Designation: E706 – 16

Standard Master Matrix for Light-Water Reactor Pressure Vessel Surveillance Standards[1]

This standard is issued under the fixed designation E706; 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 master matrix standard describes a series of standard practices, guides, and methods for the prediction of neutron-induced changes in light-water reactor (LWR) pressure vessel (PV) and support structure steels throughout a pressure vessel's service life (Fig. 1). Referenced documents are listed in Section 2. The summary information that is provided in Sections 3 and 4 is essential for establishing proper understanding and communications between the writers and users of this set of matrix standards. It was extracted from the referenced standards (Section 2) and references for use by individual writers and users. More detailed writers' and users' information, justification, and specific requirements for the individual practices, guides, and methods are provided in Sections 3 – 5. General requirements of content and consistency are discussed in Section 6.

1.2 This master matrix is intended as a reference and guide to the preparation, revision, and use of standards in the series.

1.3 To account for neutron radiation damage in setting pressure-temperature limits and making fracture analyses (**1-12**)[2] and Guide E509), neutron-induced changes in reactor pressure vessel steel fracture toughness must be predicted, then checked by extrapolation of surveillance program data during a vessel's service life. Uncertainties in the predicting methodology can be significant. Techniques, variables, and uncertainties associated with the physical measurements of PV and support structure steel property changes are not considered in this master matrix, but elsewhere (**2**, **6**, **7**), (**11-26**), and Guide E509).

1.4 The techniques, variables and uncertainties related to *(1)* neutron and gamma dosimetry, *(2)* physics (neutronics and gamma effects), and *(3)* metallurgical damage correlation procedures and data are addressed in separate standards belonging to this master matrix (**1, 17**). The main variables of concern to *(1)*, *(2)*, and *(3)* are as follows:

1.4.1 Steel chemical composition and microstructure,

1.4.2 Steel irradiation temperature,

1.4.3 Power plant configurations and dimensions, from the core periphery to surveillance positions and into the vessel and cavity walls.

1.4.4 Core power distribution,

1.4.5 Reactor operating history,

1.4.6 Reactor physics computations,

1.4.7 Selection of neutron exposure units,

1.4.8 Dosimetry measurements,

1.4.9 Neutron special effects, and

1.4.10 Neutron dose rate effects.

1.5 A number of methods and standards exist for ensuring the adequacy of fracture control of reactor pressure vessel belt lines under normal and accident loads ((**1**, **7**, **8**, **11**, **12**, **14**, **16**, **17**, **23-27**), Referenced Documents: ASTM Standards (2.1), Nuclear Regulatory Documents (2.3) and ASME Standards (2.4)). As older LWR pressure vessels become more highly irradiated, the predictive capability for changes in toughness must improve. Since during a vessel's service life an increasing amount of information will be available from test reactor and power reactor surveillance programs, procedures to evaluate and use this information must be used (**1**, **2**, **4-9**, **11**, **12**, **23-26**, **28**). This master matrix defines the current *(1)* scope, *(2)* areas of application, and *(3)* general grouping for the series of ASTM standards, as shown in Fig. 1.

1.6 The values stated in SI units are to be regarded as standard. No other units of measurement are included in this standard.

1.7 *This standard does not purport to address all of the safety concerns, if any, associated with its use. It is the*

[1] This practice is under the jurisdiction of ASTM Committee E10 on Nuclear Technology and Applications and is the direct responsibility of Subcommittee E10.05 on Nuclear Radiation Metrology.

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[2] The boldface numbers in parentheses refer to a list of references at the end of this standard.

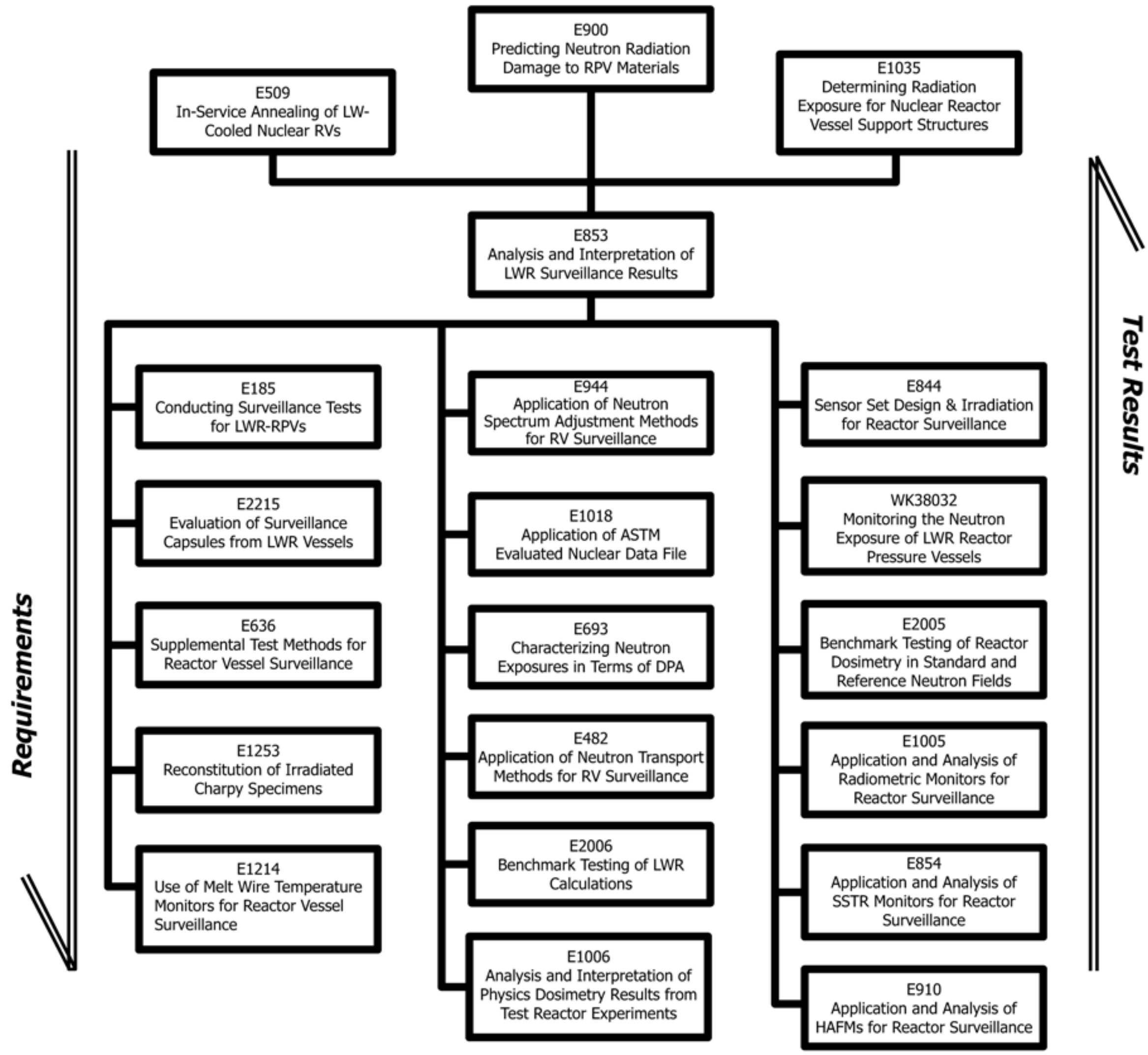

FIG. 1 Organization and Use of ASTM Standards in the E706 Master Matrix

responsibility of the user of this standard to establish appropriate safety and health practices and determine the applicability of regulatory limitations prior to use.

2. Referenced Documents

2.1 *ASTM Standards:*[3]

C859 Terminology Relating to Nuclear 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

E482 Guide for Application of Neutron Transport Methods for Reactor Vessel Surveillance

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)

E646 Test Method for Tensile Strain-Hardening Exponents (***n*** -Values) of Metallic Sheet Materials

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

E854 Test Method for Application and Analysis of Solid State Track Recorder (SSTR) Monitors for Reactor Surveillance, E706(IIIB)

[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.

E900 Guide for Predicting Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials
E910 Test Method for Application and Analysis of Helium Accumulation Fluence Monitors for Reactor Vessel Surveillance, E706 (IIIC)
E944 Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance, E 706 (IIA)
E1005 Test Method for Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance
E1006 Practice for Analysis and Interpretation of Physics Dosimetry Results from Test Reactor Experiments
E1018 Guide for Application of ASTM Evaluated Cross Section Data File, Matrix E706 (IIB)
E1035 Practice for Determining Neutron Exposures for Nuclear Reactor Vessel Support Structures
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
E2005 Guide for Benchmark Testing of Reactor Dosimetry in Standard and Reference Neutron Fields
E2006 Guide for Benchmark Testing of Light Water Reactor Calculations
E2215 Practice for Evaluation of Surveillance Capsules from Light-Water Moderated Nuclear Power Reactor Vessels
E2956 Guide for Monitoring the Neutron Exposure of LWR Reactor Pressure Vessels

2.2 *ASTM Adjunct:*[4]
ADJE090015-EA Adjunct for E900-15 Technical Basis for the Equation Used to Predict Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials

2.3 *Nuclear Regulatory Documents:*[5]
Code of Federal Regulations, Chapter 10, Part 50 Appendices G and H
Code of Federal Regulations, Chapter 10, Part 21 Reporting of Defects and Noncompliance
Regulatory Guide 1.99 Radiation Embrittlement of Reactor Vessel Materials
Regulatory Guide 1.150 Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations
Regulatory Guide 1.190 Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence

2.4 *American Society of Mechanical Engineers Standard:*[6]
Boiler and Pressure Vessel Code Sections III and XI

2.5 *Bureau International de Poids et Mesures Documents:*[7]
The SI Brochure: The International System of Units (SI)

[4] Available from ASTM International Headquarters. Order Adjunct No. ADJE090015-EA. Original adjunct produced in 2015.

[5] Available from Superintendent of Documents, U.S. Government Printing Office, Washington, DC 20402.

[6] Available from American Society of Mechanical Engineers (ASME), ASME International Headquarters, Two Park Ave., New York, NY 10016-5990, http://www.asme.org.

[7] Available from Bureau International de Poids et Mesures, http://www.bipm.org/en/publications/si-brochure/.

3. LWR Pressure Vessel Surveillance—Justification, Requirements, and Status of Work

3.1 Aging light water reactor pressure vessels (LWR-PV) accumulate significant neutron fluence exposures, with consequent changes in their state of steel embrittlement. Recognizing that accurate and validated measurement and predictive methods are needed to periodically evaluate the metallurgical condition of these reactor vessels, and in some instances reactor vessel support structures (**16, 17**), international multilaboratory work directed towards the improvement of LWR-PV surveillance has been conducted (**1, 2, 4, 29-34**).

3.2 The assessment of the radiation-induced degradation of material properties in a power reactor pressure vessel requires characterization of the neutron field from the edge of the reactor core to boundaries outside the pressure vessel. Measurements of neutron fluence, fluence rate, and spectrum for this characterization are associated with two distinct components of LWR-PV radiation surveillance procedures: *(1)* proper calculational estimates of the neutron fluence delivered to in-vessel surveillance positions, various locations in the vessel wall, and ex-vessel support structures and surveillance positions, and *(2)* understanding the interrelationship between material property changes in reactor vessels, in vessel support structures, and in metallurgical test specimens irradiated in test reactors and at accelerated neutron fluence positions near the pressure vessel in operating power reactions (see Sections 4 and 5).

3.3 The first component referred to above requires validation and calibration in a variety of neutron irradiation test facilities, including LWR-PV mock-ups, power reactor surveillance positions, and related benchmark neutron fields. The benchmarks also serve as a permanent measurement reference for neutron fluence and fluence rate detection techniques.

3.4 In order to meet the LWR-PV radiation monitoring requirements, a variety of neutron fluence, fluence rate, and damage detectors are employed. Each detector must be validated for application to the higher fluence rate and harder neutron spectrum of the test reactor test regions and to the lower fluence rate and softer neutron spectrum of the surveillance positions. Required detectors must respond to neutrons of various energies, so that multigroup spectra can be determined with accuracy sufficient for adequate damage response estimates for PV and support structure steels at end of life (EOL).

3.5 The necessity for well-established and documented test reactor and pressure vessel mock-up facilities for dosimetry and physics investigations and for irradiation of metallurgical specimens is recognized. These facilities provide well-characterized neutron environments where active and passive neutron dosimetry, various types of LWR-PV neutron field physics calculations, and temperature-controlled metallurgical damage exposures are brought together for validation and calibration. The neutron radiation field characteristics for surveillance capsule in- and ex-vessel power reactor positions are simulated in these mock-up facilities (**1, 35**).

3.6 A few operating PWR and BWR power reactor benchmark facilities have been selected for testing, validation, and calibration of physics computational methods, processing and adjustment codes, nuclear data, and dosimetry techniques (**1**, **3**, **35**).

3.7 Federal Regulation 10 CFR 50 calls for adherence to several ASTM standards that require establishment of a surveillance program for each power reactor and incorporation of fluence monitors for post-irradiation neutron field evaluation. Revised and new standards must be structured to be up-to-date, flexible, and, above all consistent (see Section 6).

4. Significance and Use

4.1 *Master Matrix*—This matrix document is written as a reference and guide to the use of existing standards and to help manage the development and application of new standards, as needed for LWR-PV surveillance programs. Paragraphs 4.2 – 4.5 are provided to assist the authors and users involved in the preparation, revision, and application of these standards (see Section 6).

4.2 *Approach and Primary Objectives:*

4.2.1 Standardized procedures and reference data are recommended in regard to *(1)* neutron and gamma dosimetry, *(2)* physics (neutronics and gamma effects), and *(3)* metallurgical damage correlation methods and data, associated with the analysis, interpretation, and use of nuclear reactor test and surveillance results.

4.2.2 Existing state-of-the-art practices associated with *(1)*, *(2)*, and *(3)*, if uniformly and consistently applied, can provide reliable (10 to 30 %, 1σ) estimates of changes in LWR-PV steel fracture toughness during a reactor's service life (**36**).

4.2.3 Reg. Guide 1.99 and Section III of the ASME Boiler and Pressure Vessel Code, Part NF2121 require that the materials used in reactor pressure vessels support "...shall be made of materials that are not injuriously affected by ...irradiation conditions to which the item will be subjected."

4.2.4 By the use of this series of standards and the uniform and consistent documentation and reporting of estimated changes in LWR-PV steel fracture toughness with uncertainties of 10 to 30 % (1σ), the nuclear industry and licensing and regulatory agencies can meet realistic LWR power plant operating conditions and limits, such as those defined in Appendices G and H of 10 CFR Part 50, Reg. Guide 1.99, and the ASME Boiler and Pressure Vessel Code.

4.2.5 The uniform and consistent application of this series of standards allows the nuclear industry and licensing and regulatory agencies to properly administer their responsibilities in regard to the toughness of LWR power reactor materials to meet requirements of Appendices G and H of 10 CFR Part 50, Reg. Guide 1.99, and the ASME Boiler and Pressure Vessel Code.

4.3 *Dosimetry Analysis and Interpretation*—(**1**, **3-5**, **21**, **28**, **29**, **35**, **37 and 38**) When properly implemented, validated, and calibrated by vendor/utility groups, state-of-the-art dosimetry practices exist that are adequate for existing and future LWR power plant surveillance programs. The uncertainties and errors associated with the individual and combined effects of the different variables (items 1.4.1–1.4.10 of 1.4) are considered in this section and in 4.4 and 4.5. In these sections, the accuracy (uncertainty and error) statements that are made are quantitative and representative of state-of-the-art technology. Their correctness and use for making EOL predictions for any given LWR power plant, however, are dependent on such factors as *(1)* the existing plant surveillance program, *(2)* the plant geometrical configuration, and *(3)* available surveillance results from similar plants. As emphasized in Section III-A of Ref (**7**), however, these effects are not unique and are dependent on *(1)* the surveillance capsule design, *(2)* the configuration of the reactor core and internals, and *(3)* the location of the surveillance capsule within the reactor geometry. Further, the statement that a result could be in error is dependent on how the neutron and gamma ray fields are estimated for a given reactor power plant (**1**, **11**, **28**, **36**, **39**, **40**). For most of the error statements in 4.3 – 4.5, it is assumed that these estimates are based on reactor transport theory calculations that have been normalized to the core power history (see 4.4.1.2) and not to surveillance capsule dosimetry results. The 4.3 – 4.5 accuracy statements, consequently, are intended for use in helping the standards writer and user to determine the relative importance of the different variables in regard to the application of the set of ASTM standards, Fig. 1, for *(1)* LWR-PV surveillance program, *(2)* as instruments of licensing and regulation, and *(3)* for establishing improved metallurgical data bases.

4.3.1 *Required Accuracies and Benchmark Field Referencing:*

4.3.1.1 The accuracies (uncertainties and errors) (Note 1) desirable for LWR-PV steel exposure definition are of the order of ±10 to 15 % (1σ) while exposure accuracies in establishing trend curves should preferably not exceed ±10 % (1σ) (**1**, **11**, **21**, **36**, **40-46**). In order to achieve such goals, the response of neutron dosimeters should therefore also be interpretable to accuracies within ±10 to 15 % (1σ) in terms of exposure units and be measurable to within ±5 % (1σ).

NOTE 1—Uncertainty in the sense treated here is a scientific characterization of the reliability of a measurement result and its statement is the necessary premise for using these results for applied investigations claiming high or at least stated accuracy. The term error will be reserved to denote a known deviation of the result from the quantity to be measured. Errors are usually taken into account by corrections.

4.3.1.2 Dosimetry "inventories" should be established in support of the above for use by vendor/utility groups and research and development organizations.

4.3.1.3 Benchmark field referencing of research and utilities' vendor/service laboratories has been completed that is:

–needed for quality control and certification of current and improved dosimetry practices;

–extensively applied in standard and reference neutron fields, PCA, PSF, SDMF, VENUS, NESDIP, PWRs, BWRs (**1**), and a number of test reactors to quantify and minimize uncertainties and errors.

4.3.2 *Status of Benchmark Field Referencing Work for Dosimetry Detectors*—PCA, VENUS, NESDIP experiments with and without simulated surveillance capsules and power reactor tests have provided data for studying the effect of deficiencies in analysis and interpretations; the PCA/PSF/SDMF perturbation experiments have provided data for more realistic PWR and BWR power plant surveillance capsule configurations and have permitted utilities' vendor/service laboratories to test, validate, calibrate, and update their practices (**1**, **4**, **5**, **47**). The PSF surveillance capsule test provided data, but of a more one-dimensional nature. PCA, VENUS, and NESDIP experimentation together with some test reactor work augmented the benchmark field quantification of these effects (**1**, **3**, **4**, **28**, **36**, **48-51**).

4.3.3 *Additional Validation Work for Dosimetry Detectors:*

4.3.3.1 Establishment of nuclear data, photo-reaction cross sections, and neutron damage reference files.

4.3.3.2 Establishment of proper quality assurance procedures for sensor set designs and individual detectors.

4.3.3.3 Interlaboratory comparisons using standard and reference neutron fields and other test reactors that provide adequate validations and calibrations, see Guide E2005.

4.4 *Reactor Physics Analysis and Interpretation*—(**1**, **3**, **5**, **11**, **28**, **35**, **36**, **39**, **52**) When properly implemented, validated, and calibrated by vendor/utility groups, state-of-the-art reactor physics practices exist that are adequate for in- and ex-vessel estimates of PV-steel changes in fracture toughness for existing and future power plant surveillance programs.

4.4.1 *Required Accuracies and Benchmark Field Referencing:*

4.4.1.1 The accuracies desirable for LWR-PV steel (surveillance capsule specimens and vessels) exposure definition are of the order of ±10 to 15 % (1σ). Under ideal conditions benchmarking computational techniques are capable of predicting absolute in- and ex-vessel neutron exposures and reaction rates per unit reactor core power to within ±15 % (but generally not to within ±5 %). The accuracy will be worse, however, in applications to actual power plants because of geometrical and other complexities (**1**, **3**, **4**, **11**, **21**, **36**, **37**, **38**, **39**, **52**).

4.4.1.2 Calculated in-and ex-vessel neutron and gamma ray fields can be normalized to the core power history or to experimental measurements. The latter may include dosimetry from surveillance capsules, other in-vessel locations, or ex-vessel measurements in the cavity outside the vessel. In each case, the uncertainty arising from the calculation needs to be considered.

4.4.2 *Power Plant Reactor Physics Analysis and Interpretation:*

4.4.2.1 *Result of Neglect of Benchmarking*—One quarter thickness location (1/4*T*) vessel wall estimates of damage exposure are not easily compared with experimental results. "Lead Factors," based on the different ways they can be calculated (fluence >0.1 or >1.0 MeV and dpa) may not always be conservative; that is, some surveillance capsules have been positioned in-vessel such that the actual lead factor is very near unity—no lead at all. Also the differences between lead factors based on fluence E > 0.1 or > 1 MeV and dpa can be significant, perhaps 50 % or more (**1**, **11**, **21**, **28**, **36**, **37**, **38**, **52**).

4.4.3 *PCA, VENUS, and NESDIP Experiments and PCA Blind Test:*

4.4.3.1 Test of transport theory methods under clean geometry and clean core source conditions shall be made. (**1**, **4**, **11**, **52**).

4.4.3.2 This is a necessary but not sufficient benchmark test of the adequacy of a vendor/utility groups' power reactor physics computational tools.

4.4.3.3 The standards recommendation should be that the vendor/utility groups' observed differences between their own calculated and the PCA, VENUS, and NESDIP measured integral and differential exposure and reaction rate parameters be used to validate and improve their calculational tools (if the differences fall outside the PCA, VENUS, and NESDIP experimental accuracy limits).

4.4.4 *PWR and BWR Generic Power Reactor Tests:*

4.4.4.1 Test of transport theory methods under actual geometry and variable core source conditions (**1**, **3**, **4**, **28**, **35**, **36**, **53**).

4.4.4.2 This is a necessary and partly sufficient benchmark test of the adequacy of a vendor/utility groups' power reactor physics computational tools.

4.4.4.3 The standards recommendation should be that the vendor/utility groups' observed differences between their own calculated and the selected PWR or BWR measured integral and differential exposure and reaction rate parameters be used to validate and improve their calculation tools (if the differences fall outside of the selected PWR or BWR experimental accuracy limits).

4.4.4.4 The power reactor "benchmarks" that have been established for this purpose are identified and discussed in Refs (**1**, **3**, **4**, **35**, **53**) and their references and in Guide E2006.

4.4.5 *Operating Power Reactor Tests:*

4.4.5.1 This is a necessary test of transport theory methods under actual geometry and variable core source conditions, but for a particular type or class of vendor/utility group power reactors. Here, actual in-vessel surveillance capsule and any required ex-vessel measured dosimetry information will be utilized as in 4.4.4 (**1**, **3**, **4**, **28**, **35**, **36**, **53**). Note, however, that operating power reactor tests are not sufficient by themselves (Reg. Guide 1.190, Section 4.4.5.1).

4.4.5.2 Accuracies associated with surveillance program reported values of exposures and reaction rates are expected to be in the 10 to 30 % (1σ) range (**36**).

4.5 *Metallurgical Damage Correlation Analysis and Interpretation*—(**1-8**, **10**, **11**, **13**, **15-29**, **36**, **37**, **38**)When properly implemented, validated, and calibrated by vendor/utility groups, state-of-the-art metallurgical damage correlation practices exist that are adequate for in- and ex-vessel estimates of PV-steel changes in fracture toughness for existing and future power plant surveillance programs.

4.5.1 *Required Accuracies and Benchmark Field Referencing:*

4.5.1.1 The accuracies desirable and achievable for LWR-PV steel (test reactor specimens, surveillance capsule specimens, and vessels and support structure) data correlation and data extrapolation (to predict fracture toughness changes both in space and time) are of the order of ±10 to 30 % (1σ). In order to achieve such a goal, however, the metallurgical parameters (ΔNDTT, upper shelf, yield strength, etc.) must be interpretable to well within ±20 to 30 % (1σ). This mandates that in addition to the dosimetry and physics variables already discussed that the individual uncertainties and errors associated with a number of other variables (neutron dose rate, neutron spectrum, irradiation temperature, steel chemical composition, and microstructure) must be minimized and results must be interpretable to within the same ±10 to 30 % (1σ) range.

4.5.1.2 Advanced sensor sets (including dosimetry, temperature and damage correlation sensors) and practices have been established in support of the above for use by vendor/ utility groups (**1**, **4**, **5**, **11**, **39**, **50**, **54**, **55**).

4.5.1.3 Benchmark field referencing of utilities' vendor/ service laboratories, as well as advanced, practices is in progress or being planned that is (**1**, **3-6**, **28**, **50**, **54-56**):

–needed for validation of data correlation procedures and time and space extrapolations (to PV positions: surface, 1/4 *T*, etc.) of test reactor and power reactor surveillance capsule metallurgical and neutron exposure data.

–being or will be tested in test reactor neutron fields to quantify and minimize uncertainties and errors (included here is the use of damage correlation materials—steel, sapphire, etc.).

4.5.2 *Benchmark Field Referencing*—The PSF (all positions: surveillance, surface, 1/4*T*, 1/2*T*, and the void box) together with the Melusine PV-simulator and other tests, such as for thermal neutron effects, provide needed validation data on all variables—dosimetry, physics, and metallurgy (**1**, **4**, **10**, **19**, **21**, **22**, **37**, **38**). Other test reactor data comes from surveillance capsule results that have been benchmarked by vendor/service laboratory/utility groups (**1**, **3**, **4**, **6**, **11**, **18**, **27**, **28**, **36**, **40-44**, **47**).

4.5.3 *Reg. Guide 1.99, NRC, EPRI Data Bases*—NRC and Electric Power Research Institute (EPRI) data bases have been studied on an ongoing basis by ASTM Subcommittees E10.02 and E10.05, vendors, utilities, EPRI, and NRC contractors to establish improved data bases for existing test and power reactor measured property change data. ASTM task groups recommend the use of updated and new exposure units (fluence total >0.1, >1.0 MeV and dpa) for the NRC, and EPRI data bases. (**1**, **2**, **6**, **7**, **13**, **18**, **27**, **36**, **40-44**, **47**)and incorporate these recommendations in the appropriate standards. ASTM subcommittee E10.02 has updated the embrittlement database and the prediction model in E900-15. The exposure unit used is total fluence for E>1MeV. The basis of the prediction model is documented in an adjunct associated with E900, available from ASTM.[4] The success of this effort depends on good cooperation between research and individual service laboratories and vendor/utility groups. An ASTM dosimetry cross section file based on the latest evaluations, as detailed in Guide E1018, and incorporating corrections for all known variables (perturbations, photo-reactions, spectrum, burn-in, yields, fluence time history, etc.) will be used as required and justified. Test reactor data will be addressed in a similar manner, as appropriate.

5. Master Matrix Description

5.1 The following index of ASTM standard practices, guides, and test methods constitutes the master matrix, describes the scope of individual standards, and provides other relevant information for the series of LWR-PV surveillance standards.[8] Fig. 1 indicates by title and ASTM designation the elements of this matrix standard and shows the general grouping for this series of ASTM standards.[9]

5.2 *Standards for Prediction and Management of Radiation Damage Effects:*

5.2.1 *Predicting Neutron Radiation Damage to Reactor Vessel Materials – E900 (E10.02)*[10]:

5.2.1.1 *Scope*—Guide E900 describes the metallurgical data base, the curve-fitting techniques, and the resulting property change versus exposure curves for the Charpy shift in brittle ductile transition temperature for LWR pressure vessel materials. The main variables of concern are: *(1)* steel types—material product form (plate, forging, or weldment), and chemical composition, *(2)* neutron irradiation temperature range, *(3)* neutron exposure units and values. This E706 ASTM guide relies on the application of several other ASTM standard practices, guides, and test methods.

5.2.1.2 *Discussion*—Commercial reactor vessels are required to have a surveillance program to monitor neutron induced changes in fracture toughness of the materials used in the construction of the reactor pressure vessel (see Section 1). The current practice is to estimate the fracture toughness using Charpy toughness data (**2**, **6-8**, **12**, **14**, **20**, **23-27**, **40**, **46**, **47**, **57**, **58**). To ensure conservative operational margins for nuclear power plants, accepted and accurate predictions of Charpy V-notch transition temperature are therefore necessary (**7**, **8**, **13**, **14**, **23-27**, **40**, **59**).

5.2.2 *In-Service Annealing of Light-Water-Cooled Nuclear Reactor Vessels – E509 (E10.02):*

5.2.2.1 *Scope and Discussion*—Guide E509 describes the procedures to be considered for conducting an in-service thermal anneal of a light-water-cooled nuclear reactor vessel and demonstrating the effectiveness of the procedure. The purpose of the in-service heat treatment is to improve the mechanical properties of the reactor vessel materials previously degraded by neutron embrittlement. The guide describes certain inherent limiting factors which must be considered in developing the annealing procedure. It also provides direction for the development of the annealing procedure and a post-annealing vessel surveillance program to monitor the effects of subsequent irradiation of the annealed-vessel beltline materials.

5.2.3 *Determining Radiation Exposures for Nuclear Reactor Support Structures – E1035 (E10.05):*

5.2.3.1 *Scope*—Practice E1035 covers the analyses and experimental methods necessary to establish a formalism to evaluate the radiation exposure for nuclear reactor support structures. This practice is applicable for all pressurized water reactors whose vessel supports will experience a lifetime

[8] For standards that are in the draft stage and have not been assigned an ASTM number, the Master Matrix will be very explicit and provide necessary detailed information on the procedures and data that are expected to be recommended in unnumbered reference standards.

[9] Cross referencing of these standards is to be done by means of the designations given in Fig. 1. Therefore, the Analysis and Interpretation of Nuclear Reactor Surveillance results Practice should be referred to as E853.

[10] Indicates the ASTM E10 subcommittee that has the primary responsibility for the preparation of the standard.

neutron fluence (E > 1 MeV) that exceeds 1×10^{17} neutron per square centimeter or 3×10^{4} displacements per atom. Its interrelationship to other Master Matrix E706 standards is shown in Fig. 1.

5.2.3.2 *Discussion*—Prediction of neutron irradiation effects on pressure vessel steels has long been a part of the design and operation of pressurized water reactor power plants as evidenced by the evolution of the Master Matrix E706. Reactor vessel support structures, depending on their location, may also experience neutron irradiation effects (**1, 10, 17, 47**). Application of this practice affords a quantitative assessment of the magnitude of that neutron irradiation. This practice, along with its sister practices, outlines the state-of-the-art requirements for the physics and dosimetry necessary to determine neutron exposures of support structures (**16**).

5.2.4 *Analysis and Interpretation of Nuclear Reactor Surveillance Results – E853 (E10.05):*

5.2.4.1 *Scope*—Practice E853 covers the methodology to be used in the analysis and interpretation of neutron exposure data obtained from LWR pressure vessel surveillance programs; and, based on the results of that analysis, establishes a formalism to be used to evaluate the present and the future condition of the pressure vessel and its support structures. This practice relies on, and ties together, the application of several supporting ASTM standard practices, guides, and test methods (see Fig. 1). In order to make this practice at least partially self-contained, however, a moderate amount of discussion is provided in areas relating to these ASTM standards and other documents. Support subject areas that are discussed include reactor physics calculations, dosimeter selection and analysis, and exposure units.

5.2.4.2 *Discussion*—This practice describes the best available procedures for the determination and evaluation of neutron exposure data that will, in turn, be used for reactor pressure vessel toughness and embrittlement predictions (E900). It can be referenced as an instrument of licensing and regulation and can be used for the establishment of improved metallurgical data bases. These improved data can be used for helping to predict the future condition of the pressure vessel. These same procedures, in conjunction with the use of Practice E1035, can be used to help predict the condition of pressure vessel support structures. The analysis and interpretation steps contained in this master practice are outlined in Table 1. This practice is intended for use in direct applications related to surveillance programs that are established in support of the operation, licensing, and regulation of LWR nuclear power plants. Procedures and data related to the analysis, interpretation, and application of test reactor results are addressed in Practice E1006.

5.3 *Mechanical Properties Surveillance Standards:*

5.3.1 *Design of Surveillance Programs for Light-Water Moderated Nuclear Power Reactor Vessels E185 (E10.02):*

5.3.1.1 *Scope*—Practice E185 covers procedures for monitoring the radiation-induced changes in the mechanical properties of ferritic materials in the beltline of light-water cooled nuclear power reactor vessels. This practice includes guidelines for designing a minimum surveillance program, selecting materials, and evaluating test results. This practice was developed for all light-water cooled nuclear power reactor vessels for which the predicted maximum neutron fluence (E > 1 MeV) at the end of the design lifetime exceeds 1×10^{21} n/m^2 (1 ×

TABLE 1 Procedures for Analysis and Interpretation of Nuclear Reactor Surveillance Results

Step	Procedure
1	Establish the basic surveillance test program for each operating power plant. Currently, Practice E185 is available and is used. However, updated versions of this practice should include the following: (a) Determination of surveillance capsule spatial fluence spectral and dpa maps for improved correlation and application of measured property change data (upper shelf, ΔNDTT, etc.). Measured surveillance capsule fission and nonfission monitor reaction and reaction rate date should be combined with reactor physics computations to make necessary adjustments for capsule perturbation effects. (b) As appropriated, use of measured/calculated dpa damage for normalization of Charpy to Charpy (and other metallurgical specimen) variations in neutron fluence, fluence rate, and spectra. Here, an increased use of a large number of metallurgical specimen iron drillings may be appropriate for dosimetry.
2	Establish a reactor physics computational method applicable to the surveillance program. Currently, Guide E482 provides general guidance in this area. However, updated versions should include the following: (a) Determination of core power distributions applicable to long-term (30 to 60-year) irradiation. Associated with this is the need for the use of updated FSAR (Final Safety Analysis Report) reactor physics information at startup. (b) Determination of potential cycle-to-cycle variations in the core power distributions. This will establish bounds on expected differences between surveillance measurements and design calculations. Ex-vessel dosimetry measurements should be used for verification of this and the previous step. (c) Determination of the effect of surveillance capsule perturbations and photofission on the evaluation of capsule dosimetry. Adjustments codes should be used, as appropriate, to combine reactor physics computations with dosimetry measurements. (d) Benchmark validation of the analytical method.
3	Establish methods for relating dosimetry, metallurgy, and temperature data from this surveillance program to current and future reactor vessel and support structure conditions. Currently, recommended Practice E853 provides general guidance in this area. An updated version of this standard should include the following considerations: (a) Improved temperature monitoring. (b) Exposure units to be used to correlate observed changes in upper shelf and RT_{NDT} with neutron environment. This should lead to improved adjustments in trend curves for upper shelf and RT_{NDT}. (c) Differences in core power distributions which may be expected during long-term operation and which may impact the extrapolation of surveillance results into the future. As previously stated, ex-vessel dosimetry should be used for verification.
4	Establish methods to verify Steps 2 and 3 and to determine uncertainty and error bounds for the interpretation of the combined results of dosimetry, metallurgical, and temperature measurements. Currently, Practice E185 provides general guidance in this area. An updated version of this standard should more completely address the separate and combined accuracy requirements of dosimetry, metallurgy, and temperature-measurement techniques.

10^{17} n/cm^2) at the inside surface of the reactor vessel. Between its provisional adoption in 1961 and 2015, Practice E185 has been revised many times. Code of Federal Regulations, Chapter 10, Part 50, Appendices G and H require adherence to versions up to and including only E185-82, and has yet to recognize subsequent versions. Later versions contain revised guidance which should be followed in cases that do not conflict with the requirements of Appendices G and H, however. 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. Significant differences between ASTM E185 revisions are listed in a table in the current version. 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.

5.3.1.2 *Discussion*—Predictions of neutron radiation effects on pressure vessel steels are considered in the design of lightwater cooled nuclear power reactors. Changes in system operating parameters are made throughout the service life of the reactor vessel to account for radiation effects. Because of 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 given 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 tests yield results useful for the evaluation of radiation effects on the reactor vessel. 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 the radiation effects. As a large amount of pertinent data becomes available it may be possible to reduce the surveillance effort for selected reactors by integrating their surveillance programs.

5.3.2 *Evaluation of Surveillance Capsules from Light-Water Moderated Nuclear Power Reactor Vessels – E2215 (E10.02):*

5.3.2.1 *Scope*— This practice covers the evaluation of test specimens and dosimetry from light water moderated nuclear power reactor pressure vessel surveillance capsules. This practice is one of a series of standard practices that outlines the surveillance program required for nuclear reactor pressure vessels. The surveillance program monitors the radiation-induced changes in the ferritic steels that comprise the beltline of a light-water moderated nuclear reactor pressure vessel. 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.[11]

5.3.2.2 *Discussion*— 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 revised Practice E185 that covers the design of new surveillance programs and this standard practice, E2215, that covers the testing and evaluation of surveillance capsules. Modifications to the standard test program and supplemental tests are described in Guide E636.

5.3.3 *Supplemental Test Methods for Nuclear Reactor Vessel Surveillance – E636 (E10.02):*

5.3.3.1 *Scope*—Practice E636 covers test methods and procedures that can be used in conjunction with, but not as alternatives to, those required by Practice E185 for the surveillance of nuclear reactor vessels. The supplemental test methods outlined include the compact toughness test, the precracked Charpy impact test, the instrumented Charpy V-notch test, and the dynamic tear test, and permit the acquisition of additional information on radiation-induced changes in fracture toughness and strength properties of the reactor vessel steels. This practice provides guidance in the preparation of test specimens for irradiation and identifies special precautions and requirements for reactor surveillance operations and post irradiation test planning. Guidance on data reduction and computational procedures is also given for individual test methods. Reference is made to other ASTM methods for the physical conduct of specimen tests and for raw data acquisition.

5.3.3.2 *Discussion*—Practice E185 describes a minimum program for the surveillance of reactor vessel mechanical property changes in service for the case where monitoring is required. Practice E636 may be applied where irradiation space limitations are not overly stringent and where the inclusion of additional specimen types is desirable to generate expanded information on radiation-induced property changes to assist the determination of best reactor vessel operation schemes.

5.3.4 *Guide for Reconstitution of Irradiated Charpy Specimens – E1253(E10.02):*

5.3.4.1 *Scope and Discussion*—There are occasions where either no full size Charpy specimen blanks are available or the material available with the desired irradiation history is not sufficient for machining of full size specimen. Guide E1253 describes the procedures for the reconstitution of Test Methods E23 Type A Charpy specimens from materials irradiation programs by welding end tabs of similar material onto remachined specimen sections that were unaffected by the initial test. Guidelines are given for the selection of suitable specimen halves and end tab materials, for dimensional control, and for avoidance of overheating the notch area.

[11] Prior to the adoption of this practice, surveillance capsule testing requirements were only contained in Practice E185.

5.3.5 *Use of Melt Wire Temperature Monitors for Reactor Vessel Surveillance – E1214 (E10.02):*

5.3.5.1 *Scope and Discussion*—Guide E1214 describes the application of temperature monitors and their use for reactor vessel surveillance of light-water power reactors as called for in Practice E185. The purpose of this practice is to recommend the selection and use of the common melt wire technique where the correspondence between melting temperature and composition of different alloys is used as a passive temperature monitor. Guidelines are provided for the selection and calibration of monitor materials; design, fabrication and assembly of monitor and container; post-irradiation examination; interpretation of the results; and estimation of uncertainties. This method is referenced and used in conjunction with Guide E844 and is intended for use for light-water power reactors.

5.4 *Computational Methodology Standards:*

5.4.1 *Application of Neutron Spectrum Adjustment Methods- E944 (E10.05):*

5.4.1.1 *Scope*—Practice E944 describes the procedures and codes recommended for use for the determination of neutron fluence spectra from multiple sensor measurements. The procedures described are, primarily, to be used for test reactor and power reactor measurements for light water reactors. The applicable range of neutron energies is from 0 to 20 MeV, provided appropriate detector response functions and input spectra (from physics calculations) are available. This guide addresses the uncertainties and errors associated with derived integral neutron field characterization and exposure parameters (total and thermal fluence and fluence rates, fluence >0.1 and >1.0 MeV and dpa).

5.4.1.2 *Discussion*—The use of test reactor and power reactor surveillance results for the prediction of EOL pressure vessel and support structures steel changes in fracture toughness requires the measurement and determination of neutron fluence spectra for neutron energies in the range from 0 to 20 MeV. For neutron energies below about 1.0 MeV, the information is needed for the assessment of the effect of lower energy neutrons on steel damage and on the interpretation and application of multiple sensor measurements. That is, *(1)* for the adjustment of reactor physics results in the thermal, 1/E, 0.01 to 1.0 MeV transition range, and a fast region above about 1.0 MeV, *(2)* for the determination of exposure values (total and thermal fluence, and fluence rate, fluence >0.1 MeV and dpa), and *(3)* for making corrections for target and product burn-in and burn-out effects for individual sensors and sensor covers (cadmium and gadolinium) (**1**, **21**, **36**, **37**, **38**, **39**, **46**).

5.4.2 *Application of ASTM Evaluated Cross Section Data File – E1018 (E10.05):*

5.4.2.1 *Scope*—Guide E1018 covers the establishment and use of an ASTM cross section and uncertainty/error file for *(1)* the analysis of single or multiple sensor measurements in LWR neutron fields, and *(2)* the calculation of spectral averaged damage cross sections for steel and for sensors that might be used as damage exposure monitors. The neutron fields include surveillance positions in operating power reactors, test reactor regions, and benchmark neutron fields. This guide describes requirements for the file, including data format, individual cross section evaluations and adjustments, and uncertainty/error estimates. The recommended cross sections are available as a single file from ASTM, along with the E1018 standard, or as individual source evaluations that can be obtained from one of the four national nuclear data centers:

–USA National Nuclear Data Center (NNDC) at Brookhaven National Laboratory, USA.

–Russian Nuclear Data Center at Fiziko-Energeticheskij at Obninsk, Russia.

–NEA Data Bank at Saclay, France.

–IAEA Nuclear Data Section at Vienna, Austria.

5.4.2.2 *Discussion*—Guide E1018 is directly related to and should be used in conjunction with Guide E944. The ASTM cross section file represents a generally available data set for use in sensor set analysis (**46**). However, the availability of this data set does not preclude the use of other validated data either proprietary or nonproprietary. Uncertainties and errors are specified in a coarser group structure including suggestions for assigning covariances between the groups. This information is required for the least squares adjustment methods applied to the determination of fluence spectra (see Guide E944).

5.4.3 *Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacements Per Atom – E693 (E10.05):*

5.4.3.1 *Scope*—Practice E693 describes a standard procedure for characterizing neutron irradiations of iron (and ferritic steels) in terms of the exposure index displacements per atom (dpa). It is assumed that the displacement cross section for iron is an adequate approximation for any ferritic steel. The application of this practice requires knowledge of the total fluence and the neutron-fluence spectrum and the availability of a cross section file, and is discussed in 5.3.2.

5.4.3.2 *Discussion*—A pressure vessel surveillance program requires a methodology for relating radiation-induced changes in materials exposed in test reactors and accelerated surveillance locations to the condition of the pressure vessel and support structures. An important consideration is that the irradiation exposures be expressed in a unit that is physically related to the damage mechanism (**1**, **2-4**, **10**, **13**, **16-18**, **36**). A primary source of neutron radiation damage in metals is the displacement of atoms from their normal lattice sites. Therefore, an appropriate damage exposure index is the number of times, on the average, that an atom has been displaced during an irradiation. This can be expressed as the total number of displaced atoms per unit volume, per unit mass, or per atom of the material. Displacements per atom is the most common. The number of dpa associated with a particular irradiation depends on the amount of energy deposited in the material by the neutrons, hence, depends on the neutron spectrum and fluence. No simple correspondence exists in general between dpa and a particular change in a material property. An appropriate starting point, however, for relative correlations of property changes produced in different neutron spectra is the dpa value associated with each environment. That is, the dpa values themselves provide a spectrum-sensitive index that may be a useful correlation parameter, or some function of the dpa values may affect correlation. The currently recommended dpa cross sections in this practice were generated using the iron ENDF/B-VI iron cross section (**60**). A recent calculation using

ENDF/B-VII produced identical results. (**61**, **62**) Although the ENDF/B-VI based dpa cross section differs from the previously recommended ENDF/B-IV dpa cross section (**60**) by about 60 % in the energy region around 10 keV, by about 10 % for energies between 100 keV and 2 MeV, and by a factor of 4 near 1 keV due to the opening of reaction channels in the cross section, the integral iron dpa values are much less sensitive to the change in cross sections. The update from ENDF/B-IV to ENDF/B-VI dpa rates when applied to the H. B. Robinson-2 pressurized water reactor resulted in "up to approximately 4 % higher dpa rates in the region close to the pressure vessel outer surface" and in "slightly lower dpa rates ... close to the pressure vessel inner wall" (**63**, **64**).

5.4.4 *Application of Neutron Transport Methods for Reactor Vessel Surveillance – E482 (E10.05):*

5.4.4.1 *Scope*—Guide E482 describes the methodology for performing radiation transport calculations to determine the neutron and gamma spectra within LWR research and power reactors. These calculations are required as a basis of the correlation of research and power reactor results and subsequent prediction of the EOL fracture toughness of LWR pressure vessel and support structure steel components. The accuracy of reactor physics calculations is considered together with benchmarking procedures for validating and calibrating the results of computations, see 4.4, (**1**, **11**, **21**, **28**, **29**, **36**, **37**, **38**, **39**, **52**).

5.4.4.2 *Discussion*—This guide is used as a reference in other ASTM standards when reactor physics (neutron and gamma) computations are recommended for LWR test and power reactor environmental characterization.

5.4.5 *Benchmark Testing of Light Water Reactor Calculations – E2006 (E10.05):*

5.4.5.1 *Scope*—Guide E2006 describes and provides reference information on *(1)* experimental benchmarking of neutron fluence calculations in more complex geometries relevant to pressure vessel surveillance and *(2)* the use of plant specific measurements to indicate bias in individual plant calculations

5.4.5.2 *Discussion*—This guide deals with the difficult problem of benchmarking neutron transport calculations carried out to determine fluences for plant specific reactor geometries. The calculations are necessary for fluence determination in locations important for material radiation damage estimation and which are not accessible to measurement. The most important application of such calculations is the estimation of fluence within the reactor vessel of operating power plants to provide accurate estimates of the irradiation embrittlement of the base and weld metal in the vessel. The benchmark procedure must not only prove that calculations give reasonable results but that their uncertainties are propagated with due regard to the sensitivities of the different input parameters used in the transport calculations.

5.4.5.3 The benchmarking processes outlined above will serve to indicate the calculational bias and allow uncertainty estimates to be made. Typical calculational (analytic) uncertainty estimates for the fast neutron fluence rate (E > 1 MeV) are 15 to 20 % (1σ) (**8**, **39**, **65-69**) at the inside of the reactor vessel and may be as large as 30 % in the cavity. Using the benchmark results is expected to lower the uncertainty in the fast neutron fluence rate to ~10 to 15 % at most locations in the region that is inside the pressure vessel and covers about 80 % of the active fuel height centered around the fuel mid-plane. The fast neutron fluence rate uncertainty at other locations is expected to be similar, but somewhat larger.

5.4.6 *Practice for Analysis and Interpretation of Physics Dosimetry for Test Reactors – E1006 (E10.05):*

5.4.6.1 *Scope*—Practice E1006 describes the methodology used in the analysis and interpretation of physics-dosimetry results from test reactors (**1**, **2**, **10**, **11**, **21**, **37**, **38**, **47**, **57**). The practice relies on, and ties together, the application of several supporting ASTM standard practices, guides, and methods. Support subject areas that are discussed include reactor physics calculations, dosimeter selection and analysis, exposure units, and neutron spectrum adjustment methods. This practice is directed towards the development and application of physics-dosimetry-metallurgical data obtained from test reactor irradiation experiments that are performed in support of the operation, licensing, and regulation of LWR nuclear power plants. It specifically addresses the physics-dosimetry aspects of the problem. Procedures related to the analysis, interpretation, and application of both test and power reactor physics-dosimetry-metallurgy results are addressed in Practice E853; Practice E185; Practice E2215; Practice E1035; Guide E900; and Test Method E646.

5.4.6.2 *Discussion*—This practice presents the best currently available methods for the determination of damage related fluence received by metallurgical specimens from irradiation experiments in test reactors. Application of this practice provides reliable and uniform input data from data bases pertaining to radiation damage of reactor materials.

5.5 *Dosimetry Sensor Measurement Standards:*

5.5.1 *Sensor Set Design and Irradiation for Reactor Surveillance – E844 (E10.05):*

5.5.1.1 *Scope*—Guide E844 covers the selection, design, irradiation, and post-irradiation handling of radiometric monitors (RM), solid state track recorders (SSTR), helium accumulation fluence monitors (HAFM), and temperature monitors (TM) sensors and sensor sets. It includes the consideration of sensor and sensor set placement, sensor set covers (thermal neutron shields), target and product burn-in and burn-out effects, photo-reaction effects, quality control of constituents, mass assay, and sensor and sensor set perturbations of the irradiation and the thermal temperature environments. Its use is primarily for test reactor and power reactor measurements for light-water reactors.

5.5.2 *Monitoring the Neutron Exposure of LWR Reactor Pressure Vessels – E2956 (E10.05):*

5.5.2.1 Scope— This guide establishes the means and frequency of monitoring the neutron exposure of the LWR reactor pressure vessel (including the extended beltline) throughout its operating life. The physics-dosimetry relationships determined from this guide may be used to estimate reactor pressure vessel damage through the application of Practice E693 and Guide E900, using fast neutron fluence (E > 1.0 MeV and E > 0.1 MeV), displacements per atom – dpa, or damage-function-correlated exposure parameters as independent exposure variables.

5.5.2.2 *Discussion*—This guide is intended to be used together with other standards to provide best estimates of the neutron exposure and exposure rate (together with uncertainties) at positions at the inner diameter and within the pressure vessel wall of a light water reactor. Also provided are estimates of gamma-ray exposure and exposure rates to interpret dosimetry sensor photo-reaction and other gamma-ray induced effects. Information used to make these estimates is obtained from coupled neutron-gamma ray transport calculations and from neutron and gamma-ray sensors located in surveillance positions on the core side of the vessel and in the reactor cavity outside the vessel wall (**1**). Benchmark field irradiations of similar monitors also provide valuable information used in the verification of the accuracy of the calculations (**1**).

5.5.3 *Benchmark Testing of Reactor Dosimetry in Standard and Reference Neutron Fields – E2005 (E10.05):*

5.5.3.1 *Scope*—Guide E2005 describes and provides reference information on *(1)* the application of standard and reference neutron fields to the calibration of integral neutron sensors and the validation of techniques and nuclear data used to measure neutron fluence rate, fluence, and fission rates, *(2)* the procedures for fluence rate transfer, and *(3)* spectral index calibration and measurement

5.5.3.2 *Discussion*—This guide describes the methodology for using the well-characterized standard and reference neutron fields to perform calibrations of fast neutron sensors and to validate the performance of laboratories engaged in read-out of neutron sensors for neutron dosimetry in LWRs.

5.5.4 *Analysis of Radiometric Monitors for Reactor Vessel Surveillance – E1005 (E10.05):*

5.5.4.1 *Scope and Discussion*—Test Method E1005 describes the use of Radiometric Monitors (RM) for neutron dosimetry in LWR applications (**1**, **11**, **21**, **29**, **36**, **37**, **38**, **39**, **50**, **52**). Measurement procedures for RM sensors by means of gamma ray or X-ray emission detection are specified. The assessment and discussion of methods and techniques for estimating uncertainties and errors are an important part of this standard (see 6.2 – 6.4). Test Method E1005 is referenced and used in conjunction with Guide E844 on Sensor Set Design. It is intended for use for test reactor and power reactor measurements for light-water reactors.

5.5.5 *Analysis of Solid State Track Recorder Monitors for Reactor Vessel Surveillance – E854 (E10.05):*

5.5.5.1 *Scope and Discussion*—Test Method E854 describes the use of Solid State Track Recorders (SSTR) for neutron dosimetry in LWR applications (**1**, **11**, **39**, **54**, **55**). Measurement procedures are specified for SSTR sensors by means of track counting techniques (**54**). The assessment and discussion of methods and techniques for estimating uncertainties and errors are an important part of this standard (see 6.2 – 6.4). Test Method E854 is referenced and used in conjunction with Guide E844 on sensor set design. It is intended for use for test reactor and power reactor measurements for light-water reactors.

5.5.6 *Analysis of Helium Accumulation Fluence Monitors for Reactor Vessel Surveillance – E910 (E10.05):*

5.5.6.1 *Scope and Discussion*—Test Method E910 describes the use of Helium Accumulation Fluence Monitors (HAFM) for neutron dosimetry in LWR applications (**1**, **11**, **56**). Applications that are covered in this test method include analysis and interpretation of helium generation data from HAFM capsules, unencapsulated radiometric monitors, Charpy specimens, and reactor vessel wall samples. The assessment and discussion of methods and techniques for estimating uncertainties and errors are an important part of this standard (see 6.2 – 6.4). Test-Method E910 is referenced and used in conjunction with Guide E844. It is intended for use for test reactor and power reactor measurements for light-water reactors.

6. General Requirements of Content and Consistency

6.1 Comments on these standards, their revision, and use should be considered at three levels. The first two levels represent necessary committee functions; the third includes remarks about details that may help the authors and users

6.1.1 *General Content*— Topics, issues, and data that need to be covered (see Sections 1, 3, 4, and 5).

6.1.2 *Organization and Consistency with Other Standards and References*—Complementarity in content, avoiding overlap with other standards, consistency in terminology, symbols, definitions, etc. Reference should be made to the appropriate Section 2 and other applicable documents and references

6.1.3 *Editorial Suggestions—Form and Style for ASTM Standards.*[12]

6.2 Standards should identify and discuss all elements of accuracy (**1**, **3-6**, **11**, **18**, **21**, **22**, **29**, **36**, **37**, **38**, **39**, **46**, **52**). Analysis and measurement accuracies (uncertainties, errors, and correlations) in the areas of concern for this set of matrix standards may be difficult to determine or estimate as discussed in Section 4. Difficult or not, they should be properly addressed in each standard (**11**, **21**, **22**, **35**, **36**, **39**, **46**, **50**). When uncertainties, errors, and correlations are well-identified, as in integral reaction rate measurements, for example, they should be estimated and summarized in an accuracy table. For more difficult kinds of uncertainties, errors, and correlations (for example, in spectrum adjustments) a state-of-the-art analysis will have to be chosen and a statement will have to be included that indicates what the uncertainty, error, and correlation estimates do and do not cover. It will be necessary to accept incomplete or nonrigorous uncertainty, error, and correlation estimates when there is no readily available alternative. This is necessary because it is a basic program purpose to improve the existing situation regarding accuracy (**1**, **11**, **13**, **21**, **22**, **35**, **36**, **39**, **46**, **52**).

6.3 Standards should be complete with regard to measurement and interpretation issues. All elements of the measurement and analysis issues should be identified and treated to the extent present capabilities allow. The Section 2references should be used and referenced, as appropriate. Problems should not be ignored because there is no proper answer yet. Semi quantitative, advisory, or cautionary statements are often used in ASTM standards. Field perturbations and photoreactions are two examples of measurement problems that are not always well understood and require benchmarking (**1**, **3**, **4**, **11**, **21**, **22**, **28**, **35**, **39-47**, **50**, **52**, **57**).

[12] Available from the ASTM website: www.astm.org.

6.4 Standards concerned with sensor measurement, data interpretation or transport calculation should refer to benchmark field calibration and validation. The dosimetry improvement effort centers around benchmark calibration and validation of measurement and calculational tools (see Sections 3 and 4). This calibration alternative or validation requirement, or both, should be entered into the ASTM standards. It is proposed to use the term "validation," as distinct from "calibration," when multiple-sensors, cross sections, or the entire adjustment procedures are subject to a benchmark neutron field check.

6.5 Standards should use consistent terminology and units for neutron field quantities and nuclear parameters. Reference is made to Terminology C859 regarding nuclear materials, Terminology E170 regarding radiation measurements and dosimetry, and the SI Brochure on the use of SI units.

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(2) Steele, L., "Review of the IAEA Specialists's Meeting on Irradiation Embrittlement, Thermal Annealing and Surveillance of Reactor Pressure Vessels," *Proceedings of the 3rd ASTM-Euratom International Symposium on Reactor Dosimetry*, Ispra, Italy, Oct. 1-5, 1979 and *Proceedings of the IAEA Specialist's Meeting*, Vienna, Austria, Feb. 26-March 1, 1979, IWG-RRPC-79/2, December 1979.

(3) Till, H., "Neutron Radiometric and Calculation Benchmarking for LWR Pressure Vessel Radiation Effects," *Proceedings of the 3rd ASTM-Euratom International Symposium on Reactor Dosimetry*, Ispra, Italy, Oct. 1-5, 1979.

(4) Fabry, A., and Kam, F. B. K., "Towards Adequate Evaluation of LWR Pressure Vessel Steel Irradiation Exposures," Proceedings of an IAEA Specialist's Meeting on "Accuracies in Correlation Between Property Change and Exposure Data from Reactor Pressure Vessel Steel Irradiations," Jülich, West Germany, Sept. 24-27, 1979.

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(6) Stahlkopf, K. E., and Marston, T. U., "A Comprehensive Approach to Radiation Embrittlement Analysis," in Proceedings of an IAEA Specialist's Meeting on "Irradiation Embrittlement, Thermal Annealing and Surveillance of Reactor Vessel," Vienna, Austria, Feb. 26-March 1, 1979.

(7) Randall, P. N., "Regulatory Aspects of Radiation Embrittlement of Reactor Vessel Steels," Proceedings of an IAEA Specialist's Meeting on Irradiation Embrittlement, Thermal Annealing, and Surveillance of Reactor Pressure Vessels," Vienna, Austria, Feb. 26-March 1, 1979.

(8) Serpan, C. Z., "Standardization of Dosimetry Related Procedures for the Prediction and Verification of Changes in LWR-PV Steel Fracture Toughness During a Reactor's Service Life: Status and Recommendations," *Proceedings of the 3rd ASTM-Euratom International Symposium on Reactor Dosimetry*, Ispra, Italy, Oct. 1–5, 1979.

(9) McElroy, W. N., et al., *LWR Pressure Vessel Surveillance Dosimetry Improvement Program: 1980 Annual Report*, NUREG/CR-1747, HEDl-TME 80-73, Hanford Engineering Development Laboratory, Richland, WA, April 1981.

(10) Alberman, A., et al., "Influence des Neutrons Thermiques sur la Fragilisation de l'Acier de Peau d'Etancheite des Reacteurs a Haute Temperature (H.T.R.)," *Proceedings of the Fourth ASTM-Euratom Symposium on Reactor Dosimetry, NUREG/CR-0029, Nuclear Regulatory Commission*, Washington, DC, July 1982, p. 839.

(11) McElroy, W. N., et al., "Surveillance Dosimetry of Operating Power Plants," *Proceedings of the Fourth ASTM-Euratom Symposium on Reactor Dosimetry, NUREG/CP-0029*, July 1982, p. 3, and in *LWR-PV SDIP 1981 Annual Report*, HEDL-SA-2546, Nuclear Regulatory Commission, Washington, DC, 1982.

(12) Hedgecock, P. D., and Perrin, J. S., "Standards for Materials Behavior Under Neutron Irradiation," *Proceedings of the Fourth ASTM-Euratom Symposium on Reactor Dosimetry*, NUREG/CP-0029, Nuclear Regulatory Commission, Washington, DC, July 1982, p. 829.

(13) Dircks, W. J., "Pressurized Thermal Shock (PTS)," NRC Report, SECY-82-465, Nov. 23, 1982 and Enclosure A, NRC Staff Evaluation of PTS, November 1982.

(14) *Standard Review Plan and Branch Technical Position MTEB 5.2: Fracture Toughness Requirements*, NUREG-75/087, Nuclear Regulatory Commission, Washington, DC, 1981.

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