



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 X1.

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 X1.

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.

Current edition approved Dec. 1, 2016. Published December 2016. Originally approved in 1961 as E185 – 61 T. Last previous edition approved in 2015 as E185 – 15 ϵ 1. DOI: 10.1520/E0185-16.

2. Referenced Documents

2.1 ASTM Standards:²

- A370 Test Methods and Definitions for Mechanical Testing of Steel Products
- A751 Test Methods, Practices, and Terminology 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, E 706 (IH)
- 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
- E2215 Practice for Evaluation of Surveillance Capsules from Light-Water Moderated Nuclear Power Reactor Vessels

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.

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

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 moderated nuclear power reactors. Changes in system operating parameters often are made throughout the service life of the reactor vessel to account for radiation effects. Due to the variability in the behavior of reactor vessel steels, a surveillance program is warranted to monitor changes in the properties of actual vessel materials caused by long-term exposure to the neutron radiation and temperature environment of the reactor vessel. This practice describes the criteria that should

be considered in planning and implementing surveillance test programs and points out precautions that should be taken to ensure that: (1) capsule exposures can be related to beltline exposures, (2) materials selected for the surveillance program are samples of those materials most likely to limit the operation of the reactor vessel, and (3) the test specimen types are appropriate for the evaluation of radiation effects on the reactor vessel.

4.2 Guides **E482** and **E853** describe a methodology for estimation of neutron exposure obtained for reactor vessel surveillance programs. Regulators or other sources may describe different methods.

4.3 The design of a surveillance program for a given reactor vessel must consider the existing body of data on similar materials in addition to the specific materials used for that reactor vessel. The amount of such data and the similarity of exposure conditions and material characteristics will determine their applicability for predicting radiation effects.

5. Surveillance Program Design

5.1 This section describes the minimum requirements for the design of a surveillance program for monitoring the radiation-induced changes in the mechanical properties of the ferritic materials that compose the reactor vessel.

5.2 Surveillance Materials:

5.2.1 *Materials Selection*—The surveillance materials shall include, at minimum, the limiting base metal and the limiting weld. If a limiting material is outside the beltline, the limiting beltline base and weld materials shall also be included. If there is no beltline weld, capsules whose target fluence (**Table 1**) is greater than two times the design fluence of the limiting weld are not required to contain weld metal, except that the first capsule must contain the limiting weld material.

NOTE 3—The predicted limiting material may change during operation due to changes that may occur in the transition temperature shift prediction formulation, or other factors. Therefore, it is prudent to include additional potentially limiting materials in the surveillance program as capsule space permits.

5.2.2 *Material Sampling*—A minimum surveillance program shall consist of the material selected in **5.2.1**, taken from the following: (1) base metal from the actual plate(s) or forging(s) used in the reactor vessel, and (2) weld metal(s) made with the same heat of weld wire and lot of flux and by the same welding procedure as that used for the reactor vessel welds. The base metals used to form the weld shall be from the reactor vessel. If a reactor vessel weld is contained in the beltline, at least one of the base metals used to fabricate the weldment(s) shall be a base metal beltline material included in

the surveillance program. Surveillance test specimens shall be removed from full reactor vessel thickness samples.

5.2.3 *Fabrication History*—The fabrication history (austenitizing, quench and tempering, and post-weld heat treatment) of the surveillance materials shall be fully representative of the fabrication history of the reactor vessel materials selected in **5.2.1** and shall be recorded.

5.2.4 *Chemical Analysis Requirements*—The chemical analysis required by the appropriate product specifications for the surveillance materials (base metal and as-deposited weld metal) shall be recorded and shall include copper (Cu), nickel (Ni), manganese (Mn), phosphorus (P), sulfur (S), silicon (Si), carbon (C), and vanadium (V), as well as all other alloying and residual elements commonly analyzed for in low-alloy steel products. The product analysis shall be as described in Test Method **A751** and verified by analyzing samples selected from the base metal and the as-deposited weld metal used for the surveillance program.

5.2.5 *Archive Materials*—Enough material to fill a minimum of three additional capsules per **5.4.2** beyond the minimum number required for the program as defined in **5.8.1** shall be retained with full documentation and identification. This archive should be in the form of full-thickness sections of the original materials (plates or forgings, and welds), because the preferred type and size of test specimens may change in the intervening years. If there is a weld in the beltline, it is recommended that the beltline base metal HAZ material associated with the archive weld material be retained should supplemental data be required. If the designer includes more than one standby capsule in the program, the specimens included in these capsules count toward the archive. However, material sufficient to fill one capsule should be retained as full-thickness sections.

NOTE 4—Experience has shown that it is no longer necessary to include the HAZ material in the surveillance program. However, it is recommended that the HAZ material be included with the archive material.⁴

5.3 Test Specimens

5.3.1 *Type of Specimens*—Charpy V-notch specimens corresponding to the Type A specimen described in Test Methods **A370** and **E23** shall be used. Tension specimens of the type, size, and shape described in Test Methods **A370** and **E8/E8M** are recommended. The gage section of irradiated and unirradiated tension specimens shall be of the same size and shape. Fracture toughness test specimens shall be consistent with the guidelines provided in Test Methods **E1820** and **E1921** and the selected type and size shall be the same for the irradiated and unirradiated condition.

5.3.2 *Specimen Orientation and Location*—Tension, Charpy and fracture toughness specimens representing the base metal (if the included weld was quenched and tempered, this is applicable to weld metal as well) shall be removed from about the quarter-thickness ($1/4$ -T or $3/4$ -T) locations with the mid-length of the specimens at least one thickness (1-T) from any

TABLE 1 Recommended Withdrawal Schedule

Sequence	Target Fluence	Notes
First	1/4 MDF	Testing Required
Second	1/2 MDF	Testing Required
Third	3/4 MDF	Testing Required
Fourth	MDF	Testing Required
Standby	< 2 MDF	Testing Not Required

⁴ Troyer, G., and Erickson, M., “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.

second heat treated surface. The base metal specimens for baseline testing and capsule irradiation should be removed from the same location ($1/4$ -T or $3/4$ -T) for each specimen type and in as close proximity as reasonable to reduce the effect of material variability on radiation-induced mechanical property change measurements. Material from the mid-thickness of the base metal shall not be used for test specimens. Specimens representing weld metal may be removed from any location throughout the thickness with the exception of locations within 13 mm ($1/2$ in.) of the root or surfaces of the welds. Specimens should be centered about the center line of the weld as shown in Fig. 1. Special attention must be given to defining the root of the weld or other material variability (for example base metal dilution) in order to avoid taking weld metal that is different in composition from the surveillance weld metal. The tension and Charpy specimens from base metal shall be oriented so that the major axis of the specimen is parallel to the surface and normal to the principal rolling direction for plates, or normal to the major working direction for forgings as shown in Test Method E23, Annex A5 (T-L orientation). The axis of the notch of the Charpy specimen for base metal and weld metal shall be oriented perpendicular to the surface of the material (expected direction of crack propagation is the principal working direction). The recommended orientation of the weld metal specimens is shown in Fig. 1. Weld metal tension specimens may be oriented in the same direction as the Charpy specimens provided that the reduced section consists entirely of weld metal. The fracture toughness specimens shall have the same orientation as the Charpy specimens.

5.4 Number of Specimens

5.4.1 Unirradiated Baseline Specimens—A minimum of 15 Charpy specimens shall be tested to establish full Charpy transition temperature curves for each material per Test Method E23. Instrumented tests are recommended and should be performed in accordance with Test Method E2298. It is recommended that upper-shelf Charpy tests be conducted at multiple temperatures using at least three specimens tested between the upper-shelf onset and onset +83°C (+150°F). At least six tension test specimens shall be tested to establish the

unirradiated tensile properties for both the base metal and the weld metal. A minimum of two specimens at room temperature (per Test Method E8/E8M) and two specimens at reactor vessel beltline operating temperature (per Test Method E21) should be tested. The remainder of the tensile specimens may be tested at intermediate temperatures as needed to define the effects of temperature on the tensile properties. It is recommended that a minimum of eight fracture toughness specimens be tested to establish the reference temperature, T_0 , per Test Method E1921 for the limiting material. Optionally, fracture toughness tests can be performed to establish the upper-shelf toughness following Test Method E1820.

5.4.2 Irradiated Specimens—The minimum number of test specimens for each irradiation exposure set (capsule) shall be as follows with exception noted in 5.2.1 regarding RPVs with no weld in the beltline:

Material	Charpy	Tension	Fracture Toughness
Each Base Metal	15	3	8 ^A
Each Weld Metal	15	3	8 ^A

^A Only fracture toughness specimens from the limiting material are required; the inclusion of the other material is recommended. It is suggested that a greater quantity of specimens be included in the irradiation capsules whenever possible.

5.5 Irradiation Requirements:

5.5.1 Encapsulation of Specimens—Specimens should be maintained in an inert environment within a corrosion-resistant capsule to prevent deterioration of the surface of the specimens during radiation exposure. Care should be exercised in the design of the capsule to ensure that the temperature history of the specimens matches, as closely as possible, the temperature experienced by the reactor vessel. Surveillance capsules shall be designed to prevent mechanical damage to the specimens and monitors during irradiation. The design of the capsule and capsule attachments shall also permit insertion of replacement capsules into the reactor vessel if required at a later time in the lifetime of the vessel. The design of the capsule holder and the means of attachment shall (1) preclude structural material degradation at the attachment, (2) avoid interference with

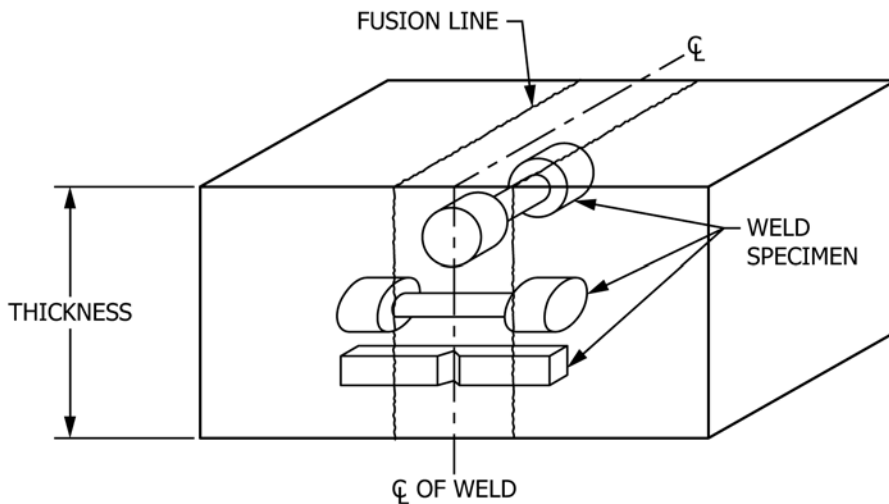


FIG. 1 Location of Test Specimens Within Weld Material

in-service inspection required by ASME Code Section XI, and (3) ensure the functionality of the capsule holder during the service life.

5.5.2 Location of Capsules:

5.5.2.1 Vessel Capsules (Required)—Surveillance capsules shall be located within the reactor vessel so that the specimen irradiation history duplicates as closely as possible, within the physical constraints of the system, the neutron spectrum, temperature history, and maximum neutron fluence experienced by the reactor vessel. The beltline material lead factor(s) should be greater than 1.5 and less than five. A lead factor near 1.5 will provide data that will closely duplicate the fluence of the vessel material and will enable monitoring through most of the operating lifetime. Capsules with higher lead factors will provide data earlier in plant life but will limit dosimetry monitoring capability for the later portion of the operating lifetime because it may be necessary to remove all the remaining capsules before the end of the design life due to the high fluence accumulation rate such that twice MDF is not exceeded. This range of lead factors has been selected to minimize the calculational uncertainties in extrapolating the surveillance measurements from the specimens to the reactor vessel wall and to optimize the ability of the program to monitor material property changes throughout the life of the reactor vessel, while keeping potential flux effects to a minimum. During the service life of the reactor vessel the lead factors for individual capsules may change as a result of changes in the reactor operating parameters or fuel management.

5.5.2.2 Accelerated Irradiation Capsules (Optional)—The design of some reactor vessel or core internals may not allow the positioning of all surveillance capsules with the recommended lead factors. Additional capsules may be positioned with higher lead factors than those described in **5.5.2.1** for accelerated irradiation. Plants with lead factors greater than five should provide a method of verifying the validity of the accelerated irradiation data. One method by which this verification may be accomplished is the inclusion of a reference material (see **5.6**).

5.5.3 Neutron Dosimeters:

5.5.3.1 Selection of Neutron Dosimeters—Neutron dosimeters for the surveillance capsules shall be selected according to Guide **E844**. The group of dosimeters selected shall be capable of providing information about fast neutron fluence, fluence rate, and spectrum; and thermal neutron fluence and fluence rate information; and displacements per atom (dpa) and dpa rate in iron.

5.5.3.2 Location of Neutron Dosimeters—Dosimeters shall be located within each surveillance capsule (see **5.5.2.1**) and each accelerated capsule (see **5.5.2.2**) if used.

5.5.3.3 Dosimetry measurement(s) may be advisable before the withdrawal of the first or between subsequent specimen capsules or after withdrawal of all capsules. If the capsule withdrawal schedule leads to long periods of operation without any dosimetry measurements, separate dosimeter(s) should be used to monitor radiation conditions independent of the sur-

veillance capsules. More information on monitoring the neutron exposure of the reactor pressure vessel may be found in Guide **E2956**.

NOTE 5—Generally, the preferred location for additional dosimetry is the air gap between the reactor pressure vessel reflective insulation and the biological shield surrounding the reactor. Dosimetry in this location can monitor the neutron exposure of the reactor vessel; both axially and azimuthally. In addition, dosimetry with various azimuthal locations can monitor changes in the core azimuthally, whereas the surveillance capsule dosimeters cannot detect core changes away from the surveillance capsule location. Operationally, dosimetry in this location is more easily removed and replaced than dosimetry located within the vessel.

5.6 Reference Materials:

5.6.1 Use of Reference Materials—The use of a reference material is optional. A reference material can provide an indication of possible deviations from the expected surveillance capsule irradiation conditions (for example, temperature and neutron fluence).

5.6.2 Selection of Reference Materials—If selected, the irradiation response of the reference material should be well characterized. The dependence of the specified mechanical property change (for example, transition temperature shift or change in yield strength) on irradiation temperature and neutron fluence should be documented to permit a useful evaluation of the irradiation conditions. In addition, reference material uniformity can affect the evaluation of reference material data; therefore, it is recommended that reference material uniformity be considered when a reference material is chosen. The selected reference material should have a measurable property change at surveillance capsule exposure conditions. The usage of reference materials has been documented.⁵ Note that historically there have been some limitations reported in the uniformity of the mechanical properties which have affected the results obtained.

5.7 Temperature Monitoring:

5.7.1 Differences between specimen irradiation temperature and design temperature, occurring as a result of capsule design features, variation in reactor coolant temperature, or both, can affect the extent of radiation induced property changes in the surveillance materials. As a minimum, the temperature of the reactor coolant shall be monitored and recorded. As an option, monitors may be included in the capsule. Since it is not practical to instrument the surveillance capsules, low melting point elements or eutectic alloys can be used as monitors to detect significant variations in exposure temperature. Monitors selected in accordance with Guide **E1214** should be used in surveillance programs to provide evidence of the maximum exposure temperature of the specimens.

5.7.2 Instructions for use of temperature monitors are provided in Guide **E1214**.

5.8 Number of Surveillance Capsules and Withdrawal Schedule:

5.8.1 Number of Capsules—A sufficient number of surveillance capsules shall be provided to monitor the effects of neutron irradiation on the reactor vessel throughout its design lifetime. At a minimum, the program shall include enough

⁵ See for example: ASTM DS54, July, 1974; NUREG/CR-4947 on HSST plates; and IAEA-TECDOC-1230, July, 2001, on the JRQ plate.

capsules to meet the testing requirements outlined in 5.8.2, plus one standby capsule (a total of five capsules). Additional capsules may be desired to monitor the effect of a major core change (which will change the fluence at the end of design life) or annealing of the vessel, or to provide supplemental toughness data for evaluating a flaw in the beltline or for monitoring vessel operation beyond design conditions. The archive material required in 5.2.5 is to be used for the additional capsules.

5.8.2 *Withdrawal Schedule*—The capsule withdrawal schedule permits periodic monitoring of long-time irradiation effects. The recommended withdrawal schedule is provided in Table 1.

5.8.2.1 The first step in the implementation of Table 1 is the estimation of the maximum design fluence at the ferritic steel reactor vessel inside surface (identified as MDF).

5.8.2.2 All plants that are required to have a surveillance program per 1.2 shall withdraw and test capsules at ¼ MDF, ½ MDF, ¾ MDF and MDF.

5.8.2.3 Capsules not required by the surveillance withdrawal schedule may be used to provide supplemental data. Supplemental testing may be required to clarify unexpected surveillance results, for operation beyond original design conditions or reactor vessel annealing programs. Supplemental testing may also be based on reconstitution of previously tested specimens following Guide E1253.

5.8.2.4 A minimum of three capsules shall be withdrawn during the first 40 years of reactor operation.

5.8.3 The schedule for capsule withdrawals is approximate and may be adjusted to coincide with a planned refueling outage.

5.8.4 The schedule for capsule withdrawals should be periodically reviewed to adjust for changing lead factors or changing reactor exposure rates resulting from changes in the reactor operating parameters or fuel management. Guidance for reevaluation of the withdrawal schedule is provided in Practice E2215.

6. Capsule Evaluations

6.1 *Baseline Data*—It is recommended that the baseline or unirradiated materials data be obtained during the design of the surveillance program and reported with the program descrip-

tion. Alternatively, the specimens may be placed in secured storage for testing at the time the first surveillance capsule is withdrawn for evaluation.

6.2 *Capsule Evaluation*—Upon removal from the reactor, the surveillance capsule shall be evaluated in accordance with Practice E2215. In addition, supplemental information can be obtained using Guide E636.

7. Report

7.1 The following information shall be provided. Where applicable, both SI units and English units shall be reported.

7.2 *Surveillance Program Description*—Description of the reactor vessel including the following:

7.2.1 Location of the surveillance capsules with respect to the reactor vessel, reactor vessel internals, and the reactor core.

7.2.2 Location in the vessel of the beltline and adjacent plates or forgings and the welds.

7.2.3 Location(s) of the peak vessel fluence.

7.2.4 Lead factor for each capsule.

7.2.5 *Surveillance Material Selection:*

7.2.5.1 Description of all beltline and adjacent materials including chemical analysis, fabrication history, Charpy data, tensile data, drop-weight data and initial RT_{NDT} for the as-fabricated condition.

7.2.5.2 Description of all archive materials including chemical analysis, fabrication history, Charpy data, tensile data, drop-weight data, initial RT_{NDT} , identification markings, and storage location for the as-fabricated condition.

7.3 *Surveillance Material Characterization:*

7.3.1 Description of the surveillance material including fabrication history, material source (heat or lot), and any differences between the surveillance material history and that of the reactor vessel material.

7.3.2 Location and orientation of the test specimens.

7.3.3 *Test Specimen and Environmental Sensor Design:*

7.3.3.1 Description of the test specimens (tension, Charpy, fracture toughness, and any other types of specimens used), neutron dosimeters, and temperature monitors.

7.4 Results of baseline (unirradiated) tests.

APPENDIX

(Nonmandatory Information)

X1. SUMMARY OF MAJOR REVISIONS TO PRACTICE E185 SINCE ITS ORIGINAL ISSUANCE

X1.1 Practice E185 has had many changes since its original release in 1961. Table X1.1 lists some of the most significant differences between the various revisions. Most of the nuclear power plant surveillance programs were developed during the two decades between 1960 and 1980. Since the plants have different program designs, it is sometimes important to understand the differences relative to later knowledge of how

surveillance programs should be designed and capsule materials tested and evaluated. Besides the changes shown in Table X1.1, other differences are described in the commentary below.

X1.2 The original version of this standard was issued as Practice E185-61T and was titled “Tentative Recommended Practice for Surveillance Test on Structural Materials in

TABLE X1.1 Significant Differences Between ASTM E185 Revisions

ASTM E185 Revision	Materials Monitored	No. of Capsules	No. of Unirradiated Specimens	No. of Irradiated Specimens per Exposure Set	Charpy Specimen Orientation	Withdrawal Schedule
1961	Materials used in fabrication	Not specified	Not specified	Charpy – minimum of 8 Tension – not specified	Not specified	Not specified
1966	Base metal with highest init. trans. temp. weld and HAZ	3 or more	Charpy – at least 15 Tension – at least 3	Charpy – 8 or more Tension – 2 or more	Parallel to working direction Notch \perp to surface	3 or more different times
1970	Base metal with highest init. trans. temp and flux, weld and HAZ with highest flux	Same as above	Same as above	Charpy – ≥ 8 Tension – ≥ 2 base and weld	Parallel to working direction Notch \perp to surface Illustrations provided	1 at 30 % of design life, 1 at EOL, 1 standby
1973	Beltline base, weld, and HAZ metal with highest init. trans. temp, largest shift in trans. temp or decrease in upper shelf energy (USE)	3 or 5 based on predicted temp shift and EOL fluence value	Charpy – at least 15 Tension – at least 3 (not required if temp shift $\geq 100^{\circ}\text{F}$ or EOL fluence $\geq 5\text{E}18$ n/cm ²)	Charpy – 12 Tension – 2 base and weld only if temp shift $\geq 100^{\circ}\text{F}$ or EOL fluence $\geq 5\text{E}18$ n/cm ²	Normal to working direction Notch \perp to surface Illustrations provided	Specified % of life or specified trans. temp. shift with last capsule as a standby
1979	Beltline base, weld, and HAZ metal with highest init. trans. temp, largest shift in trans. temp, decrease in USE, or most limiting for setting temp/press limits	3, 4, or 5 based on predicted temp shift	Charpy – at least 18 Tension – at least 3 for base and weld only	Charpy – 12 Tension – 3 base and weld only	Same as above	At specified effective full-power years based on shifts in trans. temp with last capsule as a standby
1982	Same as above	Same as above	Same as above	Same as above	Same as above	Same as above
1993	Beltline base, and weld, metal only with highest init. trans. temp, largest shift in trans. temp, decrease in USE, or most limiting for setting temp/press limits	Same as above	Same as above	Same as above	Same as above	Same as above
1998	Same as above	Same as above	Same as above	Same as above	Same as above	Same as above
2002	Same as above	2, 3, or 4 based on predicted temp shift	Charpy – at least 15 Tension – at least 6 for base and weld only Fracture toughness – at least 8	Charpy – at least 15 Tension – 3 base and weld only Fracture toughness – 8 for limiting weld and base metal	Same as above	At specified fluences
2010	Limiting beltline base, weld and other; highest init + shift	3, or 4 based on predicted temp shift	Charpy – at least 15 Tension – at least 6 for base and weld only Fracture toughness – at least 8	Charpy – 15 Tension – 3 base and weld only Fracture toughness – 8 for limiting weld and base metal	Same as above	At fractions of RPV EOL fluence
2015	Same as above	4 + 1 Standby	Same as above	Same as above	Same as above	All fractions of MDF
2016	Same as above	4 + 1 Standby	Same as above	Same as above	Same as above	min 3 in first 40 years

Nuclear Reactors.” The standard required both tension and Charpy impact specimens and specified non-standard specimens as being permitted. Neutron fluence rate, fluence spectrum and irradiation temperature should duplicate what the vessel would experience. It recommended use of the temperature monitor melt wires. Correlation monitor material was to be included to help evaluate radiation damage. A detailed description of required tests results and analysis was to be reported.

X1.3 The first revision was issued as Practice E185-66 and the title changed to “Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors.” The standard specified that the base metal, weld metal, heat-affected-zone metal and correlation materials be included in the program and that all surveillance materials have the same fabrication history as the vessel materials. Added was a recommendation that thermal aging materials be included in a

thermal environment without neutron radiation. All test specimens were to be encapsulated for corrosion protection. Measurement of neutron exposure was performed per other ASTM methods. A withdrawal schedule for the surveillance capsules was specified. The specified testing and reporting requirements were expanded.

X1.4 The next revision was issued as Practice E185-70 and the title changed to “Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels.” This revision specified that one of the base metals included in the surveillance program shall be the material in the reactor vessel at the highest fluence location. In addition, the orientation of the specimens in the materials was specified with a diagram. The reporting requirements were expanded to include description of the reactor vessel, and a complete metallurgical history of each surveillance material.

X1.5 The next revision was issued as Practice E185-73 and again the title was changed to “Standard Recommended Practice for Surveillance tests for Nuclear Reactor Vessels.” The scope was expanded to better define the irradiation damage and neutron fluence limits. The revision specified the minimum number of Charpy impact and tension test specimens. A surveillance material selection procedure was established based on RT_{NDT} , Charpy upper-shelf energy, and phosphorus and copper contents. The requirement for thermal aging capsules was discontinued. A capsule withdrawal schedule based on expected irradiation damage was defined. The reporting requirements were further expanded to include all the new data to support program requirements.

X1.6 The next revision was issued as Practice E185-79 and the title changed to “Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels.” The scope was modified to specify one irradiation exposure limit for all reactor vessels. A definition section was added to clarify the meaning of new terminology. Evaluation of the radiation-induced shift was specified in terms of the 41J (30 ft-lbf) index temperature, replacing the 68J (50 ft-lbf) and 0.89 mm (35 mils) lateral expansion index temperature required in Practice E185-73. The materials requirements were changed to specify that the actual materials used in the fabrication of the vessel shall be included in the program. Sufficient material shall be provided to make additional capsules. Also, a complete fabrication and metallurgical history, as well as chemical analysis of each material, was to be reported. Dosimetry requirements were improved and temperature monitors were required. The capsule withdrawal schedule was modified to accommodate three ranges of predicted irradiation damage of the reactor vessel. Reporting requirements were further expanded to identify dosimetry analytical techniques and methods and more detailed dosimetry analysis results. Also, it was required to extrapolate the fluence results and fracture tough-

ness through the reactor vessel walls.

X1.7 The next revision of Practice E185 was issued as Practice E185-82 with no change in the title and only editorial changes. There were no technical requirement changes.

X1.8 The next revision of Practice E185 was issued as Practice E185-94 and updated as Practice E185-98. The major differences between ASTM Practice E185-82 and Practice E185-94 were the relaxation in the lead factor from 1-3 to 5 and the elimination of the requirement to include HAZ specimens in the capsule. The revision in Practice E185-98 added the alternative use of fracture toughness specimens for testing in accordance with other fracture toughness test methods.

X1.9 The 2002 revision involved splitting Practice E185 into two separate standards: Practice E185 on design of a new surveillance program and Practice E2215 on testing and evaluation of surveillance program capsules. The design standard incorporated some significant changes since more emphasis is being placed on measuring elastic-plastic fracture toughness in addition to Charpy V-notch toughness properties. The testing requirements did not change significantly from Practice E185-98, other than being separated out as a stand-alone ASTM standard.

X1.10 The 2010 revision made clarifications and improvements including removal of ART definition, addition of definition of limiting material and standby capsule. The materials selection section was rewritten for clarity. Orientation of fracture toughness specimens was added. The minimum lead factor was changed to 1.5 and the maximum to 5. The withdrawal schedule was changed to require withdrawal and testing of at least 3 capsules to account for a longer anticipated operating lifetime (60 years or more versus 40 years). The target fluences were changed for simplification.

X1.11 In the 2015 revision, the term end-of-license (EOL) was replaced with the term maximum design fluence (MDF) to provide a consistent means to scale the capsule withdrawal schedule independent of different definitions and lengths of license life adopted by different regulatory authorities. Dosimeter measurements other than from the capsules are recommended if there are long periods between capsule withdrawals. The amount of required archive material and the material sampling were changed in consideration of small modular reactor designs. The number of capsules required to be tested changed. An additional restriction on specimen location was added consistent with ASME NB-2223.

X1.12 The 2016 revision made clear that the use of E900, E482, and E853 is optional. The ability to use non-RV material for the surveillance weld was eliminated. The requirement to withdraw three capsules in the first 40 years was added.

ASTM International takes no position respecting the validity of any patent rights asserted in connection with any item mentioned in this standard. Users of this standard are expressly advised that determination of the validity of any such patent rights, and the risk of infringement of such rights, are entirely their own responsibility.

This standard is subject to revision at any time by the responsible technical committee and must be reviewed every five years and if not revised, either reapproved or withdrawn. Your comments are invited either for revision of this standard or for additional standards and should be addressed to ASTM International Headquarters. Your comments will receive careful consideration at a meeting of the responsible technical committee, which you may attend. If you feel that your comments have not received a fair hearing you should make your views known to the ASTM Committee on Standards, at the address shown below.

This standard is copyrighted by ASTM International, 100 Barr Harbor Drive, PO Box C700, West Conshohocken, PA 19428-2959, United States. Individual reprints (single or multiple copies) of this standard may be obtained by contacting ASTM at the above address or at 610-832-9585 (phone), 610-832-9555 (fax), or service@astm.org (e-mail); or through the ASTM website (www.astm.org). Permission rights to photocopy the standard may also be secured from the Copyright Clearance Center, 222 Rosewood Drive, Danvers, MA 01923, Tel: (978) 646-2600; <http://www.copyright.com/>