Designation: E2215 – 15

Standard Practice for Evaluation of Surveillance Capsules from Light-Water Moderated Nuclear Power Reactor Vessels[1]

This standard is issued under the fixed designation E2215; 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 the evaluation of test specimens and dosimetry from light water moderated nuclear power reactor pressure vessel surveillance capsules.

1.2 Additionally, this practice provides guidance on reassessing withdrawal schedule for design life and operation beyond design life.

1.3 This practice is one of a series of standard practices that outline the surveillance program required for nuclear reactor pressure vessels. The surveillance program monitors the irradiation-induced changes in the ferritic steels that comprise the beltline of a light-water moderated nuclear reactor pressure vessel.

1.4 This practice along with its companion surveillance program practice, Practice E185, is intended for application in monitoring the properties of beltline materials in any light-water moderated nuclear reactor.[2]

1.5 Modifications to the standard test program and supplemental tests are described in Guide E636.

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

2. Referenced Documents

2.1 *ASTM Standards:*[3]

A370 Test Methods and Definitions for Mechanical Testing 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
E185 Practice for Design of Surveillance Programs for Light-Water Moderated Nuclear Power Reactor Vessels
E208 Test Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels
E509 Guide for In-Service Annealing of Light-Water Moderated Nuclear Reactor Vessels
E636 Guide for Conducting Supplemental Surveillance Tests for Nuclear Power Reactor Vessels, E 706 (IH)
E693 Practice for Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacements Per Atom (DPA), E 706(ID)
E844 Guide for Sensor Set Design and Irradiation for Reactor Surveillance, E 706 (IIC)
E853 Practice for Analysis and Interpretation of Light-Water Reactor Surveillance 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, E 706 (IIIE)
E1253 Guide for Reconstitution of Irradiated Charpy-Sized Specimens
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

2.2 *ASME Standards:*[4]

Boiler and Pressure Vessel Code, Section III Subarticle NB-2000, Rules for Construction of Nuclear Facility Components, Class 1 Components, Materials
Boiler and Pressure Vessel Code, Section XI Nonmandatory Appendix A, Analysis of Flaws, and Nonmandatory Appendix G, Fracture Toughness Criteria for Protection against Failure

[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] Prior to the adoption of these standard practices, surveillance capsule testing requirements were only contained in Practice E185.

[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.

[4] Available from American Society of Mechanical Engineers, Third Park Avenue, New York, NY 10016.

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 suficient neutron damage to warrant consideration in the selection of surveillance materials.

3.1.3 *Charpy transition temperature curve*—a graphic or curve-fitted presentation, or both, of absorbed energy, lateral expansion, or fracture appearance as a function 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 U.S. 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 *end-of-license (EOL) fluence*—the maximum predicted fluence at the inside surface of the ferritic pressure vessel (if clad, the interface of the cladding to the ferritic steel) corresponding to the end of the operating license period.

3.1.8 *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.

3.1.9 *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.

3.1.10 *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.

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

3.1.11 *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 (TTS) to the unirradiated RT_{NDT}. The TTS can be determined from the relationship found in Guide E900 or other sources, including regulations.

3.1.12 *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).

3.1.13 *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 and is included in the Practice E185 surveillance program.

3.1.14 *reference temperature (RT_{NDT})* —see subarticle NB-2300 of the ASME Boiler and Pressure Vessel Code, Section III, for the definition of RT_{NDT} for unirradiated material based on Charpy (Test Methods 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).

3.1.15 *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 Neutron radiation effects are considered in the design of light-water moderated nuclear power reactors. Changes in system operating parameters may be made throughout the service life of the reactor to account for these effects. A surveillance program is used to measure changes in the properties of actual vessel materials due to the irradiation environment. This practice describes the criteria that should be considered in evaluating surveillance program test capsules.

4.2 Prior to the first issue date of this standard, the design of surveillance programs and the testing of surveillance capsules were both covered in a single standard, Practice E185. Between its provisional adoption in 1961 and its replacement linked to this standard, Practice E185 was revised many times (1966, 1970, 1973, 1979, 1982, 1993 and 1998). Therefore, capsules from surveillance programs that were designed and implemented under early versions of the standard were often tested after substantial changes to the standard had been adopted. For clarity, the standard practice for surveillance programs has been divided into the new Practice E185 that covers the design of new surveillance programs and this standard practice that covers the testing and evaluation of surveillance capsules. Modifications to the standard test program and supplemental tests are described in Guide E636.

4.3 This practice is intended to cover testing and evaluation of all light-water moderated reactor pressure vessel surveillance capsules. The practice is applicable to testing of capsules from surveillance programs designed and implemented under all previous versions of Practice E185.

4.4 The radiation-induced changes in the properties of the reactor pressure vessel are generally monitored by measuring the index temperatures, the upper-shelf energy and the tensile properties of specimens from the surveillance program capsules. The significance of these radiation-induced changes is described in Practice E185.

4.5 Alternative methods exist for testing surveillance capsule materials. Some supplemental and alternative testing methods are available as indicated in Guide E636. Direct measurement of the fracture toughness is also feasible using the T_o Reference Temperature method defined in Test Method E1921 or *J*-integral techniques defined in Test Method E1820. Additionally, hardness testing can be used to supplement standard methods as a means of monitoring the irradiation response of the materials.

4.6 The methodology to be used in the analysis and interpretation of neutron dosimetry data and the determination of neutron fluence is defined in Practice E853.

4.7 Guide E900 describes the bases used to evaluate the radiation-induced changes in Charpy transition temperature for reactor vessel materials and provides a methodology for predicting future values.

4.8 Guide E509 provides direction for development of a procedure for conducting an in-service thermal anneal of a light-water cooled nuclear reactor vessel and demonstrating the effectiveness of the procedure including a post-annealing vessel radiation surveillance program.

5. Determination of Capsule Condition

5.1 *Visual Examination*—A complete visual exam of the capsule condition should be completed upon receipt and during disassembly at the testing laboratory. External identification marks on the capsule shall be verified. Signs of damage or degradation of the capsule exterior shall be recorded.

5.2 *Capsule Content*—The specimen loading pattern should be compared to the capsule fabrication records and any deviations shall be noted. Any evidence of corrosion or other damage to the specimens shall also be noted. The condition of any temperature monitors shall be noted and recorded.

5.3 *Irradiation Temperature History*—The average capsule temperature during full power operation shall be estimated for each reactor fuel cycle experienced by the capsule. The local reactor coolant temperature may be used as a reasonable approximation, although gamma-ray heating should be considered if it leads to a significant temperature difference. In a typical pressurized water reactor, the coolant inlet temperature may be used as an estimate of the capsule irradiation temperature using a time-weighted average (see Guide E900). In a typical boiling water reactor, the recirculation temperature may be used as an estimate of the capsule irradiation temperature.

5.4 *Peak Temperature*—Temperature monitors shall be examined and any evidence of melting shall be recorded in accordance with Guide E1214.

6. Measurement of Irradiation Exposure

6.1 The monthly power history of the reactor for all cycles prior to capsule removal shall be recorded. Other data that are needed on a fuel-cycle-specific basis include: assembly-wise core power distributions, including enrichments and burnups, axial core power distributions, axial core void distributions (BWRs only), and core and downcomer water temperatures. Other key changes that need to be recorded include the addition or removal of flux suppression rods or shield rods, uprates or derates of reactor power, and other reactor modifications such as adding neutron shielding or the removal or addition of structures such as a thermal shield. Fuel assembly, reactor internals, and reactor pressure vessel dimensional information also need to be recorded. Surveillance capsule locations and movements: including storage periods outside the reactor, shall be provided for the evaluation of irradiation exposure.

6.2 The neutron fluence rate, neutron energy spectrum and neutron fluence of the surveillance specimens and the corresponding maximum values for the reactor vessel shall be determined in accordance with Practices E853.

6.3 Neutron fluence rate and fluence values ($E > 1$ MeV) and dpa rate and dpa values per Practice E693 shall be determined and recorded using a calculated spectrum adjusted or validated by dosimetry measurements.

7. Measurement of Mechanical Properties

7.1 Generally, all the materials contained in the capsule except the HAZ specimens (if included) should be tested. Testing of the HAZ specimens is optional.[5]

7.2 *Tension Tests:*

7.2.1 *Method*—Tension testing shall be conducted in accordance with Test Methods E8/E8M and E21.

7.2.2 *Test Temperature*—In general, the test temperatures for each material shall include room temperature, service temperature, and, if a specimen is available, one intermediate temperature to define the strength versus temperature relationship. Specific consideration should be given to the specific temperatures at which unirradiated specimens have been tested.

7.2.3 *Measurements*—Determine yield strength, tensile strength, total and uniform elongation and reduction of area.

7.3 *Charpy Tests:*

7.3.1 *Method*—Charpy tests shall be conducted in accordance with Test Methods and Definitions A370 and Test Method E23. Instrumented tests are recommended and should be performed in accordance with Guide E636. Broken Charpy specimens may be reconstituted for supplemental testing in accordance with Guide E1253.

7.3.2 *Test Temperature*—Specimens for each material shall be tested at temperatures selected to define the full Charpy energy transition curve. Particular emphasis should be placed

[5] Troyer, Greg and Erickson, Marjorie, "Empirical Analyses of Effects of the Heat Affected Zone and Post Weld Heat Treatment on Irradiation Embrittlement of Reactor Pressure Vessel Steel," Effects of Radiation on Nuclear Materials: 26th Volume, STP 1572, Mark Kirk and Enrico Lucon, Eds., ASTM International, West Conshohocken, PA, 2014, pp. 155-170.