reactor Vessel Surveillance
- E1253 Guide for Reconstitution of Irradiated Charpy-Sized Specimens

¹ This practice 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 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.

E1820 Test Method for Measurement of Fracture Toughness
E1921 Test Method for Determination of Reference Temperature, T_o , for Ferritic Steels in the Transition Range

E2215 Practice for Evaluation of Surveillance Capsules from Light-Water Moderated Nuclear Power Reactor Vessels

E2298 Test Method for Instrumented Impact Testing of Metallic Materials

E2956 Guide for Monitoring the Neutron Exposure of LWR Reactor Pressure Vessels

2.2 ASME Standards:³

Boiler and Pressure Vessel Code, Section III Subsection NB-2000

Boiler and Pressure Vessel Code, Section XI Nonmandatory Appendix A, Analysis of Flaws, and Nonmandatory Appendix G, Fracture Toughness Criteria for Protection Against Failure

3. Terminology

3.1 Definitions:

3.1.1 *base metal*—as-fabricated plate material or forging material other than a weld or its corresponding heat-affected-zone (HAZ).

3.1.2 *beltline*—the irradiated region of the reactor vessel (shell material including weld seams and plates or forgings) that directly surrounds the effective height of the active core. Note that materials in regions adjacent to the beltline may sustain sufficient neutron damage to warrant consideration in the selection of surveillance materials.

3.1.3 *Charpy transition temperature curve*—a graphic or a curve-fitted presentation, or both, of absorbed energy, lateral expansion, or fracture appearance as functions of test temperature, extending over a range including the lower shelf (5 % or less shear fracture appearance), transition region, and the upper shelf (95 % or greater shear fracture appearance).

3.1.4 *Charpy transition temperature shift*—the difference in the 41 J (30 ft-lbf) index temperatures for the best fit (average) Charpy absorbed energy curve measured before and after irradiation. Similar measures of temperature shift can be defined based on other indices in 3.1.3, but the current industry practice is to use 41 J (30 ft-lbf) and is consistent with Guide E900.

3.1.5 *Charpy upper-shelf energy level*—the average energy value for all Charpy specimen tests (preferably three or more) whose test temperature is at or above the Charpy upper-shelf onset; specimens tested at temperatures greater than 83°C (150°F) above the Charpy upper-shelf onset shall not be included, unless no data are available between the onset temperature and onset +83°C (+150°F).

3.1.6 *Charpy upper-shelf onset*—the temperature at which the fracture appearance of all Charpy specimens tested is at or above 95 % shear.

3.1.7 *heat-affected-zone (HAZ)*—plate material or forging material extending outward from, but not including, the weld fusion line in which the microstructure of the base metal has been altered by the heat of the welding process.

3.1.8 *index temperature*—the temperature corresponding to a predetermined level of absorbed energy, lateral expansion, or fracture appearance obtained from the best-fit (average) Charpy transition curve.

3.1.9 *lead factor*—the ratio of the average neutron fluence ($E > 1$ MeV) of the specimens in a surveillance capsule to the peak neutron fluence ($E > 1$ MeV) of the corresponding material at the ferritic steel reactor pressure vessel inside surface calculated over the same time period.

3.1.9.1 *Discussion*—Changes in the reactor operating parameters or fuel management may cause the lead factor to change.

3.1.10 *limiting materials*—typically the weld and base material with the highest predicted transition temperature using the projected fluence at the end of design life of each material determined by adding the appropriate transition temperature shift to the unirradiated RT_{NDT} . Materials that are projected to most closely approach a regulatory limit at the end of the design life should be considered in selecting the limiting material. Guide E900 describes a method for predicting the Transition Temperature Shift (TTS). Regulators or other sources may describe different methods for predicting TTS. The basis for selecting the limiting weld and base materials shall be documented.

3.1.11 *maximum design fluence (MDF)*—the maximum projected fluence at the inside surface of the ferritic pressure vessel at the end of design life (if clad, MDF is defined at the interface of the cladding to the ferritic steel). Changes during operation will affect the projected fluence and are addressed in Practice E2215.

3.1.12 *reference material*—any steel that has been characterized as to the sensitivity of its tensile, impact and fracture toughness properties to neutron radiation-induced embrittlement.

3.1.13 *reference temperature (RT_{NDT})*—see subarticle NB-2300 of the ASME Boiler and Pressure Vessel Code, Section III, “Nuclear Power Plant Components” for the definition of RT_{NDT} for unirradiated material based on Charpy (Test Method A370) and drop weight tests (Test Method E208). ASME Code Section XI, Appendices A and G provide an alternative definition for the reference temperature (RT_{To}) based on fracture toughness properties (Test Method E1921)

3.1.14 *standby capsule*—a surveillance capsule meeting the recommendations of this practice that is in the reactor vessel irradiation location as defined by Practice E185, but the testing of which is not required by this practice.

3.2 Neutron Exposure Terminology:

3.2.1 Definitions of terms related to neutron dosimetry and exposure are provided in Terminology E170.

4. Significance and Use

4.1 Predictions of neutron radiation effects on pressure vessel steels are considered in the design of light-water

³ Available from the American Society of Mechanical Engineers, Third Park Avenue, New York, NY 10016.