Standard Practice for Analysis and Interpretation of Light-Water Reactor Surveillance Results[1]

1. Scope

1.1 This practice covers the methodology, summarized in Annex A1, 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 present and future condition of the pressure vessel and its support structures[2] **(1-74)**.[3]

1.2 This practice relies on, and ties together, the application of several supporting ASTM standard practices, guides, and methods (see Master Matrix E706) **(1, 5, 13, 48, 49)**.[2] In order to make this practice at least partially self-contained, a moderate amount of discussion is provided in areas relating to ASTM and other documents. Support subject areas that are discussed include reactor physics calculations, dosimeter selection and analysis, and exposure units.

1.3 This practice is restricted to direct applications related to surveillance programs that are established in support of the operation, licensing, and regulation of LWR nuclear power plants. Procedures and data related to the analysis, interpretation, and application of test reactor results are addressed in Practice E1006, Guide E900, and Practice E1035.

1.4 *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:*[4]

- 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, E706 (IID)
- E509 Guide for In-Service Annealing of Light-Water Moderated Nuclear Reactor Vessels
- E706 Master Matrix for Light-Water Reactor Pressure Vessel Surveillance Standards, E 706(0) (Withdrawn 2011)[5]
- E844 Guide for Sensor Set Design and Irradiation for Reactor Surveillance, E 706 (IIC)
- E854 Test Method for Application and Analysis of Solid State Track Recorder (SSTR) Monitors for Reactor Surveillance, E706(IIIB)
- E900 Guide for Predicting Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials, E706 (IIF)
- E910 Test Method for Application and Analysis of Helium Accumulation Fluence Monitors for Reactor Vessel Surveillance, E706 (IIIC)
- E944 Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance, E 706 (IIA)
- E1005 Test Method for Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance, E 706 (IIIA)
- E1006 Practice for Analysis and Interpretation of Physics Dosimetry Results for Test Reactors, E 706(II)
- E1018 Guide for Application of ASTM Evaluated Cross Section Data File, Matrix E706 (IIB)
- E1035 Practice for Determining Neutron Exposures for Nuclear Reactor Vessel Support Structures

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

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[2] ASTM Practice E185 gives reference to other standards and references that address the variables and uncertainties associated with property change measurements. The reference standards are A370, E8, E21, E23, and E208.

[3] The boldface numbers in parentheses refer to the list of references appended to this practice. For an updated set of references, see the E706 Master Matrix.

[4] 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.

[5] The last approved version of this historical standard is referenced on www.astm.org.

E1214 Guide for Use of Melt Wire Temperature Monitors for Reactor Vessel Surveillance, E 706 (IIIE)

E2006 Guide for Benchmark Testing of Light Water Reactor Calculations

2.2 *Other Documents:*

NUREG/CR-1861 HEDL-TME 80-87 LWR Pressure Vessel Surveillance Dosimetry Improvement Program: PCA Experiments and Blind Test[6]

ASME Boiler and Pressure Vessel Code, Sections III and IX[7]

Code of Federal Regulations, Title 10, Part 50, Appendixes G and H[8]

3. Significance and Use

3.1 The objectives of a reactor vessel surveillance program are twofold. The first requirement of the program is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region resulting from exposure to neutron irradiation and the thermal environment. The second requirement is to make use of the data obtained from the surveillance program to determine the conditions under which the vessel can be operated throughout its service life.

3.1.1 To satisfy the first requirement of 3.1, the tasks to be carried out are straightforward. Each of the irradiation capsules that comprise the surveillance program may be treated as a separate experiment. The goal is to define and carry to completion a dosimetry program that will, a posteriori, describe the neutron field to which the materials test specimens were exposed. The resultant information will then become part of a data base applicable in a stricter sense to the specific plant from which the capsule was removed, but also in a broader sense to the industry as a whole.

3.1.2 To satisfy the second requirement of 3.1, the tasks to be carried out are somewhat complex. The objective is to describe accurately the neutron field to which the pressure vessel itself will be exposed over its service life. This description of the neutron field must include spatial gradients within the vessel wall. Therefore, heavy emphasis must be placed on the use of neutron transport techniques as well as on the choice of a design basis for the computations. Since a given surveillance capsule measurement, particularly one obtained early in plant life, is not necessarily representative of long-term reactor operation, a simple normalization of neutron transport calculations to dosimetry data from a given capsule may not be appropriate **(1-67)**.[2]

3.2 The objectives and requirements of a reactor vessel's support structure's surveillance program are much less stringent, and at present, are limited to physics-dosimetry measurements through ex-vessel cavity monitoring coupled with the use of available test reactor metallurgical data to determine the condition of any support structure steels that might be subject to neutron induced property changes **(1, 29, 44-58, 65-70)**.

4. Establishment of the Surveillance Program

4.1 Practice E185 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.

4.1.1 From the viewpoint of the radiation analyst, the criteria explicated in Practice E185 are met by the completion of the following tasks: (*1*) Determine the locations within the reactor that provide suitable lead factors (see Practice E185) for each irradiation capsule relative to the pressure vessel; (*2*) Select neutron sensor sets that provide adequate coverage over the energy range and fluence range of interest; (*3*) Specify sensor set locations within each irradiation capsule to define neutron field gradients within the metallurgical specimen array. For reactors in which the end of life shift in RT_{NDT} of the pressure vessel beltline material is predicted to be less than 100 °F, gradient measurements are not required. In that case sensor set locations may be chosen to provide a representative measurement for the entire surveillance capsule; and (*4*) Establish and adequately benchmark neutron transport methodology to be used both in the analysis of individual sensor sets and in the projection of materials properties changes to the vessel itself.

4.1.2 The first three items listed in the preceding paragraph are carried out during the design of the surveillance program. However, the fourth item, which directly addresses the analysis and interpretation of surveillance results, is performed following withdrawal of the surveillance capsules from the reactor. To provide continuity between the designer and the analyst, it is recommended that the documentation describing the surveillance programs of individual reactors provide details of irradiation capsule construction, locations of the capsules relative to the reactor core and internals, and sensor set design that are adequate to allow accurate evaluations of the surveillance measurement by the analyst. Well documented (1) metallurgical and (2) physics-dosimetry data bases now exist for use by the analyst based on both power reactor surveillance capsule and test reactor results **(1, 12, 19-38, 58-64)**.

4.1.3 Information regarding the choice of neutron sensor sets for LWR surveillance applications is provided in Matrix E706: Guide E844, Sensor Set Design; Test Method E1005, Radiometric Monitors; Test Method E854, Solid State Track Recorder Monitors; Specification E910, Helium Accumulation Fluence Monitors; and Damage Monitors. Dosimeter materials currently in common usage and acceptable for use in surveillance programs include Cu, Ti, Fe, Ni, Nb, U^{238}, Np^{237}, U^{235}, and Co-Al. All radionuclide analysis of dosimeters should be calibrated to known sources such as those supplied by the National Institute of Standards and Technology (NIST) or The International Atomic Energy Agency (IAEA). All quality

[6] Available from NRC Public Document Room, 1717 H St., NW, Washington, DC 20555.

[7] Available from American Society of Mechanical Engineers, Three Park Ave., New York, NY 10016-5990.

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

assurance information pertinent to the sensor sets must be documented with the description of the surveillance program (**1**, **40-43**, **48**, **51-58**).

4.1.4 As indicated in 4.1.1, neutron transport methods are used both in the design of the surveillance program and in the analysis and interpretation of capsule measurements. During the design phase, neutron transport calculations are used to define the neutron field within the pressure vessel wall and, in conjunction with damage trend curves, to predict the degree of embrittlement of the reactor vessel over its service life. Embrittlement gradients are in turn used to determine pressure-temperature limitations for normal plant operation as well as to evaluate the effect of various heat-up/cool-down transients on vessel condition.

4.1.5 The neutron transport methodology used for these computations must be well benchmarked and qualified for application to LWR configurations. The PCA (Experiment and Blind Test) data documented in Ref **47** provide one configuration for benchmarking basic transport methodology as well as some of the input data used in power reactor calculations. Other suitably defined and documented benchmark experiments, such as those for VENUS (**1**, **43**, **45**) and for NESDIP (**1**, **46**, **50**), may also be used to provide method verification. However, further analytical/experimental comparisons are required to qualify a method for application to LWRs that have a more complex geometry and that require a more complex treatment of some input parameters, particularly of reactor core power distributions (**1**, **65-67**). This additional qualification may be achieved by comparison with measurements taken in the reactor cavity external to the pressure vessel of selected operating reactors (**1**, **51-57**).

4.1.6 All experimental/analytical comparisons that comprise the qualification program for a neutron transport methodology must be documented. At a minimum, this documentation should provide an assessment of the uncertainty or error inherent in applying the methodology to the evaluation of surveillance capsule dosimetry and to the determination of damage gradients within the beltline region of the pressure vessel (**1**, **12**, **19-21**, **23-29**, **36**, **38**, **43-48**, **50-57**).

4.1.7 In the application of neutron transport methodology to the evaluation of surveillance dosimetry as well as to the prediction of damage within the pressure vessel, several options are available regarding the choice of design basis power distributions, the necessary detail in the geometric mockup, and the normalization of the analytical results. The methodology chosen by any analyst should be documented with sufficient detail to permit a critical evaluation of the overall approach. Further discussions of the application of neutron transport methods to LWRs are provided in Guide E482.

4.1.8 To ensure that metallurgical results obtained from surveillance capsule measurements may be applied to the determination of the pressure vessel fracture toughness, the irradiation temperature of the surveillance test specimens must be documented (see Guide E1214).

4.2 As stated in 3.2, the requirements for the establishment of a surveillance program for reactor vessel support structures are much less stringent than for the reactor vessel, and the analyst is referred to Practice E1035, for more information.

5. Analysis of Individual Surveillance Capsules

5.1 It is recognized that for many operating power reactors, the documentation of baseline neutron transport calculations and sensor set design information may not be available. In that event, to whatever extent possible the required information should be provided by the service laboratory in the respective surveillance report (**1**, **29**, **58**).

5.2 Radiometric analysis of capsule sensor sets should follow procedures outlined in Test Method E1005. For sensors such as the fission monitors which may be gamma-ray-sensitive, photo reaction corrections should be derived from the results of gamma-ray transport calculations performed for the explicit capsule configuration under examination. Photo reaction corrections in LWR environments have been shown to be extremely configuration dependent (**1**, **29**, **58**). Gamma-ray calculations should be well benchmarked. One such suitable reactor geometry benchmark is VENUS-1 (**75**, **76**).

5.3 In calculating spectrum averaged reaction cross sections from neutron transport calculations, care should be taken to model the explicit capsule configuration and location under examination (see Guide E482.) It will be necessary to determine uncertainties associated with the determination of damage exposure parameters. The procedures outlined in Guide E944, IIA can, in many cases, be useful for accomplishing this. To achieve satisfactory uncertainty bounds for the damage parameters a sufficiently large set of foils should be used as stipulated in 4.1.3 (**1**, **29**, **36**).

5.4 The report of the capsule analysis should contain the following information. Uncertainties should be included in all data (**1**, **29**, **36**).

5.4.1 Damage exposure parameters at the position of the metallurgical specimens. These values will be used for correlation with metallurgical data to develop damage trend curves. Neutron fluence (E > 1.0 MeV) is presently required. However, iron dpa (displacements per atom) and neutron fluence (E > 0.1 MeV) should also be included for future reference. These exposure values are derived from a combination of measurements and calculations and must include estimates of uncertainty bounds,

5.4.2 The neutron spectra, reaction rates, reaction cross sections, and all other nuclear constants used in the derivation of exposure values for the capsule,

5.4.3 The gamma-ray energy spectra and reaction cross sections used to make photoreaction corrections for the neutron sensor sets,

5.4.4 The power-time history of the reactor during the irradiation period of the subject capsule, and

5.4.5 Spatial gradients of neutron fluence rate (E > 1.0 MeV), neutron fluence (E > 1.0 MeV), and dpa throughout the metallurgical specimen array.

5.4.6 In addition, the documentation supporting the benchmarking/qualification of sensor sets and reactor physics methodology should be either referenced or included as an appendix to the dosimetry report.

6. Projection of Vessel and Support Structure Condition for Future Plant Operation

6.1 *Reactor Vessel:*

6.1.1 This practice requires the use of a fully benchmarked and qualified neutron transport methodology in both the design of the surveillance program and the analysis of individual surveillance capsules. The neutron field information obtained from these computations should also be used to project damage gradients within the pressure vessel wall. Currently, such projections are based on effective vessel wall neutron fluence (E > 1.0 MeV). Guide E900 and NRC Regulatory Guide 1.99 **(74)** specify methods for determining the attenuation of the effective vessel wall fluence, up to a specified depth from the inner surface. However, it is recommended that supplementary projections based on dpa maps throughout the pressure vessel beltline region/surveillance capsule geometry be included in the surveillance report **(1, 12, 19, 20, 21, 23-29, 33, 36, 38-48, 51-67).**

6.1.2 It is recommended that all surveillance results for a generic reactor type (similar reactor geometry and fuel loading) be used as a data base to qualify the reactor physics methodology as to its applicability to a particular reactor system. This approach should, in the long term, provide a statistically significant validation of the calculations.

6.1.3 Capsules removed from symmetric positions in generic reactor geometries represent a series of repeat measurements. As such, the measured data will reflect the variability in important parameters such as water temperature, reactor dimensions, fuel loading, sensor set design, sensor set analysis, and reactor operating characteristics. By taking advantage of a large data base obtained from these repeat measurements, the uncertainties introduced by these various parameters may be better understood and possibly reduced.

6.1.4 When evaluating the results of a given surveillance capsule analysis, the measured capsule exposure should be compared directly with neutron transport analysis and with all available experimental data obtained from similar capsules removed from reactors having the same design. If the agreement between measurement and calculation is within the range indicated by the benchmark documentation for the specified methodology, the analytically derived neutron field parameters should be used for all damage determinations for the pressure vessel **(29).**

6.1.5 If the measurements differ from the calculations by more than the margins indicated by the benchmark documentation, further investigation of the measurement approach and the mode of operation of the reactor in question should be undertaken. Any adjustments made to vessel embrittlement projections based on the results of these investigations should be justified and fully documented in the surveillance report.

6.2 *Reactor Vessel Support Structures*—The analyst is referred to Practice E1035.

7. Extrapolating Reactor Vessel Surveillance Dosimetry Results

7.1 Knowledge of the time-dependent relationship between exposure parameters at surveillance locations and selected (r, θ, z) locations within the pressure vessel wall is required to allow determination of the time-dependent radiation damage to the pressure vessel. The time dependency must be known to allow proper accounting for compications due to burn-up, as well as, change in core loading configurations**(20, 65-67).**. An estimate of the uncertainty in the neutron exposure parameter values at selected (r, θ, z) points in the vessel wall **(1).** is also needed to place an upper bound on the allowable operating lifetime of the reactor vessel without remedial action **(21, 22, 71)**. (See Guide E509).

7.2 Several other ASTM practices cover various aspects of the extrapolation problem (see 2.1). The basic approach is that a benchmarked Guide E482, transport calculation is to be used to supply the neutron field information at the (r, θ, z) points in the pressure vessel wall where property deterioration information will be calculated using Guide E900, or other trend curves **(3, 12, 20, 24-27, 72-74)**. The dosimetry information obtained from reactor cavity (ex-vessel) and surveillance capsule (invessel) measurements is to be used to adjust the transport results and ensure that the transport calculation is valid. The adjustments are to be accomplished using the guidelines presented in Guide E944. Dosimetry from monitors in the reactor cavity and surveillance capsules will be used in establishing uncertainties for the calculated neutron field at selected (r, θ, z) positions in the pressure vessel wall. Time dependence of the core power distribution (due to burnup within a given cycle, or due to variations in cycle to cycle loading), surveillance capsule perturbation effects, and dosimetry monitor experimental effects must be recognized as complications, and these effects must be accounted for in the calculation and adjustment methods chosen **(1, 3, 20, 21, 65-67)**.

7.3 *Spatial Extrapolations:*

7.3.1 *Transport Codes*—a three dimensional or two dimensional [(r, θ), (x, y)] transport code is needed for the calculation of the neutron and gamma fields in the region from the core to the interior of the biological shield beyond the pressure vessel. Guide E482 should be followed for the calculations and Guide E944 for measured dosimetry adjustments.

7.3.2 *Dosimetry Sensor Analysis*—For analysis of any given set of reactor cavity or surveillance capsule dosimetry sensors, the integral reactions or reaction rates of the individual sensors, or both, should be calculated, using the results of the transport calculation, see Guides E844, E1018, E2006, Test Methods E1005, E854, and E910 (See 2.1). If the calculated and experimental integral results (C/E ratios) agree to within the required accuracy (~ 5 to 15 %, 1σ being the best attainable, see **(47)** expected from the benchmark calibration of the transport code, the transport calculation may be used directly to calculate the neutron field at all (r, θ, z) points in the pressure vessel wall. If the C/E ratios do not agree within acceptable accuracy limits, a physics-dosimetry adjustment code analysis should be performed as described in E944. Having established the required consistency, the adjusted transport code results may be used to calculate the neutron field at all points in the pressure vessel wall with the uncertainty estimates derived from the application of the adjustment codes. Direct use of the transport code results with appropriate bias factors and uncertainties is another acceptable approach.

7.3.3 *Surveillance Capsule Results*—If the calculated neutron field at the surveillance capsule is inconsistent with the experimental dosimetry results, an attempt should be made to uncover and correct errors in order to obtain consistency. Particular attention will be required to sensor monitor correction factors for perturbation, photo-reaction, impurity, burn-in, and other effects.

7.3.3.1 If the transport result indicates a higher fluence than that indicated by the dosimetry, the transport result can be used for extrapolation purposes, but with an appropriate increase in the stated uncertainty for the results.

7.3.3.2 If the transport calculation indicates a lower fluence than that which would be consistent with the dosimetry (taking account of the uncertainties in both the dosimetry and transport results) and if the discrepancy cannot be resolved, then the transport results should be scaled up proportionally to obtain agreement, following which the transport results are to be used for extrapolation purposes. In this case, appropriate increases should be made in the stated uncertainties of the final result, and documented logic should be provided to defend the assigned uncertainties.

7.3.4 *Ex-Vessel Surveillance Results*—Ex-vessel reactor cavity dosimetry is to be treated in the same manner as surveillance capsule dosimetry, but care must be exercised to ensure that the physics calculation modeling is adequate and includes the proper modeling of the reactor cavity surveillance capsule and any covers, as well as any nearby vessel support members.

7.3.4.1 The biological shield is accurately modeled.

7.3.4.2 In the final calculation of the neutron and gamma field at any point in the vessel wall, proper statistical weight should be given to ex-vessel dosimetry, taking account of modeling problems as well as the possibility that a larger logarithmic extrapolation or interpolation in absolute fluence value exists from ex-vessel positions to a ¼ T location when compared to the extrapolation or interpolation from an internal surveillance capsule position to a ¼ T location.

7.3.5 *Power Plant Dimensions*—In all calculations, as-built dimensions should be used. If they are unavailable, documented logic should be presented to defend the dimensions used, and the uncertainty in the final results should reflect the added uncertainty. It should be noted that dpa declines ~10 %/cm of radial travel, in water, and deviations of ~3 cm between design dimensions and as-built dimensions have been observed in commercial reactors.

7.4 *Time Extrapolations*—In the case where a time averaged core loading has been used to define the neutron source term, the fluence or dpa in future years is estimated by multiplying by the expected integrated time at full power. Existing problems associated with time extrapolations (for example, saturation effects and differences in the slope of trend curves for different ferritic steels) are addressed elsewhere. The reader is referred to Refs **(1, 3, 12, 21, 23-27, 72-74)**, and Guide E900 for more information on these subjects.

8. Uncertainties

8.1 Analysis and measurement accuracies (uncertainties and errors) in the areas of concern for this practice may be difficult to determine. However, they should be properly addressed **(1, 12, 19-22, 23-29, 36, 38, 39, 43, 44, 47, 48, 51, 58-64).** When uncertainties and errors are well defined, as in integral reaction rate measurements, they should be estimated and summarized in an accuracy table. For more difficult uncertainty situations, such as for damage exposure parameters, the procedure for determining uncertainties must be well documented. A statement must be included that indicates what the uncertainty estimates do and do not cover. It will be necessary to accept incomplete or nonrigorous uncertainty and error estimates when there is no readily available alternative.

9. Keywords

9.1 damage exposure parameter; dpa; embrittlement; LWR; pressure vessel; reactor surveillance; surveillance capsule

ANNEX

(Mandatory Information)

A1. PROCEDURES FOR ANALYSIS AND INTERPRETATION OF NUCLEAR REACTOR SURVEILLANCE RESULTS

A1.1 Procedures

A1.1.1 Establish the basic surveillance test program for each operating power plant. Currently, Practice E185 is available and is used. The surveillance test program should include the following:

A1.1.1.1 Determination of surveillance capsule spatial fluence rate, fluence-spectral and DPA maps for improved correlation and application of measured property change data (upper shelf, NDTT, and so forth). 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.

A1.1.1.2 As appropriate, use of measured/calculated DPA damage for normalization of Charpy to Charpy (and other metallurgical specimen) variations in neutron fluence rate, fluence, and spectra. Here, an increased use of a larger number of metallurgical specimen iron drillings may be appropriate for dosimetry.

A1.1.2 Establish a reactor physics computational method applicable to the surveillance program. Guide E482 provides general guidance in this area. The computational method should include the following:

A1.1.2.1 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.

A1.1.2.2 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.

A1.1.2.3 Determination of the effect of surveillance capsule perturbations and photofission on the evaluation of capsule dosimetry. Adjustment codes should be used, as appropriate, to combine reactor physics computations with dosimetry measurements.

A1.1.2.4 Benchmark validation of the analytical method as described in Guide E2006.

A1.1.3 Establish methods for relating dosimetry, metallurgy, and temperature data from the surveillance program to current and future reactor vessel and support structure conditions.

A1.1.3.1 Differences in core power distributions that may be expected during long-term operation and that may impact the extrapolation of surveillance results into the future. As previously stated, ex-vessel dosimetry should be used for verification.

A1.1.3.2 Establish methods to verify A1.1.2 and A1.1.3 and to determine uncertainty and error bounds for the interpretation of the combined results of dosimetry, metallurgical and temperature measurements. Practice E185 provides general guidance in this area. The uncertainty analysis should address the separate and combined accuracy requirements of dosimetry, metallurgy, and temperature-measurement techniques.

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