Designation: E 706 – 01

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). 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 (**1-106**)[2] 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 **2-7, 9-14, 21-57, 63, 69-71, 77, 78, 83-104,** 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 (**1, 3, 4, 10-13, 17, 21, 22-27, 32-39, 42, 43, 45, 49-57, 71, 77, 78, 83, 91-104,** 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 (**2, 34**). 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 (**1-4, 7, 13, 14, 21-28, 29-34, 52-57, 71, 77, 78, 91, 93,** 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 (**1-4, 6, 7, 9-15, 17, 21-34, 52-57, 69, 71-73, 77, 78, 91-104,** 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. 1-3.

[1] This master matrix 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 the list of references at the end of this standard.

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 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 Materials[3]

E 170 Terminology Relating to Radiation Measurements and Dosimetry[4]

E 184 Practice for Effects of High-Energy Neutron Radiation on the Mechanical Properties of Metallic Materials, E 706 (IB)[4,5]

E 185 Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels, E 706 (IF)[4,5]

E 380 Practice for Use of the International System of Units (SI) (the Modernized Metric System)[6]

E 482 Guide for Application of Neutron Transport Methods for Reactor Vessel Surveillance, E 706 (IID)[4,5]

E 509 Guide for In-Service Annealing of Light-Water Cooled Nuclear Reactor Vessels[4]

E 560 Practice for Extrapolating Reactor Vessel Surveillance Dosimetry Results, E 706 (IC)[4,5]

E 583 Practice for Systematizing the Development of (ASTM) Voluntary Consensus Standards for the Solution of Nuclear and Other Complex Problems[7]

E 584 Guide for Developing the (ASTM) Voluntary Consensus Standards Required to Help Implement the National Energy Plan[7]

E 636 Guide for Conducting Supplemental Surveillance Tests for Nuclear Power Reactor Vessels, E 706 (IH)[4,5]

E 646 Test Method for Tensile Strain-Hardening Exponents (n-Values) of Metallic Sheet Materials[8]

E 693 Practice for Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacements Per Atom (dpa), E 706 (ID)[4,5]

E 844 Guide for Sensor Set Design and Irradiation for Reactor Surveillance, E 706 (IIC)[4,5]

E 853 Practice for Analysis and Interpretation of Light-Water Reactor Surveillance Results, E 706 (IA)[4,5]

E 854 Test Method for Application and Analysis of Solid

[3] *Annual Book of ASTM Standards*, Vol 12.01.

[4] *Annual Book of ASTM Standards*, Vol 12.02.

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

[6] Discontinued, see 1997 *Annual Book of ASTM Standards*, Vol 14.04.

[7] Discontinued, See 1994 *Annual Book of ASTM Standards*, Volume 12.02.

[8] *Annual Book of ASTM Standards*, Vol 03.01.

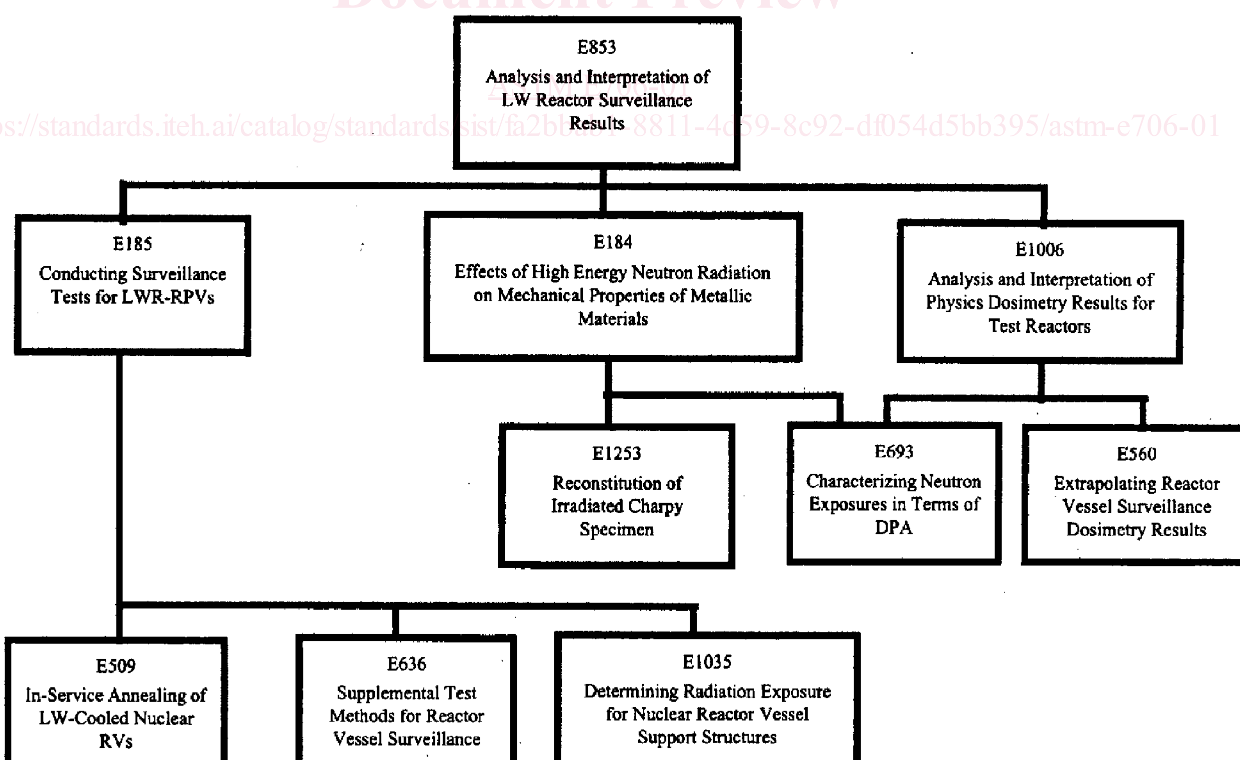

FIG. 1 Surveillance and Correlation Standards

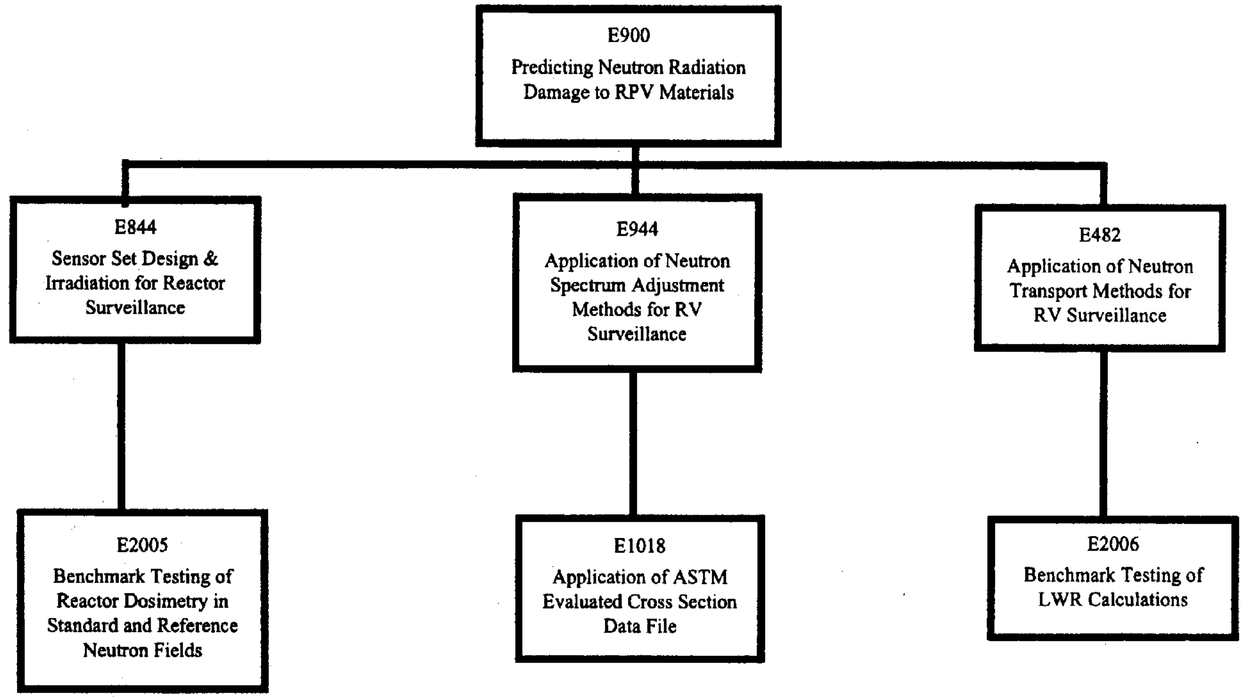

FIG. 2 Supporting Methodology Standards

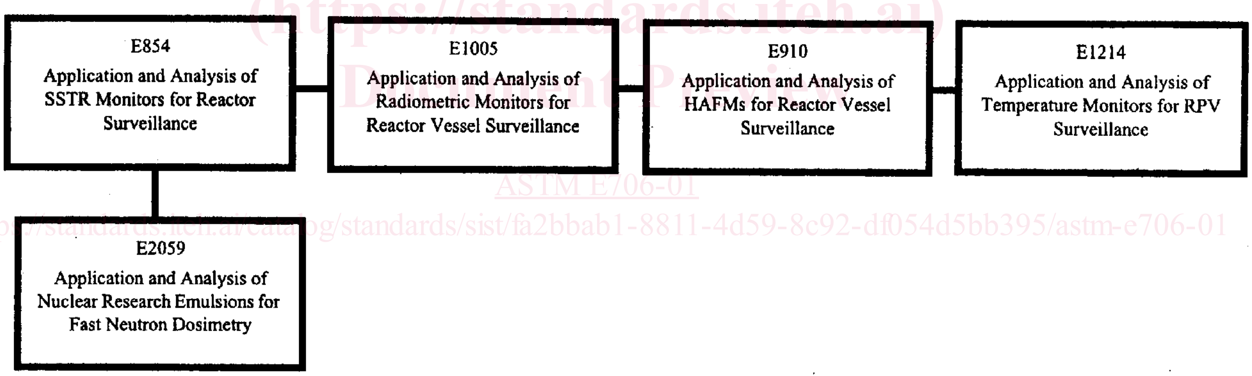

FIG. 3 Sensor Measurement Standards

State Track Recorder (SSTR) Monitors for Reactor Surveillance, E 706 (IIIB)[4,5]

E 900 Guide for Predicting Neutron Radiation Damage to Reactor Vessel Materials, E 706 (IIF)[4,5]

E 910 Test Method for Application and Analysis of Helium Accumulation Fluence Monitors for Reactor Vessel Surveillance, E 706 (IIIC)[4,5]

E 944 Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance, E 706 (IIA)[4,5]

E 1005 Test Method for Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance, E 706 (IIIA)[4,5]

E 1006 Practice for Analysis and Interpretation of Physics Dosimetry Results for Test Reactors, E 706 (II),[4,5]

E 1018 Guide for Application of ASTM Evaluated Cross Section Data File E 706 (IIB)[4]

E 1035 Practice for Determining Radiation Exposures for Nuclear Reactor Vessel Support Structures[4]

E 1214 Guide for Use of Melt Wire Temperature Monitors for Reactor Vessel Surveillance, E 706 (IIIE)[4]

E 1253 Guide for Reconstitution of Irradiated Charpy Specimens[4,5]

E 2005 Guide for Benchmark Testing of Reactor Dosimetry in Standard and Reference Neutron Fields[4,5]

E 2006 Guide for Benchmark Testing of Light Water Reactor Calculations[4,5]

E 2059 Practice for Application and Analysis of Nuclear Research Emulsions for Fast Neutron Dosimetry[4,5]

2.2 *Nuclear Regulatory Documents:*

Code of Federal Regulations, Chapter 10, Part 50, Appendixes G and H[9]

Code of Federal Regulations, "*Reporting of Defects and Noncompliance*"[9]

1.99 Regulatory Guide[9]

1.150 Regulatory Guide[9]

2.3 *American Society of Mechanical Engineers Standard:*

Boiler and Pressure Vessel Code, Sections III and XI[10]

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 **(21)**. 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 **(33, 34)**, international multilaboratory work directed towards the improvement of LWR-PV surveillance has been conducted **(2, 3, 4, 6, 44, 45, 46, 52, 58-104,** and **107-111)**. 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 **(2)**.

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 **(2, 6, 19, 20, 44-49, 58-77, 79-90)**. 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 **(2-6, 10-13, 15, 17, 26, 31, 33-46, 51, 53, 56, 57, 63, 70, 71, 72, 76, 77, 83-104)**. 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 **2, 21, 29-31, 57, 63, 77, 84-89**).

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. **2, 19, 20, 58, 68, 74-76, 79-91**). 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 **(2)**. The French have established another high-flux mock-up in the Melusine reactor **(63)**, Belgium has established a low-flux mock-up identified as the "VENUS" PWR core source and azimuthal lead factor experiments and calculational tests **(71, 74)**, and the British have established a low-flux mock-up identified as the "NESDIP" PWR cavity experiments and calculational tests **(75, 83)**. 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 Superintendent of Documents, U. S. Government Printing Office, Washington, DC 20402.

[10] 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 **(2, 29-31, 84-90)**.

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, **(2, 5, 71, 72, 84-90)**.

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 **(21, 22, 24, 25, 52-57, 71)**, 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 (**2-104** and 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." **(7)**

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* **(2, 5, 6, 9, 11, 12, 15, 40, 41, 44-49, 58-91)**—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 **(9)**, 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 **(2, 15, 29, 30, 40, 41, 71, 72, 73, 76, 77, 79-91, 94)**. 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. 2 and Fig. 3 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) **(2, 27, 38-41, 49, 77, 93-98, 103, 104)**. 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 $\pm$ 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 **(8)**.

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 **(2)**, 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 **(9)**.

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 **(5)** 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 **(2, 6, 9, 91)**. 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 **(2, 5, 6, 15, 29, 31, 40, 41, 45, 58-60, 69-77, 79-90)**.

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 **(102)**.

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.