



Designation: E706 – 23

Standard Master Matrix for Light-Water Reactor Pressure Vessel Surveillance Standards¹

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1. Scope

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

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

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

1.4 The techniques, variables, and uncertainties related to (1) neutron and gamma dosimetry, (2) physics (neutronics and gamma effects), and (3) metallurgical damage correlation procedures and data are addressed in separate standards be-

longing to this master matrix (1, 17). The main variables of concern to (1), (2), and (3) are as follows:

1.4.1 Steel chemical composition and microstructure,

1.4.2 Steel irradiation temperature,

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

1.4.4 Core power distribution,

1.4.5 Reactor operating history,

1.4.6 Reactor physics computations,

1.4.7 Selection of neutron exposure units,

1.4.8 Dosimetry measurements,

1.4.9 Neutron special effects, and

1.4.10 Neutron dose rate effects.

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

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

1.7 *This standard does not purport to address all of the safety concerns, if any, associated with its use. It is the responsibility of the user of this standard to establish appropriate safety, health, and environmental practices and determine the applicability of regulatory limitations prior to use.*

1.8 *This international standard was developed in accordance with internationally recognized principles on standardization established in the Decision on Principles for the Development of International Standards, Guides and Recommendations issued by the World Trade Organization Technical Barriers to Trade (TBT) Committee.*

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 a list of references at the end of this standard.

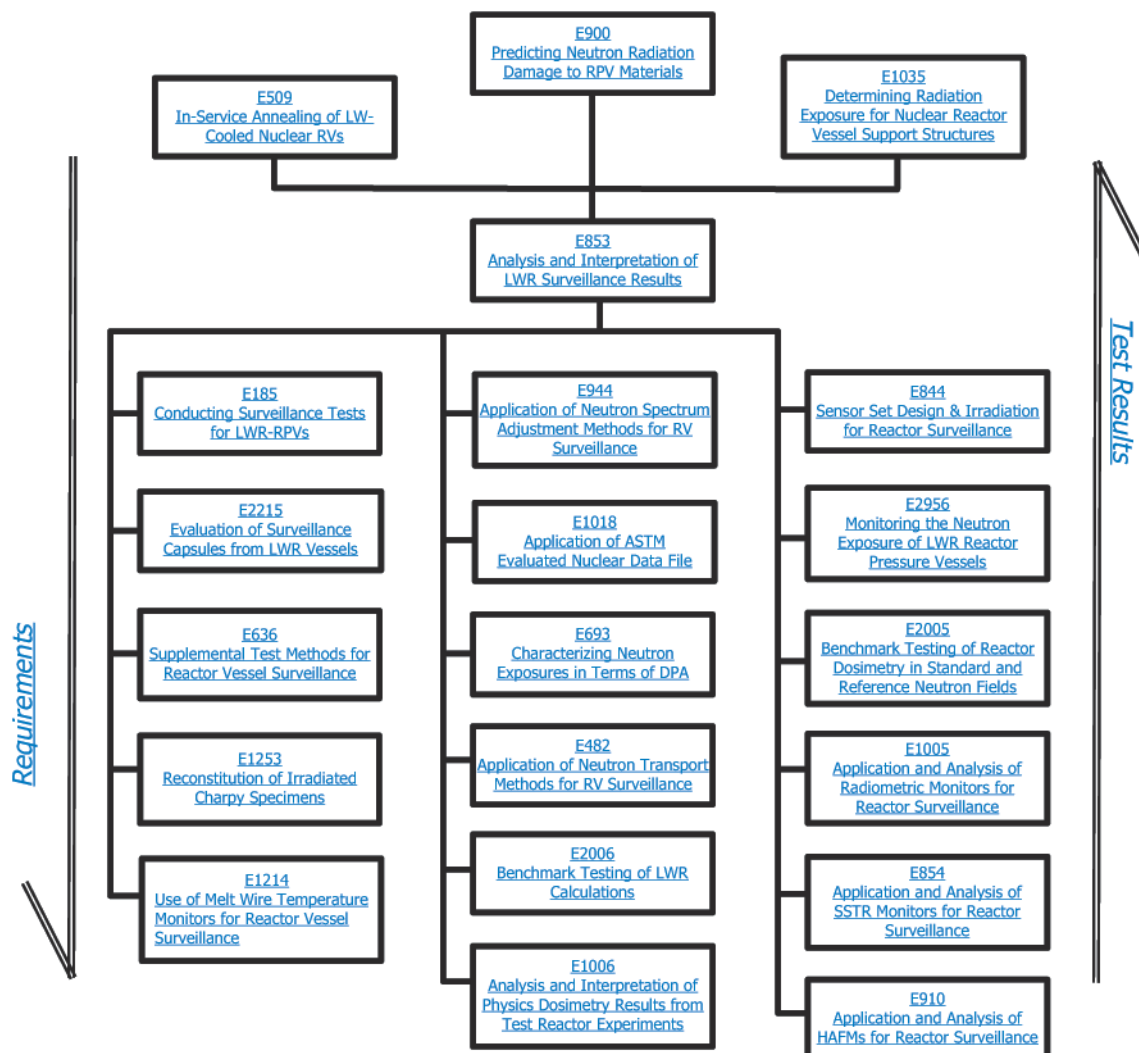


FIG. 1 Organization and Use of ASTM Standards in the E706 Master Matrix

2. Referenced Documents

2.1 ASTM Standards:³

- C859 Terminology Relating to Nuclear Materials
- E23 Test Methods for Notched Bar Impact Testing of Metallic Materials
- E170 Terminology Relating to Radiation Measurements and Dosimetry
- E185 Practice for Design of Surveillance Programs for Light-Water Moderated Nuclear Power Reactor Vessels
- E482 Guide for Application of Neutron Transport Methods for Reactor Vessel Surveillance
- E509 Guide for In-Service Annealing of Light-Water Moderated Nuclear Reactor Vessels
- E636 Guide for Conducting Supplemental Surveillance Tests for Nuclear Power Reactor Vessels

- E646 Test Method for Tensile Strain-Hardening Exponents (n -Values) of Metallic Sheet Materials
- E693 Practice for Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacements Per Atom (DPA)
- E844 Guide for Sensor Set Design and Irradiation for Reactor Surveillance
- E853 Practice for Analysis and Interpretation of Light-Water Reactor Surveillance Neutron Exposure Results
- E854 Test Method for Application and Analysis of Solid State Track Recorder (SSTR) Monitors for Reactor Surveillance
- E900 Guide for Predicting Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials
- E910 Test Method for Application and Analysis of Helium Accumulation Fluence Monitors for Reactor Vessel Surveillance
- E944 Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance

³ For referenced ASTM standards, visit the ASTM website, www.astm.org, or contact ASTM Customer Service at service@astm.org. For *Annual Book of ASTM Standards* volume information, refer to the standard's Document Summary page on the ASTM website.

- E1005 Test Method for Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance
- E1006 Practice for Analysis and Interpretation of Physics Dosimetry Results from Test Reactor Experiments
- E1018 Guide for Application of ASTM Evaluated Cross Section Data File
- E1035 Practice for Determining Neutron Exposures for Nuclear Reactor Vessel Support Structures
- E1214 Guide for Use of Melt Wire Temperature Monitors for Reactor Vessel Surveillance
- E1253 Guide for Reconstitution of Charpy-Sized Specimens
- E2005 Guide for Benchmark Testing of Reactor Dosimetry in Standard and Reference Neutron Fields
- E2006 Guide for Benchmark Testing of Light Water Reactor Calculations
- E2215 Practice for Evaluation of Surveillance Capsules from Light-Water Moderated Nuclear Power Reactor Vessels
- E2956 Guide for Monitoring the Neutron Exposure of LWR Reactor Pressure Vessels

2.2 ASTM Adjunct:⁴

ADJE090015-EA Adjunct for E900-15 Technical Basis for the Equation Used to Predict Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials

2.3 Nuclear Regulatory Documents:⁵

Code of Federal Regulations, Chapter 10, Part 50 Appendices G and H

Code of Federal Regulations, Chapter 10, Part 21 Reporting of Defects and Noncompliance

Regulatory Guide 1.99 Radiation Embrittlement of Reactor Vessel Materials

Regulatory Guide 1.150 Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations

Regulatory Guide 1.190 Computational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence

2.4 American Society of Mechanical Engineers Standard:⁶

Boiler and Pressure Vessel Code Sections III and XI

2.5 Bureau International de Poids et Mesures Documents:⁷

The SI Brochure: The International System of Units (SI)

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

3.1 Aging light-water reactor pressure vessels (LWR-PV) accumulate significant neutron fluence exposures, with consequent changes in their state of steel embrittlement. Recognizing that accurate and validated measurement and predictive methods are needed to periodically evaluate the metallurgical condition of these reactor vessels, and in some instances reactor vessel support structures (16, 17), international multi-

laboratory work directed towards the improvement of LWR-PV surveillance has been conducted (1, 2, 4, 29-34).

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

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

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

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

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

3.7 Federal Regulation 10 CFR 50 calls for adherence to several ASTM standards that require establishment of a surveillance program for each power reactor and incorporation of fluence monitors for post-irradiation neutron field evaluation. Revised and new standards must be structured to be up to date, flexible, and above all, consistent (see Section 4).

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

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

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

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

4. Significance and Use

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

4.2 Approach and Primary Objectives:

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

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

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

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

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

4.3 Dosimetry Analysis and Interpretation (1, 3-5, 21, 28, 29, 35, 37, 38)—When properly implemented, validated, and calibrated by vendor/utility groups, state-of-the-art dosimetry practices exist that are adequate for existing and future LWR power plant surveillance programs. The uncertainties and errors associated with the individual and combined effects of the different variables (items 1.4.1 – 1.4.10 of 1.4) are considered in this section and in 4.4 and 4.5. In these sections, the accuracy (uncertainty and error) statements that are made are quantitative and representative of state-of-the-art technology. Their correctness and use for making EOL predictions for any given LWR power plant, however, are dependent on such factors as (1) the existing plant surveillance program, (2) the plant geometrical configuration, and (3) available surveillance results from similar plants. As emphasized in Section III-A of Ref (7), however, these effects are not unique and are dependent on (1) the surveillance capsule design, (2) the configuration of the reactor core and internals, and (3) the location of the

surveillance capsule within the reactor geometry. Further, the statement that a result could be in error is dependent on how the neutron and gamma ray fields are estimated for a given reactor power plant (1, 11, 28, 36, 39, 40). For most of the error statements in 4.3 – 4.5, it is assumed that these estimates are based on reactor transport theory calculations that have been normalized to the core power history (see 4.4.1.2) and not to surveillance capsule dosimetry results. The 4.3 – 4.5 accuracy statements, consequently, are intended for use in helping the standards writer and user to determine the relative importance of the different variables in regard to the application of the set of ASTM standards, Fig. 1, for (1) LWR-PV surveillance program, (2) as instruments of licensing and regulation, and (3) for establishing improved metallurgical databases.

4.3.1 Required Accuracies and Benchmark Field Referencing:

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

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

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

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

(1) Needed for quality control and certification of current and improved dosimetry practices; and

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

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

4.3.3 Additional Validation Work for Dosimetry Detectors:

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

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

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

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

4.4.1 *Required Accuracies and Benchmark Field Referencing*:

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

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

4.4.2 *Power Plant Reactor Physics Analysis and Interpretation*:

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

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

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

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

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

4.4.4 *PWR and BWR Generic Power Reactor Tests*:

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

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

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

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

4.4.5 *Operating Power Reactor Tests*:

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

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

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

4.5.1 *Required Accuracies and Benchmark Field Referencing*:

4.5.1.1 The accuracies desirable and achievable for LWR-PV steel (test reactor specimens, surveillance capsule specimens, and vessels and support structure) data correlation and data extrapolation (to predict fracture toughness changes both in space and time) are of the order of ± 10 to 30 % (1σ). In order to achieve such a goal, however, the metallurgical parameters ($\Delta NDTT$, 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, 4, 5, 11, 39, 50, 54, 55).

4.5.1.3 Benchmark field referencing of utilities’ vendor/service laboratories, as well as advanced practices, is in progress or being planned that is (1, 3, 4, 11, 21, 28, 36-38, 52).