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

This standard is issued under the fixed designation E944; 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.

~~ε[1] NOTE—The title of this guide and the Referenced Documents were updated editorially in May 2017.~~

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~~ safety, health, and ~~health~~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

[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. ~~DOI: 10.1520/E0944-08.~~

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

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:

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

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, $\tilde{a}_i$):

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

[3] Available from ~~Superintendents of Documents, U. S.~~ U.S. Government Printing Office, ~~Washington, DC 20402.~~ 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, ~~P. O. Box 10412,~~ 3420 Hillview Avenue, Palo Alto, ~~CA 94303.~~ California 94304, http://www.epri.com.

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

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:

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

or in the form of group fluence rates

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

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~~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. Using~~ 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~~), even the~~). A detailed discussion of the mathematical derivations can be found in NUREG/CR-2222 and EPRI NP-2188. Even the older codes, notably SAND-II and CRYSTAL BALL, ~~can be interpreted as application of the least squares principle~~ apply a minimization algorithm although the statistical assumptions are not spelled out explicitly ~~(see~~in ~~Table 1). A detailed discussion~~the supporting documentation. Table 1 ~~of the mathematical derivations can be found in NUREG/CR-2222 and EPRI NP-2188.~~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.

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~~[A]	~~contains trial spectra library. No output uncertainties in the original code, but modified Monte Carlo code provides output uncertainties (2, 3, 4)~~
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~~	~~5, 6~~	~~minimizes deviation in magnitude, no output uncertainties.~~
SPECTRA	statistical, linear estimation	RSICC Prog. No. CCC-108	(4, 5)	minimizes deviation in magnitude, no output uncertainties.
~~IUNFLD/UNFOLD~~	~~statistical, linear estimation~~		~~7~~	~~constrained weighted linear least squares code using B-spline basic functions. 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~~	~~8~~	~~minimizes shape deviation, determines upper and lower bounds for integral parameter and contribution of foils to bounds and estimates. No statistical output uncertainty.~~
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~~	~~9, 10,11,12~~	~~RADAK is a general adjustment code not restricted to spectrum adjustment.~~
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~~	~~13~~	~~permits use of full or partial correlation uncertainty data for activation and cross section data.~~
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~~	~~14, 15~~	~~permits use of full covariance data and includes routine of sensitivity analysis.~~
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~~	~~2, 3~~	~~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.~~
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~~	~~16, 17, 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.~~
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~~	~~19~~	~~simultaneous adjustment of several spectra. Provides covariances for adjusted integral parameters. Dosimetry cross-section file included.~~
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 uncertatinties~~	~~RSICC Prog. No. PSR-529~~	~~20, 21~~	~~Two components. MAXED is a maximum entropy code. GRAVEL (22) is an iterative code.~~
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~~	~~23, 24~~	~~Several components, STAY'NL, X333, and MIEKE. Distributed by IAEA as part of the REAL-84 interlaboratory exercise on spectrum adjustment (25).~~

Program	Solution Method	Code Available From	References	Comments
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~~	~~26~~	~~Simultaneous evaluation with differential and integral data, primarily used for cross-section evaluation but extensible to spectrum adjustments.~~
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~~ The boldface numbers in parentheses refer to the list of references appended to this guide.

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 $\tilde{\Phi}$ 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 (~~27~~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 (~~27~~28, ~~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 ~~(27~~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.

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 E482. 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 (~~29~~30) and (~~16~~17). 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 (~~Refs ~~30~~31, ~~31~~32~~)).~~).

4.2.3 Benchmark neutron spectra can be included as simultaneous input in some codes if the dosimetry measurements are benchmark referenced (see Guides E2005 and E2006 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 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 neutron-transport based 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 E1018 whenever possible.

4.3.2 The group-averaged cross sections σ_{ij} depend, according to Eq 4~~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 E482). 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, ~~collapsing of the energy structure of the cross sections requires the corresponding cross section covariance matrices to be collapsed to the same structure. Experience has shown 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).

4.3.2.1 An alternative to the procedure described in 4.3.2 is to expand the input fluence spectrum to the same fine-group structure as the cross sections. In that case, an appropriate procedure is to calculate, by summation, the cumulative fluence from the lowest energy boundary up to each successive energy boundary, up to and including the upper energy of the highest energy group. The cumulative fluence thus obtained may then be interpolated by an appropriate method to find the cumulative fluence at each of the energy group boundaries used for the group-averaged cross sections. If the covariance matrix of the fluence spectrum in the fine group structure is needed, it should be generated by assuming that the fractional or percentage uncertainties remain