Designation: E1006 – 21

Standard Practice for Analysis and Interpretation of Physics Dosimetry Results from Test Reactor Experiments[1]

This standard is issued under the fixed designation E1006; 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.

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 E185, E853, and E1035, Guides E900, E2005, E2006 and Test Method E646. See also E706.

1.5 *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.*

1.6 *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.*

2. Referenced Documents

2.1 *ASTM Standards:*[2]

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

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)

E706 Master Matrix for Light-Water Reactor Pressure Vessel Surveillance Standards

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

E1005 Test Method for Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance

E1018 Guide for Application of ASTM Evaluated Cross Section Data File

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] 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.

E2005 Guide for Benchmark Testing of Reactor Dosimetry in Standard and Reference Neutron Fields
E2006 Guide for Benchmark Testing of Light Water Reactor Calculations

2.2 *Nuclear Regulatory Documents:*[3]

Code of Federal Regulations, "*Fracture Toughness Requirements*," Chapter 10, Part 50, Appendix G
Code of Federal Regulations, "*Reactor Vessel Materials Surveillance Program Requirements*," Chapter 10, Part 50, Appendix H
Regulatory Guide 1.99, Rev 2, "*Radiation Embrittlement of Reactor Vessel Materials*," U.S. Nuclear Regulatory Commission, May 1988

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 requires the knowledge of uncertainties in the prediction of radiation exposure parameters (for example, dpa (Practice E693), 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. 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**)[4] and provides procedures for performing physics calculations in test reactors. The Monte Carlo technique is used with about the same frequency as discrete ordinates techniques in test and research reactor dosimetry. The method is used more frequently in test/research reactors, as compared to power reactors, because of the very heterogeneous geometry often encountered in test/research reactors. Complex geometries can be handled in 3D space using the Monte Carlo approach.

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

$$S(x, y, z) = \int_0^\infty \nu(E)\sum_f(x, y, z, E)\cdot\varphi(x, y, z, E)dE \qquad (1)$$

where:

$\nu(E)$ = number of neutrons per fission,
$\sum_f$ = 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:

$$P = \int_E \int_V \kappa\sum\nolimits_f(x, y, z, E)\varphi(x, y, z, E)\cdot\mathrm{dxdydzd}E \qquad (2)$$

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 non-uniform, as might result from a partially inserted control rod, or from burnup effects, then a three-dimensional k calculation is required. Multigroup discrete ordinates or Monte Carlo methods are used almost exclusively to model the core (that is, not few group diffusion theory). This is particularly important where there are special purpose loops in the core or at a reflector/core boundary where the fluence spectrum changes very rapidly. In these cases, the few group diffusion models are typically not adequate.

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 it is possible to use two-dimensional transport theory.

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 If a few-group set is used to model the fission source distribution, 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 (**5**) (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

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

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

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-Discrete Ordinates Method:*

4.3.1 Transport calculations for test reactors may be performed by discrete ordinates or Monte Carlo methods, or by a combination of the two. The use of Monte Carlo codes is described in 4.5. If discrete ordinates methods are used, it is recommended that a multi-dimensional (2D or 3D) discrete ordinates code such as DORT/TORT (**6**), DANTSYS (**7**), or PARTISN (**8, 9**), be used for the transport theory calculations of both in-core and ex-core dosimeters. At least an S_8 order quadrature with a P_3 cross section expansion should be used. Because of significant spectrum changes that can occur over short distances in test reactor experiments, mesh spacing needs to be selected with care to ensure converged solutions at experiment locations. Detailed 3D discrete ordinates calculations will benefit from the use of a code that runs in parallel on multiple processors (**10, 11, 12**). The space-dependent fission source from the core calculation is input as a volumetric distributed source with a fission spectrum energy distribution. It is recommended that the ENDF/B-VII representation (**13**) of the ^{235}U thermal fission spectrum (MAT 9228, MF 5, MT 18), which is consistent with the ENDF/B Nuclear Data Standards for thermal neutrons (**14**) and based upon the latest experimental data for higher incident neutron energies (**15-17**), be used to represent the fission neutron energy distribution. This prompt fission neutron spectrum (PFNS) assumes that the build-in of other fissile isotopes with burnup is negligible. The latest applicable ENDF/B cross section data files shall be used (**13, 18**). If a three-dimensional discrete ordinates transport code is not used, it is recommended that the three-dimensional fluence rate distribution be synthesized from two two-dimensional calculations. A simple synthesis procedure that has been found to produce accurate results in benchmark dosimetry calculations is given in Refs (**2, 3**).

4.3.2 This synthesis procedure has been used successfully in a number of experiments in which the ex-core configuration is uniform axially along the full core height. For these types of problems, the three-dimensional synthesized fluence rates give dosimeter reactions that agree to within 10 % of the measured values, even off the core midplane. However, for experiments that contain short (relative to the core height) attenuating bodies, neutron streaming may occur around the edges of the body, and this effect is not well-predicted with the synthesis procedure. A "leakage iteration" procedure has been developed for such problems (**19**), but since most experiments do not experience this difficulty, it will not be discussed in this practice.

4.4 *Calculation of Bias Factors:*

4.4.1 In order to reduce the number of mesh intervals in the two-dimensional discrete ordinates calculations, it is often necessary to smear some detailed structure into a homogeneous mixture or completely ignore it. The experimental data computed with the homogeneous two-dimensional model can be corrected for the effects of local heterogeneities with bias factors. An example in which bias factors may be useful is in correcting for fluence rate perturbations caused by the experiment itself. This factor has been observed to be as high as 1.3 for a 1-in.^{2} container in an ex-core location. For in-core experiments the effects of heterogeneities within the experimental assembly should be examined.

4.4.2 Bias factors can be obtained with detailed one-dimensional (usually cylindrical) discrete ordinates calculations (**20**) in the vicinity of the desired data. Two cell calculations are usually done: one in which the experiment is modeled with as much detail as possible, and the other in which it is smeared in the same manner as in the two-dimensional calculation. In both the heterogeneous and homogeneous cases, the experiment zone should be surrounded by a homogenized zone corresponding to the same material which surrounds the experiment in the two-dimensional model. This region should be several mean free paths thick. It is recommended that the discrete ordinates calculations be performed as boundary source problems with an isotropic fluence rate boundary condition which is equal to the corresponding scalar fluence rate from the two-dimensional calculation. Group-dependent bias factors for the experiment zone are defined as the ratio of the group fluence rates for the heterogeneous and homogeneous geometries. These bias factors should multiply the multigroup fluence rates for the experiment zone in the two-dimensional calculation.

4.5 *Transport Calculations—Monte Carlo Method:*

4.5.1 While this practice permits the use of a discrete-ordinates technique for test reactor analysis (4.3), the alternative Monte Carlo technique may be preferred in many situations. This approach has the inherent advantage, over the deterministic method described in 4.3, of being able to treat three-dimensional aspects as well as geometrical complexity in explicit detail. Four Monte Carlo codes used for reactor analysis are MCNP (**21, 22, 23, 24**). MCBEND (**25, 26**), TRIPOLI (**27, 28**), and SERPENT (**29, 30**).

4.5.2 The Monte Carlo technique may be employed for the production of detailed core power distributions (for example, "eigenvalue" calculations).

4.5.3 A relevant restriction of Monte Carlo lies in the difficulty of calculating reaction rates at what are essentially "point" detectors, and some method or combination of methods employing variance reduction techniques must normally be used to modify the basic unbiased random sampling procedure. Such methods include, but are not limited to, use of a next-event estimator and of various "importance biasing" techniques involving splitting, Russian roulette, and path stretching as well as sampling from biased energy and angular distributions. In addition, an adjoint or "backward" calculation is sometimes preferable to the usual "forward" calculation, and all of the variance reduction techniques available in the forward calculation may, in principle, be used in the adjoint calculation as well.

4.5.4 A single Monte Carlo calculation generally provides information at only a few dosimeter locations due to Monte Carlo sampling uncertainty and the biasing techniques employed, whereas a deterministic calculation provides complete fluence rate information at all the geometric "points" in the model. Since the solution required is an absolute energy distribution of the fluence rate at each dosimeter location,

enough histories must be tracked to provide this differential information adequately for each detector location of interest. However, the loss of fluence rate information at other than these specific detector locations is not necessarily a severe shortcoming if the definition of "detector" is expanded to include several locations in the pressure vessel of interest in the embrittlement problem, even though no reaction rates may be available there.

4.5.5 Detailed three-dimensional Monte Carlo calculations in the adjoint mode have been used to benchmark a three-dimensional fluence rate procedure which combines the results of several less-dimensional discrete ordinates calculations:

$$\varphi\,(x, y, z) = \varphi(x, y)\varphi(y, z)/\varphi(y) \tag{3}$$

where:

x and z = transverse dimensions, and
y = dimension perpendicular to the core surface (radial dimension in cylindrical geometry).

4.5.5.1 The two methods agree within the statistical uncertainties of the Monte Carlo results (<5 %) for detectors located along the y-axis **(31).**

4.6 *Determination of Calculational Uncertainties:*

4.6.1 There is as yet no routine method to obtain the uncertainties in neutron transport calculations. A rigorous determination of variances and covariances requires a complete sensitivity analysis of the calculational procedures as it is done in the LEPRICON methodology **(32).** These methods are quite difficult and costly and may not be justified if simpler, though somewhat more conservative, uncertainty estimates lead to practically the same results. Benchmark testing, as recommended in Guide E482, gives a good indication for the size of the calculation errors and therefore provides a basis for the assignment of calculation variances. Bias factors, as discussed in 4.4, can also be used to estimate the variances introduced by the corresponding sources of systematic uncertainties. Covariances may be assigned according to the suggestions given in Guide E944.

4.6.2 If Monte Carlo calculations are used, variances and covariances associated with the statistical sampling in the calculations are directly incorporated. It is, however, necessary to take steps, for example, perturbation calculations, to address the variances and covariances due to cross section and modeling uncertainties.

4.6.3 Adjustment methods (see 4.8.3.3) provide a test for the consistency of the assigned calculation uncertainties with the rest of the input data.

4.7 *Dosimetry Experiment:*

4.7.1 *Purpose*—The dosimetry experiments provide the necessary data to verify the calculated fluence (or fluence rate) spectrum and to obtain estimates for the damage exposure and exposure rate values and their uncertainties.

4.7.2 Dosimetry experiments are performed in two different setups:

4.7.2.1 Dummy experiments using a mock-up of the metallurgical capsule containing only the dosimeters to be irradiated prior to the metallurgical experiment. This verifies and allows adjustments to the calculated fluence-spectrum results.

4.7.2.2 Metallurgical experiments containing in-situ dosimeters alongside the metallurgical specimen to be irradiated simultaneously. This allows the experimental determination of the needed exposure parameter values (fluence $E > 1.0$ and 0.1 MeV, dpa, etc.) with assigned uncertainties.

4.7.3 It is recommended to perform at least one dummy experiment for each series of associated metallurgical experiments. The advantage of the dummy experiment is that it allows greater latitude in the placement of dosimeters and the choice of irradiation time. Thus, a larger variety of dosimetry sensors may be used providing a more detailed determination of the fluence spectrum. However, in-situ dosimeters must also be placed in the metallurgical experiments to determine directly the fluence exposure to the metallurgical specimen.

4.7.4 Dosimeters used in both the dummy and metallurgical experiments are typically passive radiometric (foil) dosimeters. Other types of dosimeters (for example, solid state track recorders (SSTR), helium accumulation fluence monitors (HAFM), and damage monitors (DM)) should be added whenever appropriate. Situations may arise for longer irradiations where some radiometric dosimeters will be ineffective due to short half-life of the reaction product (see 4.7.5). There are two types of dosimeter sets that shall be used concurrently in each experiment.

4.7.4.1 *Multiple Foil (MF) Dosimeters*—The MFs contain a variety of sensor materials appropriately encapsulated and are primarily used to determine the energy dependence of the neutron spectra.

4.7.4.2 *Gradient Wires (GW)*—The GWs are dosimeters, generally in the form of wires that cover, in all directions to the largest extent possible, the dummy or metallurgical experiment in order to determine the spatial distribution of the neutron fluence. Typically, the ^{54}Fe(n,p) reaction (together with the ^{58}Fe(n,γ) reaction) is chosen for GW, but other reactions and more than one material may be used as appropriate.

4.7.5 Dosimetry sensors shall be chosen whose reaction cross sections match as closely as possible the response functions of the exposure parameters. The ^{237}Np(n,f) and ^{93}Nb(n,n') reactions are best suited for the determination of dpa. The ^{115}In(n,n') and ^{103}Rh(n,n') reactions have thresholds near 1.0 MeV and are therefore well suited for the determination of $\varphi > 1.0$ MeV. However, these two sensors can be used only in dummy experiments owing to the short half-life of the product isotopes. Two other important reactions are ^{238}U(n,f) and ^{54}Fe(n,p), but with responses above ~1 MeV and ~2 MeV, respectively. The addition of the HAFM reactions S(n, He), Ca(n, He), and N(n, He) could prove beneficial. Although experimental testing is still required, the available cross-section data for the latter three reactions indicate some low energy sensitivity. In addition, the reaction product, He, is stable, thus eliminating half-life corrections.

4.7.6 The other dosimetry sensors selected shall have response functions and threshold that are as diverse as possible in covering the neutron energy range of interest up to about 20 MeV. It has been reported that using least squares adjustment techniques, exposure parameter values can be obtained at dosimeter locations with estimated uncertainties in the range of 5 to 15 % (1σ) by using all three of the ^{237}Np(n, f), ^{238}U(n,f),