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Standard Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance, E 706 (IIA)[1]

This standard is issued under the fixed designation E 944; 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 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 and health practices and determine the applicability of regulatory limitations prior to use.*

2. Referenced Documents

2.1 *ASTM Standards:*

E 170 Terminology Relating to Radiation Measurements and Dosimetry[2]

E 262 Test Method for Determining Thermal Neutron Reaction and Fluence Rates by Radioactivation Techniques[2]

E 263 Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Iron[2]

E 264 Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Nickel[2]

E 265 Test Method for Measuring Reaction Rates and Fast-Neutron Fluences by Radioactivation of Sulfur-32[2]

E 266 Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Aluminum[2]

E 343 Test Method for Measuring Reaction Rates by Analysis of Molybdenum-99 Radioactivity from Fission Dosimeters[2]

E 393 Test Method for Measuring Reaction Rates by Analysis of Barium-140 from Fission Dosimeters[2]

E 481 Test Method for Measuring Neutron Fluence Rate by Radioactivation of Cobalt and Silver[2]

E 482 Guide for Application of Neutron Transport Methods for Reactor Vessel Surveillance, E 706(IID)[2,3]

E 523 Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Copper[2]

E 526 Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Titanium[2]

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 704 Test Method for Measuring Reaction Rates by Radioactivation of Uranium-238[2]

E 705 Test Method for Measuring Reaction Rates by Radioactivation of Neptunium-237[2]

E 706 Master Matrix for Light-Water Reactor Pressure Vessel Surveillance Standards[2]

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 910 Test Method for Application and Analysis of Helium Accumulation Fluence Monitors for Reactor Vessel Surveillance, E706(IIIC)[2,3]

E 1005 Test Method for Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance, E 706(IIIA)[2]

E 1018 Guide for Application of ASTM Evaluated Cross Section Data File, E706(IIB)[2,3]

E 2005 Guide for the Benchmark Testing of Reactor Dosimetry in Standard and Reference Neutron Field, E706 (IIE-I)[2]

E 2006 Guide for the Benchmark Testing of LWR Calculations E706 (IIE-2)[2,3]

2.2 *Nuclear Regulatory Commission Documents:*[4]

NUREG/CR-1861 PCA Experiments and Blind Test

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

[1] This guide is under the jurisdiction of ASTM Committee E10 on Nuclear Technology and Applicationsand is the direct responsibility of Subcommittee E10.05on Nuclear Radiation Metrology. A brief overview of Guide E 944 appears in Master Matrix E 706 in 5.3.1(IIA).

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

[3] The unnumbered references in parentheses refer to Section 5 as well as Figs. 1 and 2 of Matrix E 706.

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

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:*[5]

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

2.4 *Government Document:*[4]

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 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:

$$a_i, i = 1,2, ... \tag{1}$$

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 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 E 482). The results of the calculation are customarily given in the form of k group fluences or fluence rates.

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

where:

E_j and E_{j+1} are the lower and upper bounds for the j-th energy group, respectively.

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:

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

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:

$$\sigma_{ij} = \int_{E_j}^{E_{j+1}} \Phi(E)\sigma_i(E)dE/\Phi_j$$
$$i = 1,2, . . n;$$
$$j = 1,2, ... k \tag{4}$$

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, $\tilde{a}_i$):

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

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

or for group fluence rates

$$\Phi_j = \Phi_j + \Delta\Phi_j \tag{7}$$

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

or for group-averaged cross sections

$$\sigma_{ij} = \sigma_{ij} + \Delta\sigma_{ij} \tag{9}$$

The adjusted quantities must satisfy the following conditions:

$$\tilde{a}_i = \int_0^\infty \Phi(E)\sigma_i(E)dE, i = 1,2, ... n \tag{10}$$

or in the form of group fluence rates

$$\tilde{a}_i = \sum_{j=1}^{k} \sigma_{ij}\Phi_j, i = 1,2, ... n \tag{11}$$

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 algorithm in current adjustment codes are intended to make the adjustments as small as possible relative to the uncertainties of the corresponding input data. The more recent 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. Using variances and correlations of the input fluence, dosimetry, and cross section data (see 4.1.1, 4.2.2, and 4.3.3), even the older codes, notably SAND-II and CRYSTAL BALL, can be interpreted as application of the least squares principle although the statistical assumptions are not spelled out explicitly (see Table 1). A detailed discussion of the mathematical derivations can be found in NUREG/CR-2222 and EPRI NP-2188.

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

[5] Available from the Electric Power Research Institute, P. O. Box 10412, Palo Alto, CA 94303.

TABLE 1 Available Unfolding Codes

Program	Solution Method	Code Available From	References	Comments
SAND-II	semi-iterative	RSIC Prog. No. CCC-112, CCC-619, PSR-345	**1**	contains trial spectra library. No output uncertainties in the original code, but modified Monte Carlo code provides output uncertainties (**12, 22**)
SPECTRA	statistical, linear estimation	RSIC Prog. No. CCC-108	**2, 3**	minimizes deviation in magnitude, no output uncertainties.
IUNFLD/ UNFOLD	statistical, linear estimation		**5**	constrained weighted linear least squares code using B-spline basic functions. No output uncertainties.
WINDOWS	statistical, linear estimation, linear programming	RSIC Prog. No. PSR-136, 161	**6**	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		**7, 8**	RADAK is a general adjustment code not restricted to spectrum adjustment.
STAY'SL	statistical linear estimation	RSIC Prog. No. PSR-113	**9**	permits use of full or partial correlation uncertainty data for activation and cross section data.
NEUPAC(J1)	statistical, linear estimation	RSIC Prog. No. PSR-177	**10, 11**	permits use of full covariance data and includes routine of sensitivity analysis.
FERRET	statistical, least squares with log normal a priori distributions	RSIC Prog. No. PSR-145	**12, 22**	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	RSIC Prog. No. PSR-277	**14, 15, 18**	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	RSIC Prog. No. PSR-233	**19**	simultaneous adjustment of several spectra. Provides covariances for adjusted integral parameters. Dosimetry cross-section file included.

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. 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

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

or for group fluences

$$p = \sum_{j=1}^{k} \Phi_j w_j \tag{13}$$

with given weight (response) functions $w(E)$ or w_j, respectively. The response function for dpa of iron is listed in Practice E 693. 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 Φ 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 E 693, is used. The calculation of damage exposure parameters and their variances when a standard response 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 E 844, weight (response), and the standards describing the handling of the particular foil material (Test Methods E 262, E 263, E 264, E 265, E 266, E 343, E 393, E 481, E 523, E 526, E 704, and E 705). 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 E 854.

4.1.3 *HAFM*—see Test Method E 910.

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 (**20**)[5,6]). 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 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 E 853.) 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 (**20, 21**).)

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, U-235 fission and Pu-239 fission in U-238, 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 dpa (see Ref (**20**)).

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.

4.2 *Calculations*:

4.2.1 Neutron transport calculations of the input spectrum for the analysis of reactor surveillance should follow the guidelines set forth in Guide E 482. The sources of uncertainties and errors in the calculation should be investigated and variances should be assigned accordingly. Results from benchmark validations may also be used to estimate the variances (see NUREG/CR-1861).

4.2.1.1 The auto correlations for fluences may be assigned as described in 5.3 if no other information is available (see 4.2.2). The procedure used for assigning variances and covariances to input fluences should be documented.

4.2.2 The most satisfactory procedure for assigning variances and covariances to calculated fluences is a complete sensitivity analysis as described in EPRI NP-2188 and Refs (**13**) and (**14**). This method requires a large amount of calculations. It is expected, however, that the results of one calculation can be extended by way of analogy to a larger class of sufficiently similar cases (see Ref (**16, 17**)).

4.2.3 Benchmark neutron spectra can be included as simultaneous input in some codes if the dosimetry measurements are benchmark referenced (see Guides E 2005 and E 2006 and NUREG/CR-1861 and NBSIR 85-3151). Fluence rates with variances and covariances are available from the appropriate benchmark compendia.

4.2.4 Some adjustment codes allow for scaling of the calculated neutron spectrum if the accurate normalization of the calculation to the proper source strength is difficult to accomplish. This can also be accomplished by constructing a fluence covariance matrix in which a common scale term with large variance is superimposed on the original covariance matrix as described in 5.3.3. However, arbitrary scaling should be avoided in power reactor applications where the source core information is available from measurements during operation.

4.2.5 The number of independent adjustments of the input spectrum is limited by the number of different reactions used, and this is further restricted if the reactions have similar cross sections. The number of energy groups in the input spectrum need therefore not exceed significantly the number of different detectors. The smaller the number of group fluence rates, the easier and less critical is the assignment of autocorrelations (see 5.3.2). A too small number of energy groups may, however, increase the uncertainties in determining group-averaged cross sections and integral parameter values and also impose artificial correlation between energies within broad

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

groups. The group boundaries should, for the same reason, be well matched to the thresholds of the detectors. A number of energy groups between 15 and 30 appears to be a practical compromise, but some analysts have reported good results using 50 or more groups.

4.2.6 Spectrum libraries are available in some older unfolding codes like SAND-II. Library spectra are not recommended as input for adjustment procedures and should be used only if a calculation of the spectrum cannot be performed. A properly selected library spectrum may be adequate for the determination of damage parameters if the damage response region is sufficiently covered by dosimetry measurements. No library spectrum should be used which is grossly inconsistent with the dosimetry data. (Spectrum adjustments should not exceed ±50 % maximum or ±20 % in the average.) It may be advisable to try several input spectra to investigate the influence of the input spectrum on the estimated damage parameter. Large variances (>50 %) should be assigned to library spectra.

4.3 *Cross Section Sets*:

4.3.1 It is recommended to use evaluated cross section files with uncertainties as described in Guide E 1018 whenever possible.

4.3.2 The group-averaged cross sections σ_{ij} depend, according to formula (3.4), on the shape of the continuous spectrum $\Phi(E)$. Dosimetry cross section files are presently available in a 640 energy group structure, and the input spectrum needs to be expanded to this structure in order to obtain a condensed cross section set. This is done by means of a weighting spectrum, preferably the one used for fine-group calculations of the neutron spectrum (see Guide E 482). A standard weighting spectrum, such as fission spectrum, l/E spectrum, or Maxwellian in the appropriate energy region, may also be used. The expansion of the input spectrum may introduce additional uncertainties in the group-averaged cross sections, and the variances should be increased accordingly. Experience has shown, however, that the group-averaged cross sections σ_{ij} are relatively insensitive to changes in the weighting spectrum $\Phi(E)$; significant changes are observed only if both the spectrum $\Phi(E)$ and the cross section $\sigma_i(E)$ have large resonances or structure in the same interval. It is permissible in many cases to use one set of group-averaged cross sections for different, but sufficiently similar, spectra (for example, all spectra in surveillance locations in LWR's).

4.3.3 Variances and covariances for cross section data are included in recent data files following a format described in Ref **(24)**. The present data are somewhat artificial in that complete autocorrelation is assumed within stated energy ranges. It may also be noted that, as shown in NUREG/CR-2222, the amount of spectrum adjustment depends not on the variances and covariances of the group cross sections individually, but rather on their total contribution over the whole energy range. It is therefore permissible to approximate the autocorrelation from the cross section data files by the simpler representations described in 5.3.2.

4.3.3.1 There are at present very few correlations listed in data files for cross sections between different reactions. If essentially the same cross section applies to two or more different dosimetry measurements, as in bare and cadmium covered foils of the same material, the corresponding correlations can be obtained by combining common and individual variances as outlined in 5.3.3.

5. Selection and Use of Computer Codes

5.1 *General Considerations and Properties of Existing Codes*:

5.1.1 All presently used adjustment procedures are based on the principles of 3.3.1. A list of available codes is given in Table 1. In the older codes some or all of the input variance and covariance data are preselected as part of the algorithm and cannot be changed. These codes have also no provision for output uncertainties. The use of these codes is discouraged (see Table 1).

5.1.2 Neutron spectrum adjustment is a nonlinear problem if all input data are adjusted. Most adjustment codes apply a linearization in order to apply the simple and reliable linear least-squares algorithm. This may result in slight deviations from strict consistency as required by Eq 10 or Eq 11, but these deviations are mostly negligible and do not significantly disturb the output values relative to the unbiased minimum variance estimates. Large deviations occur only in connection with large adjustments which should be avoided in any adjustment procedure.

5.1.2.1 Iterative procedures are used in some adjustment algorithm to satisfy strictly Eq 10 and Eq 11. Iterations are valid only if they solve the original nonlinear least-squares problem as described, for instance, in NUREG/CR-2222.

5.1.3 Some codes, like WINDOWS, use linear programming procedures to obtain upper and lower bounds for integral parameters instead of variances and covariances. These bounds are based solely on the requirement that the fluence spectrum must be positive and not on the difficult-to-estimate variance of the input spectrum. These bounds are rather conservative, however.

5.2 *Requirements for the Use of Adjustment Codes in Reactor Surveillance*:

5.2.1 The following requirements must be satisfied to qualify an adjustment method for use in the analysis of surveillance dosimetry or similar critical applications:

5.2.1.1 The adjustment procedure must be based explicitly on a method for statistical estimation. Assumptions about statistical distributions (normal, log-normal, or other) must be clearly documented. The adjusted output should be an unbiased minimum variance estimate (based on linearization of the problem).

5.2.1.2 A test of the chosen adjustment code on some benchmark problems is valuable, but this test cannot replace a direct verification of the mathematical algorithm.

5.2.1.3 The adjustment code must provide the adjusted values for damage exposure (integral) parameters, together with variances and covariances.

5.2.1.4 The adjustment code must accommodate the input variances and covariances of all input measurements and calculations. Simplifications of correlation matrices are permissible as stated in Section 4 and 5.3.

5.2.1.5 Individual adjustments should be listed to facilitate the detection of inconsistencies in the input data. Results of a Chi–square test should also be indicated.

5.3 *Assignment of Correlations*:

5.3.1 Information about correlations in cross section data is often incomplete and rather tentative. However, correlations cannot be ignored since they may have a significant effect on the result of adjustment procedures. If a direct determination as in EPRI NP–2188 cannot be performed, a simplified model for the assignment of correlations is recommended which will suffice for some applications.

5.3.2 For auto-correlation (that is, correlations between different energy groups) of fluence or cross sections the assumption is usually made that all correlations are positive and decrease with increasing distance between the energy groups. This assumption assures some degree of smoothness of the adjusted fluences or cross sections as a function of energy. To realize this concept in a mathematical model, a distance function is defined which assigns a numerical value to the distance between two given energy groups. These distances may be expressed as the differences in mean energy, mean lethargy, or simply the group number, scaled with a suitable scaling factor to introduce the desired amount of smoothing. Let d (a,b) represent the distance between energy group a and group b assigned for the auto-correlation between group fluences Φ_a and Φ_b. The correlation between Φ_a and Φ_b is then expressed as a function of d (a,b):

$$\text{cor}\,(\Phi_a,\Phi_b) = f\,[d(a,b)],\ -1 \le f(x) \le 1 \tag{14}$$

The function f (x) must satisfy the condition that the covariance matrix is positive definite. This condition is satisfied if f (x) is a cosine Fourier transform of a positive function f (w):

$$f(x) = \int_0^\infty f(w)\cos wx\, dw,\ f(w) > 0,\ \int_0^\infty f(w)dw = 1 \tag{15}$$

Suitable functions are as follows:

$$f(x) = e^{-x^2} \tag{16}$$

$$f(x) = e^{-1\times 1} \tag{17}$$

Using these definitions, the covariance matrices can be calculated according to the formula

$$\text{cov}\,(\Phi_a,\Phi_b) = \text{cor}\,(\Phi_a,\Phi_b)\sqrt{\text{var}\,(\Phi_a)\ \text{var}\,(\Phi_b)} \tag{18}$$

instead of storing the full covariance matrices.

5.3.2.1 Function (5.4) is somewhat easier for the calculation since the correlation between any two groups is the product of the correlations between neighboring intervening groups. Function (5.3) decreases faster so that non-negligible correlations extend only to close neighbors.

5.3.3 Another type of correlation occurs if two or more measurements have a common source of error in addition to individual errors, all of which are mutually uncorrelated. Let

$$x = a + \alpha c \tag{19}$$
$$y = b + \beta c$$

with individual variances σ_a^2 and σ_b^2 for a and b, respectively, σ_c^2 for the common term c. The variances and covariances for x and y are then:

$$\text{cov}\,(x,x) = \sigma_a^2 + \alpha^2\sigma_c^2 \tag{20}$$
$$\text{cov}\,(x,y) = \alpha\beta\sigma_c^2$$
$$\text{cov}\,(y,y) = \sigma_b^2 + \beta^2\sigma_c^2$$

5.4 *Output Uncertainties*—of damage exposure values depend on the accuracy of the fluence calculation and dosimetry measurements and on the selection of dosimetry sensors (see Ref (**20**)). The achievable accuracy depends on the neutron field under investigation. Assuming due care in calculations and measurements, the following output variances can be expected for the damage exposure parameters $\Phi > 1.0$ MeV, $\Phi > 0.1$ MeV, and dpa (1σ):

Benchmark Fields - 5 % (see NUREG/CR-1861)
Research Reactors 5 – 15 % (see Ref (**20**))
Power Reactors 5 – 30 % (see Ref (**21**))

The quoted uncertainties are for dpa as damage exposure parameter. The uncertainties for $\Phi > 1.0$ MeV are slightly lower and for $\Phi > 0.1$ MeV slightly higher given the same input data.

6. Keywords

6.1 dosimetry; exposure parameters; irradiation damage; least squares; neutron; reactor surveillance; spectrum adjustment

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