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Standard Practice for Analysis and Interpretation of Light-Water Reactor Surveillance Results, E706(IA)[1]

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~~ε1 NOTE—Keywords were added editorially in April 1996.~~

[1] This practice is under the jurisdiction of ASTM Committee E-10 on Nuclear Technology and Applications and is the direct responsibility of Subcommittee E10.05 on Nuclear Radiation Metrology.

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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] **(1-70).**[3]

1.2 This practice relies on, and ties together, the application of several supporting ASTM standard practices, guides, and methods ~~that are in various stages of completion~~ (see ~~Fig. 1 and~~ Master Matrix E 706) **(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 Since several of the standards shown in Fig.~~

NOTE 1—(Figure 1 ~~are not currently~~ is deleted in ~~place, some of~~ the ~~requirements listed in Annex A1 should, at this time, be treated as recommendations. Appropriate caution should be exercised until each~~ latest update. The user is refered to Master Matrix E 706 for the latest figure of the standards ~~has been put into use.~~

~~1.4 This~~ interconnectivity).

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 ~~Matrix E 706 (IE),~~ Practice E 560, ~~Matrix E 706 (IC), E706 (II),~~ Practice E 1006, Guide E 900, and ~~E 706(IG).~~

~~1.5~~ Practice E 1035.

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~~ References

2.1 *ASTM Standards:*

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, E706 (IB)[4]

E 185 Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels, E706 (IF)[4]

E 482 Guide for Application of Neutron Transport Methods for Reactor Vessel Surveillance, E706 (IID)[4]

E 560 Practice for Extrapolating Reactor Vessel Surveillance Dosimetry Results, E706 (IC)[4]

E 636 Guide for Conducting Supplemental Surveillance Tests for Nuclear Power Reactor Vessels, E706 (IH)[4]

E 693 Practice for Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacements Per Atom (DPA), E706 (ID)[4]

[2] ASTM Practice E 185 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] *Annual Book of ASTM Standards*, Vol 12.02.

E 706 Master Matrix for Light-Water Reactor Pressure Vessel Surveillance Standards[4]
~~IE Damage Correlation for Reactor Vessel Surveillance~~[5]
~~IIE Benchmark Testing of Reactor Vessel Dosimetry~~[6]
~~IIID Application and Analysis of Damage Monitors for Reactor Vessel Surveillance~~[6]
~~IIIE Application and Analysis of Temperature Monitors~~[6]
E 844 Guide for Sensor Set Design and Irradiation for Reactor Surveillance, E706 (IIC)[4]
E 854 Test Method for Application and Analysis of Solid State Track Recorder (SSTR) Monitors for Reactor Surveillance, E706 (IIIB)[4]
E 900 Guide for Predicting Neutron Radiation Damage to Reactor Vessel Materials, E706 (IIF)[4]
E 910 Specification for Application and Analysis of Helium Accumulation Fluence Monitors for Reactor Vessel Surveillance, E706 (IIIC)[4]
E 944 Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance, E706 (IIA)[4]
E 1005 Test Method for Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance, E706 (IIIA)[4]
E 1006 Practice for Analysis and Interpretation of Physics Dosimetry Results for Test Reactors, E706 (II)[4]
E 1018 Guide for Application of ASTM Evaluated Cross Section Data ~~File (ENDF/A)—Cross Section and Uncertainty~~ File, E706 (IIB)[4]
E 1035 Practice for Determining Radiation Exposures for Nuclear Reactor Vessel Support Structures, E706 (IG)[4]
E 1214 Guide for Use of Melt Wire Temperature Monitors for Reactor Vessel Surveillance, E706 (IIIE)[4]
E 2005 Guide for the Benchmark Testing of Reactor Dosimetry in Standard and Reference Neutron Fields, E706 (IIE-1)[4]
E 2006 Guide for the Benchmark Testing of Light Water Reactor Calculation[4]

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[5]

~~[5] For standards that are in the draft stage and have not received an ASTM designation, see Section 5 and Figs. 1 and 2~~

[5] Available from American Society of ~~Matrix E 706.~~ Mechanical Engineers, Three Park Ave., New York, NY 10016-5990.

~~*Annual Book of ASTM Standards*, Vol 12.02. The reference in parentheses refers to Section 5 and Figs. 1 and 2 of Matrix E 706.~~

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

Code of Federal Regulations, Title 10, Part 50, Appendixes G and H[7]

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 E 185 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 E 185 are met by the completion of the following tasks: (*1*) Determine the locations within the reactor that provide suitable lead factors (see Practice E 185) 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 E 706: ~~IIC,~~ Guide E 844, Sensor Set Design; ~~IIIA,~~ Test Method E 1005, Radiometric Monitors; ~~IIIB,~~ Test Method E 854, Solid State Track Recorder Monitors; ~~IIIC,~~ Specification E 910, 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, 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 ~~Bureau~~ Institute of Standards and Techrnology (N~~B~~IST) or The International Atomic Energy Agency (IAEA). All quality 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.

[7] Available from ~~NRC Public Document Room, 1717 H St. NW,~~ Superintendent of Documents, U. S. Government Printing Office, Washington, DC ~~20555.~~ 20402.

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 Practice ~~E 560, IC,~~ E 560 and Guide ~~E 482, IID.~~ E 482.

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 ~~Matrix E 706 (IIIE)).~~ Guide E 1214).

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 E 1035, ~~IG~~ 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 ~~E 1005, IIIA.~~ E 1005. 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).**

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 ~~E 482, IID.)~~ E 482.) It will be necessary to determine uncertainties associated with the determination of damage exposure parameters. The procedures outlined in Guide E 944, 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 flux (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, all such projections are based on neutron fluence (E > 1.0 MeV). 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 E 1035.

7. Uncertainties

7.1 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.

8. Keywords

8.1 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 E 185 is available and is used. ~~However, updated versions of this standard~~ The surveillance test program should include the following:

A1.1.1.1 Determination of surveillance capsule spatial flux-fluence-spectral and DPA maps for improved correlation and application of measured property change data (upper shelf, NDTT, ~~etc.).~~ 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 flux, 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. Currently, Guide E 482 and Practice E 560 provide general guidance in this area. ~~However, updated versions of these standards~~ 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 E 2006.

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. Currently, Practice E 560 provides general guidance in this area. ~~An updated version of this standard~~ The analysis should include the following considerations.

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. Currently, Practice E 185 provides general guidance in this area. ~~An updated version of this standard~~ The uncertainty analysis should ~~more completely~~ address the separate and combined accuracy requirements of dosimetry, metallurgy, and temperature-measurement techniques.

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