



Standard Master Matrix for Light-Water Reactor Pressure Vessel Surveillance Standards, E 706(0)¹

This standard is issued under the fixed designation E 706; 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-Fig. 2). Some of these are existing ASTM standards, some are ASTM standards that have been modified, and some are proposed ASTM standards. General requirements of content and consistency are discussed in Section 6. More detailed writers' and users' information, justification, and specific requirements for the nine practices, ten guides, and three methods are provided in Sections 3-5. Referenced documents are discussed in Section 2. The summary-type 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 documents, Section 2 and references for use by individual writers and users.

1.2 This master matrix is intended as a reference and guide to the preparation, revision, and use of standards in the series and for planning and scheduling purposes. This index is to ensure the accomplishment of an objective irrespective of the time required, the number of ASTM committees concerned, or the complexity of the issues involved.

1.3 This master matrix standard provides a guide to ASTM standards related to the energy-critical areas that are to be developed under the category of Fission Reactor Development, Section 10, of Guide E 584-77 and as discussed in Practice E 583-97.

1.4 To account for neutron radiation damage in setting pressure-temperature limits and making fracture analyses (see Refs 1-5, 7-12, 52, 58-60, 66, 67 and Recommended Guide E 509), 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, 3, 8-11, 17, 18 and 19, 22-29, 32, 33, 38-46, 60, 66, 67, 72, and Recommended Guide E 509). 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 this master matrix (1, 24). 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 edge 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 spectral effects, and

1.4.10 Neutron dose rate effects.

1.5 A number of potential methods and standards exist for ensuring the adequacy of fracture control of reactor pressure vessel belt lines under normal and accident loads (1, 11, 12, 17-20, 21-24, 41-46, 60, 66, 67, 80, 82, Recommended Guide E 509, and 2.3 ASME Standards). 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, better procedures to evaluate and use this information can and must be developed (1-3, 5, 7-13, , 41-46, 58, 60-62, 66, 67, and Recommended Guide E 509). This master matrix, therefore, defines the current (1) scope, (2) areas of application, and (3) general grouping for the series of 22 ASTM standards, as shown in Figs. 1-2.

1.6 The values stated in SI units are to be regarded as the standard.

1.7 *This standard may involve hazardous materials, operations, and equipment. This standard does not purport to address all of the safety concerns, if any, associated with its*

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use. It is the 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:

- C 859 Terminology Relating to Nuclear Materials²
- E 170 Terminology Relating to Radiation Measurements and Dosimetry³
- E 184 Practice for Effects of High-Energy Neutron Radiation on the Mechanical Properties of Metallic Materials, E 706 (IB)^{3,4}
- E 185 Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels, E 706 (IF)^{3,4}
- E 482 Guide for Application of Neutron Transport Methods for Reactor Vessel Surveillance, E 706 (IID)^{3,4}
- E 509 Guide for In-Service Annealing of Light-Water Cooled Nuclear Reactor Vessels³
- E 560 Practice for Extrapolating Reactor Vessel Surveillance Dosimetry Results, E 706 (IC)^{3,4}
- E 583 Practice for Systematizing the Development of (ASTM) Voluntary Consensus Standards for the Solution

of Nuclear and Other Complex Problems⁵

- E 584 Guide for Developing the (ASTM) Voluntary Consensus Standards Required to Help Implement the National Energy Plan⁵
- E 636 Guide for Conducting Supplemental Surveillance Tests for Nuclear Power Reactor Vessels, E 706 (IH)^{3,4}
- E 646 Test Method for Tensile Strain-Hardening Exponents (n-Values) of Metallic Sheet Materials⁶
- E 693 Practice for Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacements Per Atom (dpa), E 706 (ID)^{3,4}
- E 844 Guide for Sensor Set Design and Irradiation for Reactor Surveillance, E 706 (IIC)^{3,4}
- E 853 Practice for Analysis and Interpretation of Light-Water Reactor Surveillance Results, E 706 (IA)^{3,4}
- E 854 Test Method for Application and Analysis of Solid State Track Recorder (SSTR) Monitors for Reactor Surveillance, E 706 (IIIB)^{3,4}
- E 900 Guide for Predicting Neutron Radiation Damage to Reactor Vessel Materials, E 706 (IIF)^{3,4}
- E 910 Test Method for Application and Analysis of Helium Accumulation Fluence Monitors for Reactor Vessel Surveillance, E 706 (IIIC)^{3,4}
- E 944 Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance, E 706 (IIA)^{3,4}

² Annual Book of ASTM Standards, Vol 12.01.

³ Annual Book of ASTM Standards, Vol 12.02.

⁴ The reference Master Matrix designation in parentheses refers to Section 5 items.

⁵ Discontinued, See 1994 Annual Book of ASTM Standards, Volume 12.02.

⁶ Annual Book of ASTM Standards, Vol 03.01.

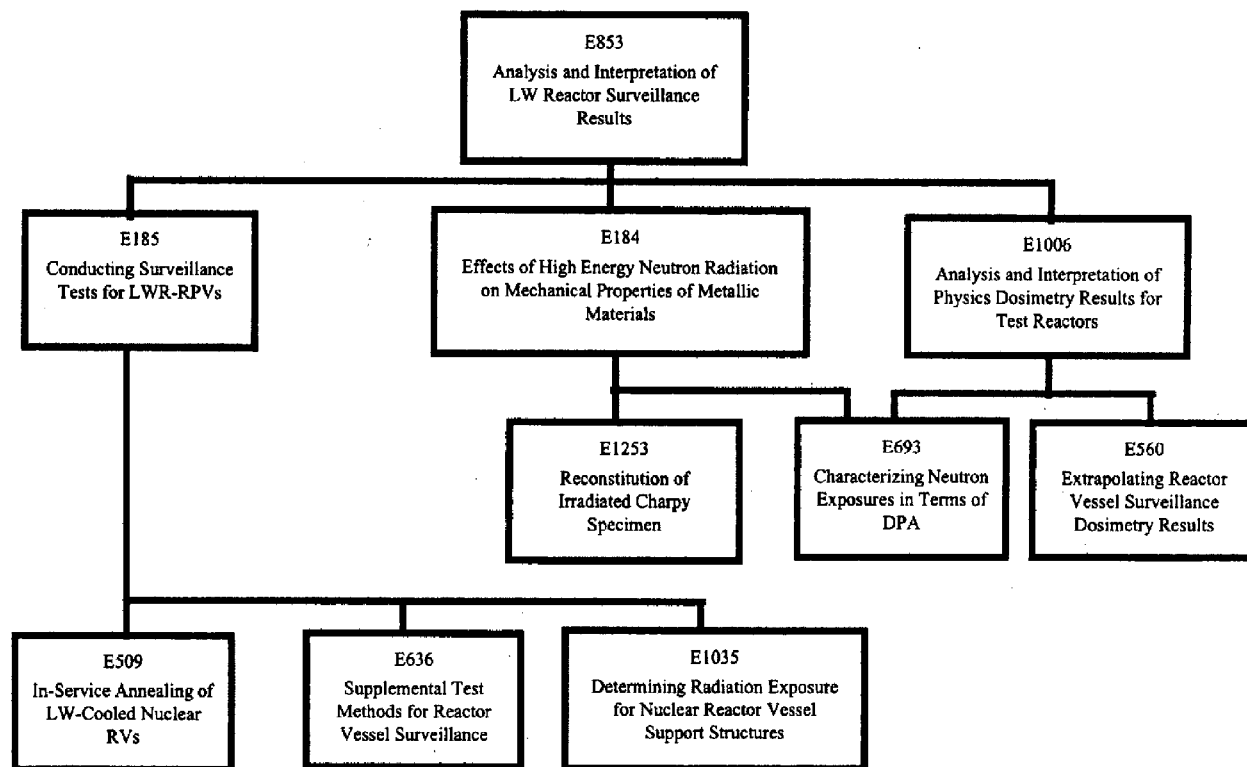


FIG. 1 Surveillance and Correlation Standards

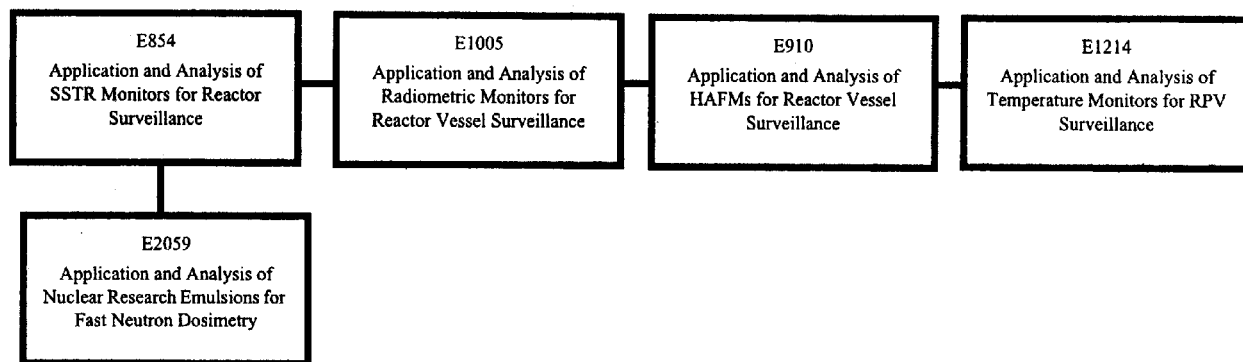


FIG. 2 Sensor Measurement Standards

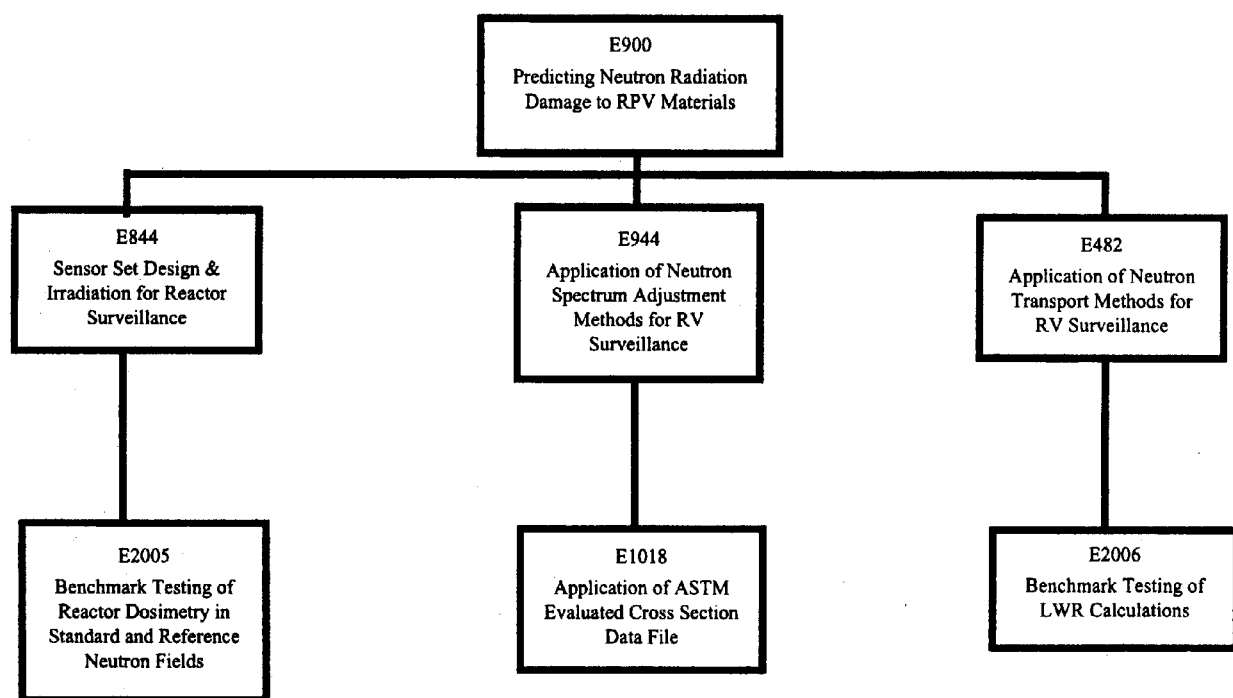


FIG. 3 Supporting Methodology Standards

E 1005 Test Method for Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance, E 706 (IIIA)^{3,4}

E 1006 Practice for Analysis and Interpretation of Physics Dosimetry Results for Test Reactors, E 706 (II)^{3,4}

E 1018 Guide for Application of ASTM Evaluated Cross Section Data File E 706 (IIB)³

E 1035 Practice for Determining Radiation Exposures for Nuclear Reactor Vessel Support Structures³

E 1214 Guide for Use of Melt Wire Temperature Monitors for Reactor Vessel Surveillance, E 706 (IIIE)³

E 1253 Guide for Reconstitution of Irradiated Charpy Specimens^{3,4}

E 2005 Guide for Benchmark Testing of Reactor Dosimetry in Standard and Reference Neutron Fields^{3,4}

E 2006 Guide for Benchmark Testing of Light Water Reactor Calculations^{3,4}

E 2059 Practice for Application and Analysis of Nuclear Research Emulsions for Fast Neutron Dosimetry^{3,4}

SI 10 Standard for the Use of the International System of Units (SI): The Modern Metric System⁷

2.2 Nuclear Regulatory Documents:

Code of Federal Regulations, Chapter 10, Part 50, Appendixes G and H⁸

Code of Federal Regulations, "Reporting of Defects and Noncompliance"⁸

⁷ Discontinued, see 1997 Annual Book of ASTM Standards, Vol 14.04.

⁸ Available from Superintendent of Documents, U. S. Government Printing Office, Washington, DC 20402.

1.99 Regulatory Guide⁸

1.150 Regulatory Guide⁸

2.3 *American Society of Mechanical Engineers Standard: Boiler and Pressure Vessel Code, Sections III and XI*⁹

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

3.1 Aging light water reactor pressure vessels (LWR-PV) are accumulating significant neutron fluence exposures, with consequent changes in their state of steel embrittlement. The U.S. Nuclear Regulatory Commission estimates that there are a few operating U.S. PWRs that will have beltline materials with marginal toughness, relative to the existing requirements of Appendixes G and H of 10 CFR Part 50 and Regulatory Guide 1.99 sometime within their service life (17). 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 (23, 24), international multilaboratory work directed towards the improvement of LWR-PV surveillance has been conducted (1, 2, 3, 5, 34, 35, 41, , and 92-96). The primary concern here is to improve and standardize surveillance tests, neutron dosimetry, damage correlation, and the associated reactor analysis procedures and data used for predicting the integrated effect of neutron exposure to LWR pressure vessels and support structures (1).

3.2 Objectives of the international multilaboratory work are (1) to establish updated and improved surveillance tests, neutron dosimetry, damage correlation, and the associated reactor analysis procedures and data in ASTM standards for LWR-PV surveillance programs, and (2) to perform supporting validation and calibration experiments in benchmark neutron fields, reactor test regions, and operating power reactor surveillance positions. The goal of this activity is to establish consistent and accurate procedures and data as well as to guide the acquisition, reporting, and documentation of the required neutron field characterization information that is used to correlate irradiation effects information and predict end-of-life (EOL) changes in PV steels and support structures.

3.3 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 of the pressure vessel. Measurements of neutron flux, fluence, and spectrum for this characterization are associated with two distinct components of LWR-PV irradiation 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 flux positions near the pressure vessel in operating power reactions (see Sections 4 and 5).

3.4 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 flux and fluence detection techniques, which are continually under development, and widely applied by laboratories with different levels of capability (1, 5, 15, 16, 34-38, . The second surveillance procedure component requires a serious extrapolation of neutron-induced mechanical property change data obtained from test reactors and power reactor surveillance positions to locations inside the body of the pressure vessel and inside of ex-vessel support structures (1-5, 8-11, 13, 21, 40, 42, 45, 46, 52, 59, 60, 61, 65, 66, 72-82). The neutron flux at the vessel inner wall is up to one order of magnitude lower than at surveillance specimen positions and up to two orders of magnitude lower than for test reactor positions. At the vessel outer wall, the neutron flux is one order of magnitude or more lower than at the vessel inner wall. Furthermore, the neutron spectrum at, within, and leaving the vessel is substantially altered (see Table 1 and Refs 1, 17, 21, 46, 52, 66, 73-78).

3.5 In order to meet the LWR-PV radiation monitoring requirements, a variety of neutron flux, fluence, and damage detectors are employed, most of which are passive (see Refs. 1, 15, 16, 47, 57, 63-65,). Each detector must be validated for application to the higher flux and harder neutron spectrum of the test reactor test regions and to the lower flux and degraded 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 estimated for PV and support structure steels at EOL.

3.6 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 was recognized early. High [Oak Ridge Research Reactor-Pool Side Facility (ORR-PSF)] and low flux [Pool Critical Assembly (PCA)] versions of pressure vessel mock-ups have been established (1). The French have established another high-flux mock-up in the Melusine reactor (52), Belgium has established a low-flux mock-up identified as the “VENUS” PWR core source and azimuthal lead factor experiments and calculational tests (60, 63), and the British have established a low-flux mock-up identified as the “NESDIP” PWR cavity experiments and calculational tests (64, 72). As specialized benchmarks, these facilities will 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

⁹ Available from American Society of Mechanical Engineers, 345 E. 47th St., New York, NY 10017.

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 E 185 is available and is used. However, updated versions of this practice should include the following: (a) Determination of surveillance capsule spatial flux-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 data should be combined with reactor physics computations to make necessary adjustments for capsule perturbation effects. (b) As appropriate, use of measured/calculated dpa damage for normalization of Charpy to Charpy (and other metallurgical specimen) variations in neutron flux, fluence, 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 ASTM Guide E 482 and Recommended Practice E 560 provide general guidance in this area. However, updated versions of these standards 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 the surveillance program to current and future reactor vessel and support structure conditions. Currently Recommended Practice E 560 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, ASTM Practice E 185 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.

surveillance capsule in- and ex-vessel power reactor positions will be simulated in these mock-up facilities (1, 21, 73-79).

3.7 The necessity for a few selected operating PWR and BWR power reactor benchmark facilities for testing, validation, and calibration of physics computational methods, processing and adjustment codes, nuclear data, and dosimetry techniques was also recognized, (1, 4, 60, 61, 73-79).

3.8 The results of the measurement and calculational strategies outlined here are being made available for use by the nuclear industry as ASTM standards. Federal Regulation 10 CFR 50 already calls for adherence to several ASTM standards that require establishment of a surveillance program for each power reactor and incorporation of flux monitors for post-irradiation neutron field evaluation. As a result of PV pressurized thermal shock (PTS) studies (17, 18, 19, 41-46, 60), some new direction in the requirements for the ASTM LWR Standards can be anticipated. Consequently, revised and new standards in preparation will be carefully 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 to help manage in the development and application of standards 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 Improved and 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 (Recommended Guide E 509).

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.

4.2.3 Existing conservatism or non-conservatism associated with the variables (1.4) related to (1), (2), and (3) must be reduced by improved practices and subsequent documentation and reporting of surveillance program results.

4.2.4 Application of improved practices and more complete documentation and reporting of test and power reactor results is essential to develop improved metallurgical data bases for reference standards, such as Reg. Guide 1.99 and Section III of the ASME Boiler and Pressure Vessel Code, Part NF2121, which requires 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."

4.2.5 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 at the 10 to 30 % (1σ) confidence level, the nuclear industry and licensing and regulatory agencies can continue to establish realistic LWR power plant operating conditions and limits, such as those now defined in Appendixes G and H of 10 CFR Part 50, Reg. Guide 1.99, and the ASME Boiler and Pressure Vessel Code.

4.2.6 The uniform and consistent application of this series of standards will allow the nuclear industry and licensing and regulatory agencies to properly administer their responsibilities in regard to LWR power reactors that may develop materials with marginal toughness relative to existing and future requirements of Appendixes 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, 4, 5, 7, 9, 10, 13, 30, 31, 34-38,)—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 1.4.1-1.4.10 of 1.4 are considered in this section and 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, 13, 30, 31, 60, 61, 62, 65, 66, 83). 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. If the latter had been the case, then, the error effect of the individual detector perturbations might be negligible or, at least, considerably lessened. 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 22 ASTM standards, Fig. 1, Fig. 3 and Fig. 2 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) 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, 28-31, 38, 66, 82-87, 88, 89). 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 (6).

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 is in progress 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 *Surveillance Capsule Dosimetry Detector Analysis and Interpretation*—Significant uncertainties have, in the past, been introduced in the interpretation of dosimetry detectors when the following issues are not taken into consideration. In the absence of considerations of these effects, the combined effect can be worse than the individual uncertainties quoted below.

4.3.2.1 *Result of Neglect of Flux Perturbations*—Changes in exposure values by 10 to 20 % when have historically occurred due to the neglect of flux perturbations when fluence estimates are based on iron monitors alone and by 30 to 50 % for fission monitors alone. Uncorrected combined results have historically caused correlation discrepancies in the 40 to 70 % range (7).

4.3.2.2 *Result of Neglect of Photo-Reactions*—Changes in exposure values by 10 to 50 % can occur for fission and non-fission threshold monitors (4) when this correction is ignored.

4.3.2.3 *Result of Neglect of Burn-In of Fissile Products for Threshold Fission Reactions*—Changes in exposure values by large percentages (>10 %), depending on fluence, have historically occurred due to the neglect of burn-in corrections.

4.3.2.4 *Result of Neglect of Difference Between Surveillance Capsule Flux and Reactor Power Time Histories*—Changes in exposure values by 10 to 40 %, depending on fuel loadings and reactor operations, have historically occurred when care is not taken to correctly treat the reactor time history.

4.3.3 *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 these effects (4.3.2); 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, 5, 7, 80). 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, 4, 5, 13, 21, 30, 31, 47-49,).

4.3.4 Additional Validation Work for Dosimetry Detectors:

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

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

4.3.4.3 Interlaboratory comparisons using standard and reference neutron fields and other test reactors that provide adequate validations and calibrations, see Guide E 2005.

4.3.5 *Impact of Validation Work Associated with 4.3.4*—Elimination of utilities’ vendor/service laboratories’ “laboratory-to-laboratory” biases of 10 to 20 % or more for individual and combined dosimetry detector results, including corrections for photo-reactions, fission yields, burn-in, etc.

4.4 *Reactor Physics Analysis and Interpretation* (1, 4, 5, 7, 13, 21, 30, 31, 60-66, 73-79)—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, 4, 5, 21, 30, 31, 35-38, 60-66, 89).

4.4.1.2 There are at least three ways in which calculated in- and ex-vessel neutron and gamma ray fields can be normalized for power reactors to (5):

- the core power history.
- surveillance dosimetry results.
- measurements in the cavity behind the vessel and its thermal insulator.

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 ($\frac{1}{4} T$) vessel wall estimates of damage exposure are not currently compared with experimental results. Current results may be uncertain by at least 50 % (1σ) or more. (The IAEA calculational intercomparison of PWR PV damage rate calculations: “Reactor Shielding Benchmark No. 2,” shows factors of 2 or more differences) (5). Further, current “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, 13, 21, 30, 31, 35-38, 60-66, 89).

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, 5, 60, 61, 63-65, 71, 72).

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, 4, 5, 13, 21, 30, 31, 60-62, 73-79).

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 are being established for this purpose are identified and discussed in Refs (1, 4, 5, 60, 61, 73-79) and their references.

4.4.5 *Operating Power Reactor Tests:*

4.4.5.1 This is both a necessary and sufficient 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, 4, 5, 13, 21, 30, 31, 60-62, 73-79).

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, depending on the number of plant years of experience.

4.5 *Metallurgical Damage Correlation Analysis and Interpretation* (1-5, 7-13, 17, 20-34, 35-46, 50-53, 59, 60, 66,)—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 ($\Delta NDTT$, 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, 5, 7, 15, 16, 47-57, 65, 66, 70).

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, 4, 5, 7, 8, 13, 15, 16, 47-57,):

- needed for validation of data correlation procedures and time and space extrapolations (to PV positions: surface, $\frac{1}{4}$ 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 Surveillance Capsule Dosimetry and Metallurgical Data Analysis and Interpretation:

4.5.2.1 *Result of Neglect of Detector Effects* (4.3.2)—Changes of 50 to 100 % for estimates of changes in PV steel fracture toughness. (This does not include exposure unit difference, flux spectrum, or flux level effects; assumes all correlations are based on fluence >1 MeV.)

4.5.2.2 *Result of Neglect of Flux Spectrum and Exposure Units*—Using fluence >1 MeV, changes of 20 to 60 % for $\frac{1}{4}$ T to $\frac{3}{4}$ T positions and about 70 % for ex-vessel positions depending on the reactor configuration (this assumes that dpa is a reasonable measure of actual residual damage) (1, 21, 30, 31, 36-40, 52, 59, 60, 73-79).

4.5.2.3 *Result of Neglect of Flux Level*—Potentially large changes (1, 8-10, 30, 31, 66, 86). Existence of effect has yet to be separately demonstrated; it must be shown not to be associated with uncertainties and errors in other variables.

4.5.2.4 *Result of Neglect of Temperature*—Changes of up to 40 % or more depending on melt wire results, thermal hydraulics, and gamma heating.

4.5.2.5 *Results of Neglect of Chemistry*—Changes of up to a factor of 3 for copper and up to 100 % for other elements, depending on the steel.

4.5.2.6 *Results of Neglect of Microstructure*—Changes up to 50 to 100 %, depending on the steel.

4.5.3 Impact of Neglect of Combined Effects of 4.5.2:

4.5.3.1 Changes up to a factor of 3 for prediction of changes in PV steel fracture toughness, depending on the reactor configuration and normalization procedures (4.4.1.2).

4.5.3.2 Completely unacceptable accuracy (a swing of a factor of 2 or more) for establishing data base reference standards such as Reg. Guide 1.99.

4.5.4 *Benchmark Field Referencing*—The PSF (all positions: surveillance, surface, $\frac{1}{4}$ T, $\frac{1}{2}$ 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, 5, 32, 36-40, 43, 52, 59). Other test reactor data comes from surveillance capsule results that have been benchmarked by vendor/service laboratory/utility groups (1, 4, 5, 8, 13, 20, 25-31, 66, 80, 82-86).

4.5.5 *Reg. Guide 1.99, MPC, NRC, EPRI Data Bases*—Materials Property Council (MPC), NRC and Electric Power Research Institute (EPRI) data bases will be studied by ASTM Subcommittees E10.02 and E10.05, MPC, vendors, utilities, EPRI, and NRC contractors in an ongoing basis 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 MPC, NRC, and EPRI data bases. (1-3, 8-11, 17, 20, 25-31, 80, 82-86) and incorporate these recommendations in the appropriate standards. 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 E 1018, and incorporating corrections for all known variables (perturbations, photo-reactions, spectrum, burn-in, yields, flux 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 methods constitutes the master matrix, describes the scope of individual standards, and provides other relevant information for the series of 22 LWR-PV surveillance standards.¹⁰ Fig. 1, Fig. 3 and Fig. 2 indicate by title and ASTM designation the elements of this matrix standard and show the general grouping for this series of ASTM standards.¹¹

5.2 Surveillance and Correlation Standards:

5.2.1 *Analysis and Interpretation of Nuclear Reactor Surveillance Results-E 706 (IA), E 853 (E10.05)*:¹²

5.2.1.1 *Scope*—Practice E 853 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 methods that are in various stages of development and revision (see Fig. 1, Fig. 3 and Fig. 2). 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.

5.2.1.2 *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. 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, then, can be used with E 706 (IC), Surveillance Test Results Extrapolation, for helping to

¹⁰ For standards that are in the draft stage and have not been assigned an ASTM number, the Master Practice, I-A, will be very explicit and provide necessary detailed information on the procedures and data that are expected to be recommended in unnumbered reference standards.

¹¹ Cross referencing of these standards is to be done by means of the designations given in Fig. 1, Fig. 3 and Fig. 2. Therefore, the Analysis and Interpretation of Nuclear Reactor Surveillance Results Practice should be referred to as E706 (I A).

¹² Indicates the ASTM E10 subcommittee that has the primary responsibility for the preparation of the standard.

predict the future condition of the pressure vessel. These same procedures, in conjunction with the use of IG, Determining Radiation Exposure for Nuclear Reactor Support Structures, 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: E 706(II), E 706(IC), and E 706(IG).

5.2.2 Physics-Dosimetry-Metallurgy Interface Standard for LWR, FBR, and MFR Programs—E 706 (IB), E 184 (E10.02):

5.2.2.1 Scope—Practice E 184 serves as an interface standard for other physics-dosimetry-metallurgy standards and covers special procedures for selecting a material and tests for radiation studies, determining radiation conditions, conducting tests for the mechanical properties of irradiated metallic materials, and reporting of data.

5.2.2.2 Discussion—The purpose of this practice and the supporting standards is to achieve better correlation and interpretation of new and old data in the field of radiation effects testing and provide an interface between the mechanical property and physics-dosimetry standards in the Master Matrix E 706. Although the Master Matrix E 706 standards were originally developed for application to LWR-PV surveillance, they are also useful for FBR and MFR applications. While primarily intended to apply to ferrite and austenitic steels, this practice may be applied to any appropriate metallic materials. In addition to tests for mechanical properties, these procedures should be helpful in planning other types of radiation effects testing. It should be recognized that the effect of high-energy neutron radiation on the mechanical properties of the materials being studied is determined by the change in mechanical properties during radiation. Therefore, the post-irradiated mechanical properties must be compared with the pre-irradiation properties of the materials.

5.2.3 Extrapolating Reactor Vessel Surveillance Dosimetry Results—E 706 (IC), E 560 (E10.05):

5.2.3.1 Scope—Practice E 560 is intended to be used to relate neutron flux density derived from monitors in surveillance capsules to the corresponding flux density at the inner diameter and within the pressure vessel wall of a light water-moderated reactor. It describes analytical and analytical-experimental approaches that can be used to determine the variation in neutron flux density and energy spectrum between a surveillance capsule location and the pressure vessel wall. Procedures for reporting the results of these analyses are also suggested. It does not cover the relationship of the neutron damage observed at the surveillance location to that occurring within the pressure vessel wall. Rather, it deals solely with the neutron flux density monitoring aspects of the surveillance program.

5.2.3.2 Discussion—Practice E 560 should be revised to include the specification of procedures for the extrapolation and interpolation of surveillance data from surveillance capsule locations to and within the pressure vessel wall and to

ex-vessel positions (1-5, 17, 20 and 21, 41-46, 60-66, 73-80, 82-86, 88, 89). Areas discussed within the existing scope include (1) surveillance capsule perturbation effects, (2) use of benchmark fields and benchmarked calculations such as those for the PCA, VENUS, and NESDIP mock-ups, use of physics calculations to provide dpa calculations at surveillance positions and within the vessel wall and support structures, (4) the effect of uncertainties and errors in reactor dimensions, and (5) the effect of reactor power cycle-to-cycle shifts in core fuel loadings. Recommended reactor physics procedures and data that support the extrapolation of the surveillance data are addressed in the Guide E 482, E 706 (IID). Practice E 1035 currently provides guidance for determining the radiation exposure to reactor vessel support structures, but the guidance provided there should be more detailed and more closely linked to E 560 with regard to procedures, benchmarking, and validation.

5.2.4 Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacements Per Atom—E 706 (ID), E 693 (E10.05):

5.2.4.1 Scope—Practice E 693 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-flux spectrum and the availability of a cross section file, and is discussed in 5.3.2.

5.2.4.2 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 (1-5, 17, 21, 23-31, 35, 52, 59,). 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 (89, 91). 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 current dpa cross section is based on the ENDF/B-IV library. Since the iron cross sections have been extensively revised in the recent ENDF/B-VI cross section, plans are underway to upgrade the iron dpa cross section to be consistent with the current recommendations for transport cross sections. Preliminary work suggests

that this planned revision of the energy-dependence of the dpa exposure metric will not significantly change the relative dpa between LWR pressure vessel surveillance capsules and benchmark fields such as PCA and NESDIP.

5.2.5 *Surveillance Tests for Nuclear Reactor Vessels—E 706 (IF), E 185 (E10.02):*

5.2.5.1 *Scope*—Practice E 185 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 E 706 (IF) practice was developed for all light-water cooled nuclear power reactor vessels for which the predicted maximum neutron fluence ($E > 1 \text{ MeV}$) at the end of the design lifetime exceeds $1 \times 10^{21} \text{ n/m}^2$ ($1 \times 10^{17} \text{ n/cm}^2$) at the inside surface of the reactor vessel.

5.2.5.2 *Discussion*—Predictions of neutron radiation effects on pressure vessel steels are considered in the design of light-water 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.

5.2.6 *Determining Radiation Exposures for Nuclear Reactor Support Structures—E 706 (IG), E 1035 (E10.05):*

5.2.6.1 *Scope*—Practice E 1035 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 neutron fluence ($E > 1 \text{ MeV}$) that exceeds 1×10^{17} neutron per square centimetre or 3×10^{-4} displacements per atom. Its interrelationship to other Master Matrix E 706 standards is shown in Fig. 1, Fig. 3 and Fig. 2.

5.2.6.2 *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 E 706. Reactor vessel support structures, depending on their location,

may also experience neutron irradiation effects (1, 24, 59, 80). 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 (23).

5.2.7 *Supplemental Test Methods for Nuclear Reactor Vessel Surveillance—E 706 (IH), E 636 (E10.02):*

5.2.7.1 *Scope*—Practice E 636 covers test methods and procedures that can be used in conjunction with, but not as alternatives to, those required by Practice E 185 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 postirradiation 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.

5.2.7.2 *Discussion*—Practice E 185 describes a minimum program for the surveillance of reactor vessel mechanical property changes in service for the case where monitoring is required. Practice E 636 may be applied where radiation 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.

5.2.8 *Practice for Analysis and Interpretation of Physics Dosimetry for Test Reactors—E 706 (II), E 1006 (E10.05):*

5.2.8.1 *Scope*—Practice E 1006 describes the methodology used in the analysis and interpretation of physics-dosimetry results from test reactors (1, 2, 3, 35-39, 52, 59, 66, 72, 80, 81,). 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 E 853 (E 706-IA); Practice E 560 (E 706-IC); Practice E 185 (E 706-IF), Matrix E 706-IG, Guide E 900 (E 706-IIF); and Test Method E 646. Other applicable ASTM metallurgical test references may be found in References.

5.2.8.2 *Discussion*—This new E 706(II) practice presents the best currently available methods for the determination of damage related fluences received by metallurgical specimen

from irradiation experiments in test reactors. Application of this practice will provide reliable and uniform input data from data bases pertaining to radiation damage of reactor materials. Procedures for the use of test reactor data in embrittlement prediction for reactor pressure vessels are addressed in Practice E 560.

5.2.9 In-Service Annealing of Light-Water-Cooled Nuclear Reactor Vessels-E 706 (IJ), E 509 (E10.02):

5.2.9.1 Scope and Discussion—Guide E 509 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.

5.2.10 Guide for Reconstitution of Irradiated Charpy Specimens-E 706 (IK), (E10.02):

5.2.10.1 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 E 1253 describes the procedures for the reconstitution of Methods 23 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.

5.3 Supporting Methodology Standards:

5.3.1 Application of Neutron Spectrum Adjustment Methods-E 706 (IIA), E 944 (E10.05):

5.3.1.1 Scope—Practice E 944 describes the procedures and codes recommended for use for the determination of neutron flux-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 trial spectra (from physics calculations) are available. This guide will address the uncertainties and errors associated with derived integral neutron field characterization and exposure parameters (total and thermal flux-fluence, flux-fluence >0.1 and >1.0 MeV and dpa).

5.3.1.2 Discussion—The use of test reactor and power reactor surveillance results for the prediction of end of life (EOL) pressure vessel and support structures steel changes in fracture toughness requires the measurement and determination of neutron flux-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 flux-fluence, flux-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). (See Refs **1, 14, 30, 31, 35-38, 65, 66, 88 and 89**).

5.3.2 Application of ASTM Evaluated Cross Section Data File-E 706 (IIB), E 1018 (E10.05):

5.3.2.1 Scope—Guide E 1018 covers the establishment and use of an ASTM cross section and uncertainty/error tape 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 from the National Nuclear Data Center (NNDC), Brookhaven National Laboratory.

5.3.2.2 Discussion—This new E 1018 ASTM guide is directly related to and should be used in conjunction with Guide E 944 (IIA). The ASTM cross section file represents a generally available data set for use in sensor set analysis (**14, 88 and 89**). 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 more recent least squares adjustment methods applied to the determination of flux-fluence spectra (see the E 706 (IIA) Guide).

5.3.3 Sensor Set Design and Irradiation for Reactor Surveillance-E 706 (IIC), E 844 (E10.05):

5.3.3.1 Scope—Guide E 844 covers the selection, design, irradiation, and post-irradiation handling of radiometric monitors (RM), solid state track recorders (SSTR), helium accumulation fluence monitors (HAFM), damage monitors (DM), 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.

5.3.3.2 Discussion—Characterization of the irradiation (neutron and gamma) and thermal (temperature) environments for LWR test reactor and operating BWR and PWR power plants is essential for the proper analysis, interpretation, and application of research and power reactor test and surveillance program results. Reactor physics (neutron and gamma) and thermal hydraulics (neutron and gamma heating) computations must be validated and calibrated by active and passive sensor and sensor set measurements (**1-5, 7-10, 13, 15, 16, 30, 31,**

34-38, 60-66, 88, 89). Guide E 844 is used, therefore, as a reference in other ASTM standards, when passive sensor and sensor set measurements are recommended for LWR test and power reactor environmental characterization measurements.

5.3.4 Application of Neutron Transport Methods for Reactor Vessel Surveillance-E 706 (IID), E 482 (E10.05):

5.3.4.1 Scope—The revised Guide E 482 describes the methodology for performing radiation transport calculations to determine the neutron and gamma flux 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. Consequently, the procedures will be developed to be applicable for design and operation calculations and for the evaluation and extrapolation of research and surveillance program measurement results obtained in operating reactors. The accuracy of reactor physics calculations will be considered together with benchmarking procedures for validating and calibrating the results of computations, see 4.4, **(1, 13, 21, 30, 31, 34-38, 52, 60-66, 68, 69,)**.

5.3.4.2 Discussion—This E 706 (IID) ASTM guide addresses the radiation transport methods, data, results, and the uncertainties and errors associated with LWR calculated integral radiation field characterization and exposure parameters such as total neutron flux-fluence >0.1 and >1.0 MeV and dpa. 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. The scope and content of the current guide reflects the fact that discrete ordinate radiation transport techniques are the standard calculational method. In future revisions, better guidance will be provided to the user on when and why to use 1D, 2D synthesis, full 2D, and 3D calculations. When 3D calculations are desired, users often turn to Monte Carlo transport techniques. As the use of Monte Carlo techniques becomes more common place for research reactors, this guide should be expanded to address the use of Monte Carlo techniques.

5.3.5 Benchmark Testing of Reactor Dosimetry in Standard and Reference Neutron Fields-E 706 (IIE-1), E 2005 (E10.05):

5.3.5.1 Scope—This Guide E 2005 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.

5.3.5.2 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.

5.3.6 Benchmark Testing of Light Water Reactor Calculations-E 706 (IIE-2), E 2006 (E10.05):

5.3.6.1 Scope—This Guide E 2006 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.

5.3.6.2 Discussion—This guide deals with the difficult problem of benchmarking neutron transport calculations carried out to determine fluences for plant specific geometries and describes general approaches for using neutron fields with known characteristics to corroborate calculational methodology and nuclear data.

5.3.7 Predicting Neutron Radiation Damage to Reactor Vessel Materials-E 706 (IIF), E 900 (E10.02):

5.3.7.1 Scope—Guide E 900 describes the metallurgical data base, the curve-fitting techniques, and the resulting property change versus exposure curves (currently, only for fluence >1 MeV) 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, weldment, or HAZ material) and material specification (A533B, A302B, or A508), (2) neutron irradiation temperature range—approximately 277 to 310°C (530 to 590°F), (3) neutron exposure units and values—currently restricted to fluences >1 MeV in the 1×10^{16} to 10^{20} n/cm² range. This E 706 (IIF) ASTM guide will rely on the application of several other ASTM standard practices, guides, and methods that are in various stages of development. It is expected that different sets of property change versus exposure curves will be needed based on (1) a total metallurgical data base, (2) a test reactor experimental metallurgical data base, and (3) a power reactor surveillance capsule metallurgical data base. The selection of the appropriate set of curves for a given application will be left to the discretion of the user. Furthermore, for any given BWR or PWR power plant, these curves will be intended for use, primarily, for predictions required before plant specific surveillance data are available.

5.3.7.2 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, 3, 8-12, 17, 18, 19 and 20, 33, 41-46, 60, 67, 80-82, 88, 97)**. To ensure conservative operational margins for nuclear power plants, accepted and accurate interpretations of Charpy V-notch toughness results are therefore necessary **(11, 12, 17, 18, 19 and 20, 41-46, 82, 98)**.

5.4 Sensor Measurement Standards:

5.4.1 Analysis of Radiometric Monitors for Reactor Vessel Surveillance-E 706 (IIIA) (RM), E 1005 (E10.05):

5.4.1.1 Scope and Discussion—Method E 1005 describes the use of Radiometric Monitors (RM) for neutron dosimetry in LWR applications **(1, 30, 31, 34, 36-38, 47-49, 52, 60, 63-66,)**. Measurement procedures for RM sensors by means of gamma ray or X-ray emission detection are specified. Determination of reaction rates including systematic effects such as interference from competing reactions, counting losses, monitor selfshielding branching ratios and fission yields are addressed. The useful flux-fluence range and limitations for the application of RM for LWR reaction and reaction rate measurements are specified. Procedures for the calculation of RM

reactions and reaction rates from radioactivity measurements and the irradiation time history are included. The assessment and discussion of methods and techniques for estimating uncertainties and errors is an important part of this standard (see 6.2-6.4). For example, the standard addresses (1) benchmark calibrations and (2) the use of multiple reaction products (^{140}Ba , ^{103}Ru , ^{95}Zr , ^{137}Cs , etc.) from fission monitors for the assessment of individual sensor time history-corrections. Test Method E 1005 is referenced and used in conjunction with Guide E 844 on Sensor Set Design. It is intended for use for test reactor and power reactor measurements for light-water reactors.

5.4.2 Analysis of Solid State Track Recorder Monitors for Reactor Vessel Surveillance-E 706 (IIIB), E 854 (E10.05):

5.4.2.1 Scope and Discussion—Method E 854 describes the use of Solid State Track Recorders (SSTR) for neutron dosimetry in LWR applications (**1, 15, 54, 55, 65, 66**). Measurement procedures are specified for SSTR sensors by means of optical (and later more advanced) track counting techniques (**15, 55**). The useful flux-fluence range and limitations for the application of SSTR for LWR time-integrated reaction rate measurements are specified. SSTR are truly passive fluence detectors that provide a permanent record for a dosimetry experiment without the need for time-dependent corrections, such as decay factors that arise with RM detectors. The assessment and discussion of methods and techniques for estimating uncertainties and errors is an important part of this standard (see 6.2-6.4). Method E 854 is referenced and used in conjunction with Guide E 844 on Sensor Set Design. It is intended for use for test reactor and power reactor measurements for light-water reactors.

5.4.3 Analysis of Helium Accumulation Fluence Monitors for Reactor Vessel Surveillance-E 706 (IIIC), E 910 (E10.05):

5.4.3.1 Scope and Discussion—Test Method E 910 describes the use of Helium Accumulation Fluence Monitors (HAFM) for neutron dosimetry in LWR applications (**1, 16, 56, 57, 66**). Applications that are covered in this method include analysis and interpretation of helium generation data from HAFM capsules, unencapsulated radiometric monitors, Charpy specimens, and reactor vessel wall samples. Determination of boron levels in Charpy specimens by thermal neutron irradiation followed by helium analysis is another application of interest for defining the effect of low energy neutrons to steel embrittlement (**1, 59**). Measurement procedures are specified for HAFM sensors by means of high-sensitivity helium gas mass spectrometry (**16**). The useful flux-fluence range and limitations for the application of HAFM for LWR time-integrated reaction rate measurements are specified. HAFMs, like SSTR, are truly passive fluence-detectors that provide time-integrated reaction rate measurements without the need for time-dependent corrections, such as decay factors that arise with RM detectors. The assessment and discussion of methods and techniques for estimating uncertainties and errors is an important part of this standard (see 6.2-6.4). Method E 910 is referenced and used in conjunction with Guide E 844. It is intended for use for test reactor and power reactor measurements for light-water reactors.

5.4.4 Use of Melt Wire Temperature Monitors for Reactor Vessel Surveillance-E 706 (IIIE), E 1214 (E10.02):

5.4.4.1 Scope and Discussion—Guide E 1214 describes the application of temperature monitors and their use for reactor vessel surveillance of light-water power reactors as called for in Practice E 185. 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 IIIE method is referenced and used in conjunction with the Guide E 844 and is intended for use for light-water power reactors.

5.4.5 Analysis of Nuclear Research Emulsions for Fast Neutron Dosimetry-E 706 (IIIF), E 2059 (E10.05):

5.4.5.1 Scope and Discussion—Practice E 2059 describes the application of nuclear research emulsions (NRE) for fast neutron dosimetry. NRE are passive detectors which provides neutron detection through neutron scattering on the hydrogen in the emulsion. This practice discusses the limitations of NREs in low power reactor environments that include gamma sensitivity, the low energy limit, and the track density limit. The purpose of this practice is to describe the process of fielding, reading, and interpreting of the NRE.

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—Parallelism in content, overlapping 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.

6.2 Standards should identify and discuss all elements of accuracy (**1, 3-5, 6, 7, 9, 10, 18, 25-31, 34-40, 47-49, 60-66, 88 and 89**). 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 (**3, 6, 28-31, 35, 38, 40, 47-49, 52, 60, 61, 65, 66, 73, , 88 and 89**). 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-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, 6, 17, 30, 31, 38, 40, 60-66, 73-78, 88 and 89).

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 2 references 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, 4, 5, 13, 38, 40, 47-49, 60-66, 71-87, 88 and 89).

6.4 Standards concerned with sensor measurement, data interpretation or transport calculation should refer to bench-

mark 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 C 859 regarding nuclear materials, E 170 regarding radiation measurements and dosimetry, and Standard SI 10 on the use of SI units.

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