Designation: ~~E2215 – 10~~ 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 ~~radiation-induced~~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
~~E560 Practice for Extrapolating Reactor Vessel Surveillance Dosimetry Results, E 706(IC) (Withdrawn 2009)[4]~~
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

[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.

2.2 *ASME Standards:*[4]

~~American Society of Mechanical Engineers,~~ Boiler and Pressure Vessel Code, ~~Sections~~Section III ~~and XI~~ Subarticle NB-2000, Rules for Construction of Nuclear Facility Components, Class 1 Components, Materials

~~ASME Boiler and Pressure Vessel Code Case N-629, Use of Fracture Toughness Test Data to Establish Reference Temperature for Pressure Retaining Materials, Section XI, Division 1~~

~~ASME~~ Boiler and Pressure Vessel ~~Code Case N-631~~Code, Section XI ~~Use of Fracture Toughness Test Data to Establish Reference Temperature for Pressure Retaining Materials Other Than Bolting for Class 1 Vessels, Section III, Division 1~~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 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 ~~40.7~~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 ~~test~~ temperature ~~above~~at which the fracture appearance of all Charpy specimens tested is at or above 95 % shear.

3.1.7 *end-of-license ~~(EOL)—~~(EOL) fluence*—the ~~design lifetime in terms of years~~ 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 ~~peak~~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.~~surface calculated over the same time period.

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

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 ~~at EOL~~ 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 ~~reference temperature shift~~ TTS can be determined from the relationship found in Guide E900~~. The basis for selecting the limiting material shall be documented.~~ 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 embrittlement.~~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, ~~"Nuclear Power Plant Components"~~ for the definition of RT_{NDT} for unirradiated material based on Charpy (Test ~~Method~~Methods ~~E23~~A370) and drop weight tests (Test Method E208). ASME Code ~~Cases N-629 and N-631~~Section XI, Appendices A and G provide an alternative definition for the reference temperature ~~(RT~~(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,~~ location as defined by Practice E185~~but whose withdrawal~~ , 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 ~~standard~~ 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 ~~Charpy~~ index temperatures, the ~~Charpy~~ 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. ~~The application of this data is the subject of Guide E900 and other documents listed in Section 2.~~

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*~~J-integral techniques defined in Test Method E1820. Additionally, hardness testing can be used to supplement standard methods as a means of monitoring the ~~radiation~~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 ~~thermal~~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 ~~prior to capsule removal.~~experienced by the capsule. The local reactor coolant temperature may be used as a reasonable ~~approximation.~~ 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. ~~Vessel dimensional information and capsule locations~~ 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 ~~and E560~~.

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 on defining the ~~40.7~~41 J (30 ft-lbf) index temperature and the upper-shelf energy level. It is recommended that upper-shelf Charpy tests be conducted using at least three specimens tested and evaluated in accordance with 3.1.5 of this practice. Instrumented tests are recommended and should be performed in accordance with Guide E636.

7.3.3 *Measurements*—For each test specimen, measure the impact energy, lateral expansion, and percent shear fracture appearance.

7.4 *Hardness Tests (Optional)*—Hardness tests may be performed on irradiated Charpy specimens. The measurements shall be taken (prior to Charpy testing, if possible, to avoid sampling material deformed by the test) in areas away from the fracture zone or the edges of the specimens. The tests shall be conducted in accordance with Test Methods and Definitions A370.

7.5 *Fracture Toughness Tests (Optional):*

7.5.1 *Specimens*—~~Supplemental fracture~~ Fracture toughness tests may be conducted following Guide E636 using either fracture mechanics specimens from the surveillance capsule or broken Charpy specimens that have been ~~reconstituted.~~ reconstituted and precracked. Procedures for reconstitution of Charpy specimens are given in Guide E1253.

7.5.2 *Upper-Shelf Fracture Toughness*—Testing to characterize upper-shelf toughness using the ~~*J-integral*~~*J*-integral method should be conducted in accordance with Test Method E1820.

7.5.3 *Transition Fracture Toughness*—The reference temperature for ferritic steels in the transition range, T_o, can be established using the methodology provided in Test Method E1921.

7.6 *Retention of ~~Broken~~ Test Specimens*—It is recommended that all broken and unbroken test specimens be maintained in good condition and retained in the event that additional analysis is required to explain anomalous results. Identification of ~~broken~~all test

[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.

specimens ~~should be maintained.~~ shall be maintained. After it is determined that additional testing or analysis to explain anomalous results is not required, then it is recommended that specimens be either retained or used for appropriate research to increase understanding of embrittlement. Final disposition of specimens should only be performed after a thorough evaluation of the potential usefulness of the specimen materials. These specimens can provide materials that are useful for supporting reactor vessel license renewal or for verification of annealing.

8. Evaluation of Test Data

8.1 *Tension Tests:*

8.1.1 Determine the amount of ~~radiation~~radiation-induced strengthening and loss of ductility by comparing irradiated test results with unirradiated data from the surveillance capsule documentation package.

8.2 *Charpy Tests:*

8.2.1 *Curve Fitting*—Average curves shall be drawn through the Charpy data to ~~describe~~display the Charpy impact energy, lateral expansion and percent shear fracture appearance as a function of the test temperature. A similar analysis of unirradiated Charpy data from the surveillance capsule documentation should also be performed. The preferred method for determining the average curves is statistical fitting to a hyperbolic tangent function.[6]

8.2.2 Occasionally a single data point will unduly influence the average curve. In this case, the test record and specimen should be examined for possible causes of discrepancy and its disposition documented.

8.2.3 *Index Temperatures*—Charpy index temperatures shall be determined for the ~~40.7~~41 J (30 ft-lbf) energy level and 0.89 mm (35 mils) lateral expansion level. Optionally, the fracture appearance transition temperature corresponding to 50 % shear fracture can be determined. Radiation-induced shifts in the index temperatures shall be determined by ~~comparing~~subtracting the measured ~~values to values obtained from the analysis of the unirradiated data.~~unirradiated index temperatures from the irradiated index temperatures. If the differences among these three shift measurements exceed 15°C, then the test records and specimens should be examined for possible causes of discrepancy and the outcome of the examination documented.

8.2.4 ~~*Upper=Shelf*~~*Upper-Shelf Energy*—The Charpy upper-shelf energy should be determined according to the definition given in 3.1.5. The radiation-induced change in the upper-shelf energy shall be determined by comparing this data to unirradiated data from the surveillance capsule documentation.

8.3 *Reference Material*—If reference material specimens are included in the surveillance capsule, they shall be tested and evaluated. The measured irradiation response of the reference material specimens should fall within the scatter band of the pre-existing database.[7] In cases where the reference material test results exhibit excessive scatter relative to the available data, the source of the scatter should be investigated. Potential reasons that can be investigated include deviations from the expected surveillance capsule exposure conditions, a lack of uniformity of properties in the reference material itself, or both.

8.4 *Hardness Tests (Optional)*—The hardness data may be correlated to the yield or tensile strength of the ~~material.~~ material, or other parameters. Justification for any correlation used shall be provided with the report.

8.5 *Fracture Toughness Tests (Optional):*

8.5.1 *Upper-Shelf Fracture Toughness*—The resistance to crack initiation and extension on the upper shelf may be expressed in terms of the ~~*J-integral*~~*J*-integral as described in Test Method E1820.

8.5.2 *Transition Fracture Toughness*—An appropriate reference temperature for fracture toughness in the transition region can be determined using the procedure in Test Method E1921. This reference transition temperature can be used to define an alternate reference temperature ~~(RT~~(RT_{To} in place of RT_{NDT}) as defined in ASME Code ~~Cases N-629 and N-631.~~Section XI, Appendices A and G.

9. Withdrawal Schedule Review

9.1 The primary consideration in the review of the withdrawal schedule shall be ~~assuring~~ensuring that the vessel is appropriately monitored throughout its projected design life. This should include a review of the original objectives of the surveillance program and the adequacy of the program to meet future needs. This shall also include monitoring the neutron exposure of the reactor vessel throughout its projected design life using a combination of neutron fluence tracking analysis methods and fluence measurements. The fluence measurements may consist of both in-vessel and ex-vessel neutron dosimetry. This practice provides guidelines to aid in that analysis. The circumstances of any particular reactor surveillance program may require considerations of factors beyond these guidelines.

9.2 The withdrawal schedule shall be reviewed upon completion of testing of each of the surveillance capsules. Proposed adaptations must accommodate the restrictions imposed by the design of the original surveillance program. The number and

[6] ~~Mager, T. R., Server, W. L., and Beaudoin, B. F.,~~Eason, E. D., Wright, J. E., and Odette, G. R., ~~*"Use of the Hyperbolic Tangent Function for Fitting Transition Temperature Toughness*~~*Improved Embrittlement Correlations for Reactor Pressure Vessel Steels,* ~~Data," WCAP-14370, Westinghouse Electric Corporation, May 1995.~~NUREG/CR-6551, U.S. Nuclear Regulatory Commission, September 1998.

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

contents of the capsules included in a reactor vessel surveillance program will vary depending on the prevailing practice at the time the program was designed and the original perception of the radiation sensitivity of the vessel ~~materials.~~materials and reactor type.

9.3 ~~The withdrawal schedule is stated in terms of the end-of-license (EOL) fluences at the peak location on the vessel inner wall. Actual numbers will be determined by the effective capsule lead factors.~~*Withdrawal Schedule Review for Design Life:*

9.3.1 Update the MDF based on new dosimetry, fluence analysis and current operating plans.

9.3.2 Update the projected transition temperature shift (TTS) for the most limiting vessel material based on any new material specific information. TTS may be estimated using the equations provided in Guide E900.

9.3.3 Adjust the withdrawal schedule to meet the recommendations in Table 1.

9.4 *Anticipated Operation Beyond Design Life:*

9.4.1 When operation beyond design life is anticipated, a plan for reactor vessel surveillance should be developed to ensure that the vessel beltline is appropriately monitored throughout the period of operation. This may include fabrication and insertion of a new capsule, moving an existing capsule to a higher lead factor location, or participation in an integrated surveillance program.

9.4.2 Upate EOL fluence, TTS, EOL reactor vessel material property projections and limiting material for the operating period beyond design life.

9.4.3 The goal is to have limiting beltline material (or a surrogate material, if this is not practical) index temperature measurements at a fluence greater than the projected EOL renewal fluence, but less than twice the EOL renewal fluence. The testing should be performed before the limiting material vessel fluence reaches the fluence of the previous highest fluence surveillance measurement.

9.5 ~~The withdrawal schedule from the companion version of Practice~~ Capsules not required by the surveillance withdrawal schedule to be tested may be used to provide supplemental data. Supplemental testing may be required for plant license renewal or reactor vessel annealing programs following Guide ~~E185~~E509~~-09 is provided in~~. However, it is recommended that ~~Table 1~~capsule fluence not exceed twice MDF, or twice EOL fluence if operating beyond design life. Supplemental testing may also be based on reconstitution of previously tested specimens following Guide E1253.

~~9.4.1 The first step in the review of Table 1 is to update the estimated peak EOL vessel inside surface fluence (EOL ID) based on new dosimetry and current operating plans. These revised projections should be compared to the original basis.~~

~~9.4.2 The second step in the review of Table 1 is to update the projected transition temperature shift (ΔRT_{NDT}) for the most limiting vessel material based on any new material specific information or planned fuel management or other operation changes. ΔRT_{NDT} may be estimated using the equations provided in Guide E900. These revised projections should be compared to the original basis.~~

~~9.4.3 If a reactor vessel material is projected to exceed an EOL transition temperature shift (ΔRT_{NDT}) of 111°C (200°F), then a capsule with a target fluence of ½ peak EOL shall be withdrawn and tested. All plants for which the surveillance program was designed to Practice E185-09 are required to withdraw and test capsules at ¼ peak EOL, ¾ peak EOL and peak EOL target fluence.~~

~~9.4.4 Capsules not required by the surveillance withdrawal schedule may be used to provide supplemental data. Supplemental testing may be required for plant license renewal or reactor vessel annealing programs following Guide E509. However, it is recommended that capsule fluence not exceed 2 times peak EOL. Supplemental testing may also be based on reconstitution of previously tested specimens following Guide E1253.~~

~~9.4.5 The schedule for capsule withdrawals is approximate and may be adjusted to coincide with a planned refueling outage.~~

9.6 The schedule for capsule withdrawals is approximate and may be adjusted to coincide with a planned outage. A fluence higher than the target is preferred to a lower value.

TABLE 1 Suggested Withdrawal Schedule

Sequence	Target Fluence	Notes
~~First~~	~~¼ EOL ID~~	~~Testing Required~~
~~Second~~	~~½ EOL ID~~	~~Testing Required if projected ΔRT_{NDT} > 111°C (200°F)~~
~~Third~~	~~¾ EOL ID~~	~~Testing Required~~
~~Fourth~~	~~EOL ID~~	~~Testing Required~~
~~Standby~~	~~< 2 EOL ID~~	~~Testing Not Required~~

TABLE 1 Suggested Withdrawal Schedule[A,B]

Sequence	Target Fluence	Notes
First	¼ MDF	Testing Required
Second	½ MDF	Testing Required
Third	¾ MDF	Testing Required
Fourth	MDF	Testing Required
Standby	< 2 MDF	Testing Not Required

[A] If the original surveillance program contained less than 5 capsules, adjust the withdrawal schedule to provide monitoring through the period of operation attempting to meet the general intent of this guidance.
[B] See 9.4 for anticipated operation beyond design life.