Designation: E 1006 – 96

Standard Practice for Analysis and Interpretation of Physics Dosimetry Results for Test Reactors, E706(II)[1]

This standard is issued under the fixed designation E 1006; 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 practice covers the methodology summarized in Annex A1 to be used in the analysis and interpretation of physics-dosimetry results from test reactors.

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 Matrix E 706).

1.3 Support subject areas that are discussed include reactor physics calculations, dosimeter selection and analysis, exposure units, and neutron spectrum adjustment methods.

1.4 This practice is directed towards the development and application of physics-dosimetry-metallurgical data obtained from test reactor irradiation experiments that are performed in support of the operation, licensing, and regulation of LWR nuclear power plants. It specifically addresses the physics-dosimetry aspects of the problem. Procedures related to the analysis, interpretation, and application of both test and power reactor physics-dosimetry-metallurgy results are addressed in Practices E 185, E 560, E 853, and E 1035, Matrix E 706(IE), Guide E 900, and Test Method E 646.

1.5 *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:*

E 185 Practice for Conducting Surveillance Tests for Light Water-Cooled Nuclear Power Reactor Vessels, (IF)[2,3]

E 482 Guide for Application of Neutron Transport Methods for Reactor Vessel Surveillance, (IID)[2,3]

E 560 Practice for Extrapolating Reactor Vessel Surveillance Dosimetry Results, (IC)[2,3]

E 646 Test Method for Tensile Strain-Hardening Exponents (*n*-Values) of Metallic Sheet Materials[4]

E 693 Practice for Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacements Per Atom (DPA), (ID)[2,3]

E 706 Master Matrix for Light-Water Reactor Pressure Vessel Surveillance Standards, (O)[3]

IE Damage Correlation for Reactor Vessel Surveillance[5]

IIE Benchmark Testing of Reactor Vessel Dosimetry[5]

IIID Analysis and Application of Damage Monitors for Reactor Vessel Surveillance[5]

E 844 Guide for Sensor Set Design and Irradiation for Reactor Surveillance, (IIC)[2,3]

E 853 Practice for Analysis and Interpretation of Light-Water Reactor Surveillance Results, (IA)[2,3]

E 854 Test Method for Application and Analysis of Solid State Track Recorder (SSTR) Monitors for Reactor Surveillance, (IIIB)[2,3]

E 900 Guide for Predicting Neutron Radiation Damage to Reactor Vessel Materials, (IIF)[2,3]

E 910 Specification for Application and Analysis of Helium Accumulation Fluence Monitors for Reactor Vessel Surveillance, (IIIC)[2,3]

E 944 Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance, (IIA)[2,3]

E 1005 Test Method for Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance, (IIIA)[2,3]

E 1018 Guide for Application of ASTM Evaluated Cross Section Data File, (IIB)[2,3]

E 1035 Practice for Determining Radiation Exposures for Nuclear Reactor Vessel Support Structures[2]

2.2 *Nuclear Regulatory Documents:*

Code of Federal Regulations, "*Fracture Toughness Requirements*," Chapter 10, Part 50, Appendix G[6]

Code of Federal Regulations, "*Reactor Vessel Materials Surveillance Program Requirements*," Chapter 10, Part 50, Appendix H[6]

Regulatory Guide 1.99, Rev 2, "*Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel*

[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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[2] The reference in parentheses refers to Section 5 as well as to Figs. 1 and 2 of Matrix E 706.

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

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

[5] For standards that are in the draft stage and have not received an ASTM designation, see Section 5 as well as Figures 1 and 2 of Matrix E 706.

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

Materials," U.S. Nuclear Regulatory Commission, April 1977[6]

3. Significance and Use

3.1 The mechanical properties of steels and other metals are altered by exposure to neutron radiation. These property changes are assumed to be a function of chemical composition, metallurgical condition, temperature, fluence (perhaps also fluence rate), and neutron spectrum. The influence of these variables is not completely understood. The functional dependency between property changes and neutron radiation is summarized in the form of damage exposure parameters that are weighted integrals over the neutron fluence spectrum.

3.2 The evaluation of neutron radiation effects on pressure vessel steels and the determination of safety limits require the knowlege of uncertainties in the prediction of radiation exposure parameters (for example, dpa (Practice E 693), neutron fluence greater than 1.0 MeV, neutron fluence greater than 0.1 MeV, thermal neutron fluence, etc.). This practice describes recommended procedures and data for determining these exposure parameters (and the associated uncertainties) for test reactor experiments.

3.3 The nuclear industry draws much of its information from databases that come from test reactor experiments. Therefore, it is essential that reliable databases are obtained from test reactors to assess safety issues in Light Water Reactor (LWR) nuclear power plants.

4. Establishment of the Physics-Dosimetry Program

4.1 *Reactor Physics Computational Mode*:

4.1.1 *Introduction*—This section provides a reference set of procedures for performing reactor physics calculations in experimental test reactors such as: Oak Ridge Pool Critical Assembly (PCA), Oak Ridge Bulk Shielding Reactor (BSR), Oak Ridge Research Reactor (ORR), University of Virginia Reactor, United Kingdom DIDO and PLUTO reactors, Belgium BR-2 reactor, F. R. Germany FRG1, FRG2, FRJ-1, and FRJ-2 reactors, The Netherlands JRC-HFR reactor, the Buffalo Nuclear Science and Technology Facility's reactor, etc. Although it is recognized that variations in methods will occur at various facilities, the present benchmarked calculational sequence has been used successfully in several studies **(1-4)**[7] and provides procedures for performing physics calculations in test reactors. Emphasis in these guidelines is placed on use of deterministic methods, but a short discussion of Monte Carlo techniques is also included.

4.2 *Determination of Core Fission Source Distribution*—The total fission source distribution, in source neutrons per unit volume per unit time, defined as:

$$S(x, y, z) = {}_0\int^{\infty} \nu(E)\Sigma_f(x, y, z, E)\cdot\phi(x, y, z, E)dE \tag{1}$$

where:

$\nu(E)$	=	number of neutrons per fission,
Σ_f	=	macroscopic fission cross section, and
ϕ	=	fluence rate.

is determined from a *k*-eigenvalue calculation of the reactor core, with the neutron fluence rate normalized to give the correct measured power output from the reactor, for example:

$$P = \int_E \int_V \kappa\Sigma_f(x, y, z, E)\phi(x, y, z, E)\cdot \mathrm{dxdydzd}E \tag{2}$$

where:

κ	=	effective energy yield per fission, and
P	=	experimentally determined thermal power with the integral calculated over all energies E and the core volume *v*.

4.2.1 An accurate value for the reactor power, *P*, is imperative for absolute comparison with experimental data.

4.2.2 If the axial core configuration is nonuniform, as might result from a partially inserted control rod, or from burnup effects, then a three-dimensional *k* calculation is required. This is usually calculated with a diffusion theory code such as VENTURE **(5)** or PDQ7 **(6)** using a few energy groups (<10). Some care must be exercised in averaging the few-group cross sections for the core calculation, and a general outline of the process is discussed at the end of this section.

4.2.3 Whenever the axial shape of the neutron fluence rate is separable from the shape in the other variables, then a full three-dimensional calculation is not required. In many experimental reactors, the axial dependence of the fluence rate is well approximated by a cosine shifted slightly from the midplane. In this case only a two-dimensional calculation (with a buckling approximation for axial leakage) is needed. In this case diffusion theory is usually used, but it is also possible to use two-dimensional transport theory if additional sophistication is required (for example, to obtain a more accurate treatment near control rods).

4.2.4 For reactor cores that generate a non-negligible amount of thermal power, the shape of the fission source may change with time due to burnup and changes in control rod positions. In this case, the source should be averaged over the time period during which the experiment was performed.

4.2.5 An important aspect of computing the fission source is using few-group cross sections that have been accurately weighted for the reactor configuration of interest. It is recommended that a fine-group cross-section library of approximately 100 groups with at least 10 thermal groups be used to generate the few-group set. Resonance shielding of the fine-group cross sections can be done with any of the methods acceptable for LWR analysis **(7)** (shielding factor, Nordheim, integral transport theory, etc.). The fine-group cross-section library shall be collapsed with weighting spectra obtained from cell calculations for each type of unit cell found in the core. If experiments are located near control rods or reflectors, then a separate calculation shall be performed for adjacent cells to account for the influence of these regions on the thermal spectrum in the experiment.

4.3 *Transport Calculations*:

4.3.1 It is recommended that a multi-dimensional (2D or 3D) discrete ordinates code such as DORT/TORT **(8)** or DANTSYS **(9)** be used for the transport theory calculations of both in-core and ex-core dosimeters. At least an S8 order quadrature with a P3 cross section expansion should be used. The space-dependent fission source from the core calculation is input as a volumetric distributed source with a fission spectrum

[7] The boldface numbers in parentheses refer to the list of references appended to this practice.