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Standard Practice for Analysis and Interpretation of Physics Dosimetry Results for Test Reactors, E 706(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).~~ 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 E 185, E 560, E 853, and E 1035, ~~Matrix E 706(IE), Guide~~ Guides E 900, E 2005E 2006and 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:*

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[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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E 185 Practice for Conducting Surveillance Tests for Light Water-Cooled Nuclear Power Reactor Vessels, E 706 (IF)[2,3]
E 482 Guide for Application of Neutron Transport Methods for Reactor Vessel Surveillance, E 706 (IID)[2,3]
E 560 Practice for Extrapolating Reactor Vessel Surveillance Dosimetry Results, E 706 (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), E 706 (ID)[2,3]
E 706 Master Matrix for Light-Water Reactor Pressure Vessel Surveillance Standards, E 706 (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, E 706 (IIC)[2,3]
E 853 Practice for Analysis and Interpretation of Light-Water Reactor Surveillance Results, E 706 (IA)[2,3]
E 854 Test Method for Application and Analysis of Solid State Track Recorder (SSTR) Monitors for Reactor Surveillance, E 706 (IIIB)[2,3]
E 900 Guide for Predicting Neutron Radiation Damage to Reactor Vessel Materials, E 706 (IIF)[2,3]
E 910 Specification for Application and Analysis of Helium Accumulation Fluence Monitors for Reactor Vessel Surveillance, E 706 (IIIC)[2,3]
E 944 Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance, E 706 (IIA)[2,3]
E 1005 Test Method for Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance, E 706 (IIIA)[2,3]
E 1018 Guide for Application of ASTM Evaluated Cross Section Data File, E 706 (IIB)[2,3]
E 1035 Practice for Determining Radiation Exposures for Nuclear Reactor Vessel Support Structures[2]
E 2005 Guide for the Benchmark Testing of Reactor Dosimetry in Standard and Reference Neutron Field, E 706 (IIE-I)[2,3]
E 2006 Guide for the Benchmark Testing of LWR Calculations, E 706 (IIE-2)[2,3]

2.2 *Nuclear Regulatory Documents:*

Code of Federal Regulations, "*Fracture Toughness Requirements*," Chapter 10, Part 50, Appendix G[5]

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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~~
[5] Available from Superintendent of ~~Matrix E 706.~~ Documents, U.S. Government Printing Office, Washington, DC 20402.

Code of Federal Regulations, "*Reactor Vessel Materials Surveillance Program Requirements*," Chapter 10, Part 50, Appendix H[5]

Regulatory Guide 1.99, Rev 2, "*Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials*," U.S. Nuclear Regulatory Commission, April 1977[5]

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.~~ 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)**[6] 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:

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

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:

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

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~~ may be 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).

~~[6] Available from Superintendent~~

[6] The boldface numbers in parentheses refer to the list of ~~Documents, U.S. Government Printing Office, Washington, DC 20402.~~ references appended to this practice.

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 energy distribution. It is recommended that the ENDF/B-VI representation **(10)** of the ^{235}U thermal fission spectrum (MAT 9228, MF 5, MT 18), which is based on the Madland-Nix formalism **(11)** be used to represent the fission neutron energy distribution. Even though the ENDF/B-IV and ENDF/B-V files have been shown to give generally acceptable results, the latest ENDF/B cross section data files shall be used. 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 **(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 **(12)**, 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 **(13)** 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 *Alternative Technique—Monte Carlo Calculation*:

4.5.1 While this practice recommends the use of a discrete-ordinates technique for test reactor analysis (4.3), the alternative Monte Carlo technique may be applied 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. Equally, there are drawbacks, and these have led to the limited use of Monte Carlo analysis in test reactor dosimetry in the U.S., although it has been used effectively in the United Kingdom **(14).** ~~The most common~~Two Monte Carlo codes used for reactor analysis are MCBEND **(15)** and TRIPOLI **(16)**.

4.5.2 Since the Monte Carlo technique is not usually employed for the production of detailed core power distributions (for example," eigenvalue" calculations), it would be necessary to initiate a dosimetry analysis, using this method, by way of a diffusion theory core-source calculation as described in 4.2. If a similar approach is taken from the transport calculation, the final results for both will be limited by the initial diffusion theory calculation of the core.

4.5.3 A more 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 provides information at only a few dosimeter locations, 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. The comparative cost of the alternative technique depends markedly on the codes used, the situation under investigation, and the scope of the problem, and is not easily defined.

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:

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

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 **(17).**

4.6 *Determination of Calculational Uncertainties*:

4.6.1 There is as yet no ~~generally accepted~~ 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 **(18).** 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 E 482, 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 E 944.

4.6.2 If Monte Carlo calculations are used, variances and covariances ~~for~~ associated with the statistical sampling in the calculations are obtained directly. It ~~may,~~ is, however, ~~be~~ necessary to add 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 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. ~~Such situations~~ 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 $^{58}Fe(n, \gamma)$ 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 $\phi > 1.0$ 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 ~~i~~ 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 $^{237}Np(n, f)$, $^{238}U(n, f)$, and $^{54}Fe(n, p)$ reactions; in the range of 10 to 20 % (1σ) by using the latter two reactions; and only in the range of 20 to 30 % (1σ) if the $^{54}Fe(n, p)$ reaction alone were to be used; see Refs (**14, 19,20**). It is recommended to use at least six different reactions for each MF set. Suitable sensors with associated thresholds and other pertinent information are discussed in Guide E 844, Specification E 910, and Test Methods E 1005 and ~~E 854, and Matrix E 706(IIID).~~ E 854. See also Refs (**14, 19, 20,21,22,23,24**) for typical MF sets and adjustment code results.

4.8 *Estimation of Neutron Exposure Parameters*:

4.8.1 Reports on the results of metallurgical irradiation experiments shall contain the estimates for the uncertainties in the determination of neutron exposure parameter values in the form of variances (or standard deviations) and covariances (or correlations). These data are necessary to perform reliable tests of damage models and to ensure consistency in data banks comprising large numbers of metallurgical experiments from test reactors. An excellent discussion of the uncertainties in neutron transport calculations of neutron exposure parameters can be found in Refs **(25)** and **(26)**.

4.8.2 ~~U~~Credible uncertainty data are very difficult to obtain from calculated spectra alone ~~with any degree of confidence~~ (see 4.6). The combination of calculations and appropriate dosimetry measurements by means of a least squares adjustment method greatly improves the values and reliability of uncertainty data as discussed in 4.7.5 (see Guides E 482, E 944, E 1018, E 2005, and ~~Matrix E 706(IIID)).~~ E 2006).

4.8.3 The application of a least squares adjustment method serves threefold purposes each of which is equally important:

4.8.3.1 Determination of the best (maximum likelihood or minimum variance) estimate for the damage exposure parameter values.

4.8.3.2 Determination of uncertainty bounds for these parameters.

4.8.3.3 Test for consistency for all input data.

4.8.4 Each of the determinations and tests in 4.8.3.1-4.8.3.3 shall be performed and reported as recommended in Guide E 944. State-of-the-art information on the development, testing, and application of adjustment methods is provided in Refs (~~**14, 19-24**~~ **18-26**).

5. Documentation

5.1 The documentation of test reactor physics-dosimetry results shall include the following items:

5.1.1 A complete spatial map of the exposure parameter values dpa, $\phi > 1.0$ MeV, $\phi > 0.1$ MeV (and others, if needed) including a scheme to interpolate between spatial mesh points.

5.1.2 Uncertainties of the exposure parameter values as explained in 4.8. (These uncertainties are expected to be in the range of 5 to 15 %, 1σ standard deviation, if appropriate dosimetry measurements have been performed. An explanation shall be provided if these values are exceeded in either direction).

5.1.3 Description of the methodology used including procedures for assigning input uncertainties.

5.2 The following information shall also be available in the form of an appendix for possible use in later reviews. At a very minimum, it shall be kept in archives if it is not included in the main report.

5.2.1 The documentation of all dosimeter sensor QA results, as-built dosimeters, dosimetry, capsules, irradiation test rig, and the replacement of dosimetry and metallurgy; including x, y, z, or r, θ, z coordinates for each dosimetry sensor and metallurgy specimen.

5.2.2 The documentation of the test reactor components, as-built core region and test region dimensions, materials, and irradiation history.

5.2.3 Nuclear data and constants used, raw measurement data, derived dosimetry sensor reactions and reaction rates, and auxiliary computations with intermediate results and verification procedures.

6. Keywords

6.1 discrete ordinates; dosimetry; Monte Carlo; neutron exposure parameters; radiation transport; test reactor

ANNEX

(Mandatory Information)

A1. METHODOLOGY FOR THE ANALYSIS AND INTERPRETATION OF PHYSICS-DOSIMETRY RESULTS FROM TEST REACTORS

A1.1 Establish a physics-dosimetry program in parallel with material irradiation experiments which are designed to correlate damage in test specimens with neutron exposure parameters, chemical composition, temperature, etc. This program includes the following steps:

A1.1.1 *Step 1*—Establish a reactor physics computational model to mock-up the reactor core and irradiation experiment. Typical reactor physics calculations can be divided into the following four parts:

A1.1.1.1 Determination of the absolute fission source distribution with a core criticality calculation for the expected reactor power.

A1.1.1.2 A transport theory calculation that uses the source obtained in A1.1.1.1 to determine absolute and relative neutron group fluence rates for the subsequent calculation of dosimetry sensor reactions and reaction rates for comparison with experimental data.

A1.1.1.3 Determination of any required bias factors to correct the group fluence rates fromA1.1.1.2 for localized heterogeneities.

A1.1.1.4 Calculation of absolute exposure rate parameters, such as fluence rate ($E > 1.0$ and 0.1 *MeV*) and dpa/s in iron or for damage monitors such as sapphire if they are to be used.

A1.1.1.5 Guidelines for calculations in A1.1.1.1 through A1.1.1.4 are presented. It is assumed that off-midplane measurements are taken so that three-dimensional results must be simulated. For experiments that can be modeled in one- or two-dimensional geometries, some of the procedures can be simplified.

A1.1.2 *Step 2*—Select, test, benchmark, and establish a least squares adjustment method that will provide physics-dosimetry derived exposure parameter values with statistical estimates of their uncertainties.

A1.1.3 *Step 3*—Establish and complete a dummy dosimetry experiment to obtain appropriate dosimetry sensor reactions and reaction rates to verify the fluence spectral computations and to supplement the input data for the subsequent application of the least squares adjustment method using the results of in-situ dosimetry from the materials irradiation experiments.

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