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Standard Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance[1]

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1. Scope

1.1 This guide covers the analysis and interpretation of the physics dosimetry for Light Water Reactor (LWR) surveillance programs. The main purpose is the application of adjustment methods to determine best estimates of neutron damage exposure parameters and their uncertainties.

1.2 This guide is also applicable to irradiation damage studies in research reactors.

1.3 *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.*

1.4 *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]

E170 Terminology Relating to Radiation Measurements and Dosimetry

E262 Test Method for Determining Thermal Neutron Reaction Rates and Thermal Neutron Fluence Rates by Radioactivation Techniques

E263 Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Iron

E264 Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Nickel

E265 Test Method for Measuring Reaction Rates and Fast-Neutron Fluences by Radioactivation of Sulfur-32

E266 Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Aluminum

E393 Test Method for Measuring Reaction Rates by Analysis of Barium-140 From Fission Dosimeters

E481 Test Method for Measuring Neutron Fluence Rates by Radioactivation of Cobalt and Silver

E482 Guide for Application of Neutron Transport Methods for Reactor Vessel Surveillance

E523 Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Copper

E526 Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Titanium

E693 Practice for Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacements Per Atom (DPA)

E704 Test Method for Measuring Reaction Rates by Radioactivation of Uranium-238

E705 Test Method for Measuring Reaction Rates by Radioactivation of Neptunium-237

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

E910 Test Method for Application and Analysis of Helium Accumulation Fluence Monitors for Reactor Vessel 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

E2005 Guide for Benchmark Testing of Reactor Dosimetry in Standard and Reference Neutron Fields

E2006 Guide for Benchmark Testing of Light Water Reactor Calculations

[1] This guide 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. A brief overview of Guide E944 appears in Master Matrix E706 in 5.4.1.

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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.

2.2 *Nuclear Regulatory Commission Documents:*[3]

NUREG/CR-1861 PCA Experiments and Blind Test

NUREG/CR-2222 Theory and Practice of General Adjustment and Model Fitting Procedures

NUREG/CR-3318 LWR Pressure Vessel Surveillance Dosimetry Improvement Program: PCA Experiments, Blind Test, and Physics-Dosimetry Support for the PSF Experiment

NUREG/CR-3319 LWR Power Reactor Surveillance Physics-Dosimetry Data Base Compendium

NUREG/CR-5049 Pressure Vessel Fluence Analysis and Neutron Dosimetry

2.3 *Electric Power Research Institute:*[4]

EPRI NP-2188 Development and Demonstration of an Advanced Methodology for LWR Dosimetry Applications

2.4 *Government Document:*[3]

NBSIR 85–3151 Compendium of Benchmark Neutron Fields for Reactor Dosimetry

3. Significance and Use

3.1 Adjustment methods provide a means for combining the results of neutron transport calculations with neutron dosimetry measurements (see Test Method E1005 and NUREG/CR-5049) in order to obtain optimal estimates for neutron damage exposure parameters with assigned uncertainties. The inclusion of measurements reduces the uncertainties for these parameter values and provides a test for the consistency between measurements and calculations and between different measurements (see 3.3.3). This does not, however, imply that the standards for measurements and calculations of the input data can be lowered; the results of any adjustment procedure can be only as reliable as are the input data.

3.2 *Input Data and Definitions:*

3.2.1 The symbols introduced in this section will be used throughout the guide.

3.2.2 Dosimetry measurements are given as a set of reaction rates (or equivalent) denoted by the following symbols:

$$a_i, \; i = 1,2, \ldots \tag{1}$$

These data are, at present, obtained primarily from radiometric dosimeters, but other types of sensors may be included (see 4.1).

3.2.3 The neutron spectrum (see Terminology E170) at the dosimeter location, fluence or fluence rate $\Phi(E)$ as a function of neutron energy E, is obtained by appropriate neutronics calculations (neutron transport using the methods of discrete ordinates or Monte Carlo, see Guide E482). The results of the calculation are customarily given in the form of multigroup fluences or fluence rates.

$$\Phi_j = \int_{E_j}^{E_{j+1}} \Phi\,(E)dE, \; j = 1,2, \ldots k \tag{2}$$

where:

E_j and E_{j+1} are the lower and upper bounds for the j-th energy group, respectively, and k is the total number of groups.

3.2.4 The reaction cross sections of the dosimetry sensors are obtained from an evaluated cross section file. The cross section for the i-th reaction as a function of energy E will be denoted by the following:

$$\sigma_i(E), \; i = 1,2, \ldots \tag{3}$$

Used in connection with the group fluences, Eq 2, are the calculated group-averaged cross sections σ_{ij}. These values are defined through the following equation:

$$\sigma_{ij} = \int_{E_j}^{E_{j+1}} \Phi(E)\sigma_i(E)dE/\Phi_j \tag{4}$$

$$i = 1,2, \ldots n;$$

$$j = 1,2, \ldots k$$

3.2.5 Uncertainty information in the form of variances and covariances must be provided for all input data. Appropriate corrections must be made if the uncertainties are due to bias producing effects (for example, effects of photo reactions).

3.3 *Summary of the Procedures:*

3.3.1 An adjustment algorithm modifies the set of input data as defined in 3.2 in the following manner (adjusted quantities are indicated by a tilde, for example, $\tilde{a}_i$):

$$\tilde{a}_i = a_i + \Delta a_i \tag{5}$$

$$\tilde{\Phi}(E) = \Phi(E) + \Delta\Phi(E) \tag{6}$$

or for group fluence rates

$$\tilde{\Phi}_j = \Phi_j + \Delta\Phi_j \tag{7}$$

$$\tilde{\sigma}_i(E) = \sigma_i(E) + \Delta\sigma_i(E), \tag{8}$$

or for group-averaged cross sections

$$\tilde{\sigma}_{ij} = \sigma_{ij} + \Delta\sigma_{ij} \tag{9}$$

The adjusted quantities must satisfy the following conditions:

$$\tilde{a}_i = \int_0^\infty \tilde{\Phi}(E)\tilde{\sigma}_i(E)dE, \; i = 1,2, \ldots \text{n} \tag{10}$$

or in the form of group fluence rates

$$\tilde{a}_i = \sum_{j=1}^{k} \tilde{\sigma}_{ij}\tilde{\Phi}_j, \; i = 1,2, \ldots \text{n} \tag{11}$$

Since the number of equations in Eq 11 is much smaller than the number of adjustments, there exists no unique solution to the problem unless it is further restricted. The mathematical algorithms in current adjustment codes are intended to make the adjustments as small as possible relative to the uncertainties of the corresponding input data. Codes like STAY'SL, FERRET, LEPRICON, and LSL-M2 (see Table 1) are based explicitly on the statistical principles such as "Maximum Likelihood Principle" or "Bayes Theorem," which are generalizations of the well-known least squares principle, and are taking into account variances and correlations of the input fluence, dosimetry, and cross section data (see 4.1.1, 4.2.2, and 4.3.3). A detailed discussion of the mathematical derivations can be found in NUREG/CR-2222 and EPRI NP-2188. Even

[3] Available from U.S. Government Printing Office, Superintendent of Documents, 732 N. Capitol St., NW, Washington, DC 20401-0001, http://www.access.gpo.gov.

[4] Available from the Electric Power Research Institute, 3420 Hillview Avenue, Palo Alto, California 94304, http://www.epri.com.

TABLE 1 Available Neutron Spectrum Adjustment and Unfolding Codes

Program	Solution Method	Code Available From	References	Comments
SAND-II	semi-iterative	RSICC Prog. No. CCC-112, CCC-619, PSR-345	(**1**, **2**)[A]	contains trial spectra library. No output uncertainties in the original code, but modified Monte Carlo code provides output uncertainties (**3**)
SPECTRA	statistical, linear estimation	RSICC Prog. No. CCC-108	(**4**, **5**)	minimizes deviation in magnitude, no output uncertainties.
IUNFLD/ UNFOLD	statistical, linear estimation		(**6**)	constrained weighted linear least squares code using B-spline basic functions. No output uncertainties.
WINDOWS	statistical, linear estimation, linear programming	RSICC Prog. No. PSR-136, 161	(**7**)	minimizes shape deviation, determines upper and lower bounds for integral parameter and contribution of foils to bounds and estimates. No statistical output uncertainty.
RADAK, SENSAK	statistical, linear estimation	RSICC Prog. No. PSR-122	(**8**, **9**, **10**, **11**)	RADAK is a general adjustment code not restricted to spectrum adjustment.
STAY'SL	statistical linear estimation	RSICC Prog. No. PSR-113	(**12**)	permits use of full or partial correlation uncertainty data for activation and cross section data.
NEUPAC(J1)	statistical, linear estimation	RSICC Prog. No. PSR-177	(**13**, **14**)	permits use of full covariance data and includes routine of sensitivity analysis.
FERRET	statistical, least squares with log normal a priori distributions	RSICC Prog. No. PSR-145	(**15**, **16**)	flexible input options allow the inclusion of both differential and integral measurements. Cross sections and multiple spectra may be simultaneously adjusted. FERRET is a general adjustment code not restricted to spectrum adjustments.
LEPRICON	statistical, generalized linear least squares with normal a priori and a posteriori distributions	RSICC Prog. No. PSR-277	(**17**, **18**, **19**)	simultaneous adjustment of absolute spectra at up to two dosimetry locations and one pressure vessel location. Combines integral and differential data with built-in uncertainties. Provides reduced adjusted pressure vessel group fluence covariances using built-in sensitivity database.
LSL-M2	statistical, least squares, with log normal a priori and a posteriori distributions	RSICC Prog. No. PSR-233	**20**	simultaneous adjustment of several spectra. Provides covariances for adjusted integral parameters. Dosimetry cross-section file included.
UMG	Statistical, maximum entropy with output uncertainties	RSICC Prog. No. PSR-529	(**21**, **22**)	Two components. MAXED is a maximum entropy code. GRAVEL (**23**) is an iterative code.
NMF-90	Statistical, least squares	IAEA NDS	(**24**, **25**)	Several components, STAY'NL, X333, and MIEKE. Distributed by IAEA as part of the REAL-84 interlaboratory exercise on spectrum adjustment (**26**).
GMA	Statistical, general least squares	RSICC Prog. No. PSR-367	(**27**)	Simultaneous evaluation with differential and integral data, primarily used for cross-section evaluation but extensible to spectrum adjustments.

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

the older codes, notably SAND-II and CRYSTAL BALL, apply a minimization algorithm although the statistical assumptions are not spelled out explicitly in the supporting documentation. Table 1 lists some of the available unfolding codes; however, the first four codes listed: SAND-II, SPECTRA, IUNFLD/ UNFOLD, and WINDOWS have severe limitations in that they do not typically provide uncertainty characterization of the resulting unfolded spectrum and the adjusted damage exposure parameters.

3.3.1.1 An important problem in reactor surveillance is the determination of neutron fluence inside the pressure vessel wall at locations which are not accessible to dosimetry. Estimates for exposure parameter values at these locations can be obtained from adjustment codes which adjust fluences simultaneously at more than one location when the cross correlations between fluences at different locations are given. LEPRICON has provisions for the estimation of cross correlations for fluences and simultaneous adjustment. LSL-M2 also allows simultaneous adjustment, but cross correlations must be given.

3.3.2 The adjusted data $\tilde{a}_i$, etc., are, for any specific algorithm, unique functions of the input variables. Thus, uncertainties (variances and covariances) for the adjusted parameters can, in principle, be calculated by propagation the uncertainties for the input data. Linearization may be used before calculating the uncertainties of the output data if the adjusted data are nonlinear functions of the input data.

3.3.2.1 The algorithms of the adjustment codes tend to decrease the variances of the adjusted data compared to the corresponding input values. The linear least squares adjustment codes yield estimates for the output data with minimum variances, that is, the "best" unbiased estimates. This is the primary reason for using these adjustment procedures.

3.3.3 Properly designed adjustment methods provide means to detect inconsistencies in the input data which manifest themselves through adjustments that are larger than the corresponding uncertainties or through large values of chi-square, or both. (See NUREG/CR-3318 and NUREG/CR-3319.) Any detection of inconsistencies should be documented, and output data obtained from inconsistent input should not be used. All input data should be carefully reviewed whenever inconsistencies are found, and efforts should be made to resolve the inconsistencies as stated below.

3.3.3.1 Input data should be carefully investigated for evidence of gross errors or biases if large adjustments are required. Note that the erroneous data may not be the ones that required the largest adjustment; thus, it is necessary to review all input data. Data of dubious validity may be eliminated if proper corrections cannot be determined. Any elimination of data must be documented and reasons stated which are independent of the adjustment procedure. Inconsistent data may also be omitted if they contribute little to the output under investigation.

3.3.3.2 Inconsistencies may also be caused by input variances which are too small. The assignment of uncertainties to the input data should, therefore, be reviewed to determine whether the assumed precision and bias for the experimental and calculational data may be unrealistic. If so, variances may be increased, but reasons for doing so should be documented. Note that in statistically based adjustment methods, listed in Table 1 the output uncertainties are determined only by the input uncertainties and are not affected by inconsistencies in the input data (see NUREG/CR-2222). Note also that too large adjustments may yield unreliable data because the limits of the linearization are exceeded even if these adjustments are consistent with the input uncertainties.

3.3.4 Using the adjusted fluence spectrum, estimates of damage exposure parameter values can be calculated. These parameters are weighted integrals over the neutron fluence

$$p = \int_o^\infty \tilde{\Phi}(E) w(E) dE \tag{12}$$

or for group fluences

$$p = \sum_{j=1}^{k} \tilde{\Phi}_j w_j \tag{13}$$

with given weight (response) functions $w(E)$ or w_j, respectively. The response function for dpa of iron is listed in Practice E693. Fluence greater than 1.0 MeV or fluence greater than 0.1 MeV is represented as $w(E) = 1$ for E above the limit and $w(E) = 0$ for E below.

3.3.4.1 Finding best estimates of damage exposure parameters and their uncertainties is the primary objective in the use of adjustment procedures for reactor surveillance. If calculated according to Eq 12 or Eq 13, unbiased minimum variance estimates for the parameter p result, provided the adjusted fluence $\Phi^{\sim}$ is an unbiased minimum variance estimate. The variance of p can be calculated in a straightforward manner from the variances and covariances of the adjusted fluence spectrum. Uncertainties of the response functions, w_j, if any, should not be considered in the calculation of the output variances when a standard response function, such as the dpa for iron in Practice E693, is used. The calculation of damage exposure parameters and their variances should ideally be part of the adjustment code.

4. Selection of Input Data

4.1 *Sensor Sets:*

4.1.1 Radiometric Measurements (*RM*)—This is at present the primary source for dosimetry data in research and power reactors. RM sensor selection, preparation, and measurement, including determination of variances and covariances, should be made according to Guide E844 and the standards describing the handling of the particular foil material (Test Methods E262, E263, E264, E265, E266, E393, E481, E523, E526, E704, and E705). Other passive dosimetry sensors of current interest in research and power reactors and in ex-vessel environments are solid state track recorders (SSTR), helium accumulation fluence monitors (HAFM), and damage monitors (DM). Use of these sensors is described in separate ASTM standards as follows:

4.1.2 *SSTR*—see Test Method E854.

4.1.3 *HAFM*—see Test Method E910.

4.1.4 The preceding list does not exclude the use of other integral measurements, for example, from fission chambers or nuclear emulsions (see NUREG/CR-1861).

4.1.5 Accurate dosimetry measurements and proper selections of dosimetry sensors are particularly important if the uncertainties in the calculated spectrum are large (see Ref **28**). In this case, it is necessary either to have several dosimetry sensors which respond to various parts of the neutron energy range of interest or to utilize a sensor which closely approximates the energy response of the damage exposure parameters. Since determination of a variety of damage exposure parameters is desirable, some combination of dosimeter responses is usually necessary to achieve the smallest possible output uncertainties. Reactions currently used which are regarded as providing the best overlap with the iron dpa cross section are ^{237}Np(n,f) and ^{93}Nb(n,n')^{93m}Nb. Other reactions used to measure neutrons above 1 MeV are ^{63}Cu(n,α), ^{46}Ti(n,p), ^{54}Fe(n,p), ^{58}Ni(n,p), and ^{238}U(n,f). (See Practice E853.) If the calculated spectrum has small uncertainties, the requirements of good spectral coverage or good overlap with damage response are not as critical, but redundant dosimetry is still recommended to minimize chances of erroneous results. (See Refs **28, 29**.)

4.1.6 Non-threshold sensors such as ^{235}U(n,f), ^{239}Pu(n,f), and all (n,γ) reactions are frequently used. These detectors have the highest sensitivity at low neutron energies (below 1 keV) and are useful for the validation of calculated spectra in the low energy range and for the estimation of effects caused by low energy neutrons (for example, ^{235}U fission and ^{239}Pu fission in ^{238}U, etc.). They are not as important as the threshold reactions for the determination of damage exposure parameters values but can serve as useful supplements, particularly in the determination of iron dpa (see Ref **28**).

4.1.7 The number of reactions used in an adjustment procedure need not be large as long as the energy range under investigation is adequately covered. A small number of well-established dosimetry sensors combined with high-quality measuring procedures is preferable to a large number of measurements which include inconsistent or irrelevant data.