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StandardGuide for
Application of ASTM Evaluated Cross Section Data File, Matrix E706 (IIB)[1]

This standard is issued under the fixed designation E1018; 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 establishment and use of an ASTM evaluated nuclear data cross section and uncertainty file for analysis of single or multiple sensor measurements in neutron fields related to light water reactor LWR-Pressure Vessel Surveillance (PVS). These fields include in- and ex-vessel surveillance positions in operating power reactors, benchmark fields, and reactor test regions.

1.2 Requirements for establishment of ASTM-approved cross section files address data format, evaluation requirements, validation in benchmark fields, evaluation of error estimates (covariance file), and documentation. A further requirement for components of the ASTM-approved cross section file is their internal consistency when combined with sensor measurements and used to determine a neutron spectrum.

1.3 Specifications for use include energy region of applicability, data processing requirements, and application of uncertainties.

1.4 This guide is directly related to and should be used primarily in conjunction with Guides E482 and E944, and Practices E560, E185, and E693.

1.5 The ASTM cross section and uncertainty file represents a generally available data set for use in sensor set analysis. However, the availability of this data set does not preclude the use of other validated data, either proprietary or nonpropri-etary. When alternate cross section files are used that deviate from the requirements laid out in this standard, the deviations should be noted to the customer ofr the dosimetry application.

1.6 *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:*[2]

E170 Terminology Relating to Radiation Measurements and Dosimetry

E185 Practice for Design of Surveillance Programs for Light-Water Moderated Nuclear Power Reactor Vessels

E482 Guide for Application of Neutron Transport Methods for Reactor Vessel Surveillance, E706 (IID)

E560 Practice for Extrapolating Reactor Vessel Surveillance Dosimetry Results, E 706(IC) (Withdrawn 2009)[3]

E693 Practice for Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacements Per Atom (DPA), E 706(ID)

E706 Master Matrix for Light-Water Reactor Pressure Vessel Surveillance Standards, E 706(0) (Withdrawn 2011)[3]

E844 Guide for Sensor Set Design and Irradiation for Reactor Surveillance, E 706 (IIC)

E853 Practice for Analysis and Interpretation of Light-Water Reactor Surveillance Results, E706(IA)

E854 Test Method for Application and Analysis of Solid State Track Recorder (SSTR) Monitors for Reactor Surveillance, E706(IIIB)

E910 Test Method for Application and Analysis of Helium Accumulation Fluence Monitors for Reactor Vessel Surveillance, E706 (IIIC)

E944 Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance, E 706 (IIA)

E1005 Test Method for Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance, E 706 (IIIA)

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

3. Terminology

3.1 *Definitions of Terms Specific to This Standard:*

3.1.1 *benchmark field*—a limited number of neutron fields have been identified as benchmark fields for the purpose of dosimetry sensor calibration and dosimetry cross section data development and testing (**1, 2**).[4] See Terminology E170. These fields are permanent facilities in which experiments can be repeated. In addition, differential neutron spectrum measurements have been performed in many of the fields to provide, together with transport calculations and integral measurements, the best state-of-the-art neutron spectrum evaluation. To supplement the data available from benchmark fields, most of which are limited in fluence rate intensity, reactor test regions for dosimetry method validation have also been defined, including both in-reactor and ex-vessel dosimetry positions. Table 1 lists some of the neutron fields that have been used for data development, testing, and evaluation. Other benchmark fields used for testing LWR calculations are described in E2005 Guide for the Benchmark Testing of Reactor Dosimetry in Standard and Reference Neutron Fields, E706 (IIE-1).

3.1.1.1 *standard field*—these fields are produced by facilities and apparatus that are stable, permanent, and whose fields are reproducible with neutron fluence rate intensity, energy spectra, and angular fluence rate distributions characterized to state-of-the-art accuracy. Important standard field quantities must be verified by interlaboratory measurements. These fields exist at the National Institute of Standards and Technology (NIST) and other laboratories.

3.1.1.2 *reference field*—these fields are produced by facilities and apparatus that are permanent and whose fields are reproducible, less well characterized than a standard field, but acceptable as a measurement reference by the community of users.

3.1.1.3 *controlled environment*—these environments are well-defined neutron fields with some spectral definitions, employed for a restricted set of validation experiments over a range of energies.

3.1.2 *dosimetry cross sections*—cross sections used for dosimetry application and which provide the total cross section for production of particular (measurable) reaction products. These include fission cross sections for production of fission products, activation cross sections for the production of radioactive nuclei, and cross sections for production of measurable stable products, such as helium.

3.1.3 *evaluated data*—values of physical quantities representing a current best estimate. Such estimates are developed by experts considering measurements or calculations of the quantity of interest, or both. Cross section evaluations, for example, are conducted by teams of scientists such as the ENDF/B Cross Section Evaluation Working Group (CSEWG) (see also section 3.1.5.2).

3.1.4 *Evaluated Nuclear Data File (ENDF)*—consists of neutron cross sections and other nuclear data evaluated from available experimental measurements and calculations. Two types of ENDF files exist.

3.1.4.1 *ENDF/B files*—evaluated files officially approved by CSEWG [see ENDF documents 102 (**3**), 201 (**4**), and 216 (**5**)] after suitable review and testing.

3.1.4.2 *ENDF/A files*—evaluated files including outdated versions of ENDF/B, the International Reactor Dosimetry File (IRDF-2002) (**6**), the Japanese Evaluated Nuclear Data Library (JENDL) (**7**), BROND (USSR) (**8**) and other evaluated cross section libraries. These files include partial as well as complete evaluations.

3.1.5 *integral data/differential data*—integral data are data points that represent an integrated sensor's response over a range of energy. Examples are measurements of reaction rates

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[4] The boldfaced numbers in parentheses refer to the list of references at the end of this guide.

TABLE 1 Partial List of Neutron Fields for Validating Dosimetry Cross Sections

Neutron Field	Sample Facility Location	Energy		Useful Energy Range for Data Testing[A]	Reference Documentation
		Median	Average		
Standard Fields					
Thermal Maxwellian	NIST	...	...	<0.51 eV	
^{252}Cf Fission	NIST (**24**)	1.68 MeV	2.13 MeV	100 keV–8 MeV	Ref **24** Designation XCF-5-N1
^{235}U Thermal Fission	NIST (**24**) Mol-χ_{25} (**25, 26**)	1.57 MeV	1.97 MeV	250 keV–3 MeV	Ref **24** Designation XU5-5-N1
ISNF	NIST (**27**) NISUS (**28**) Mol-$\Sigma\Sigma$ (**29**)	0.56 MeV	~1.0 MeV	10 keV–3.5 MeV	Ref **24** Designation ISNF(5)-1-L1
Reference Fields					
BIG TEN	LANL (**30, 31**)	0.33 MeV	0.58 MeV	10 keV–3 MeV	Ref **30** Fast Reactor Benchmark 20
CFRMF	EGG-Idaho (**30, 32**)	0.375 MeV	0.76 MeV	4 keV–2.5 MeV	Ref **30** Dosimetry Benchmark 1
Controlled Environments					
PCA-PV	ORNL (**33**)	...	...	100 keV–10 MeV	Ref **33**
EBR-II	ANL-West (**34**)	...	...	1 keV–10 MeV	Ref **34**
FFTF	HEDL (**35**)	...	...	1 keV–10 MeV	Ref **35**

[A] The requirements for the data testing energy range are much more strict for reference and standard fields than for controlled fields. These testing energy ranges reflect comparison with calculations based on published spectra for reference and standard fields, but only address data reproducibility for controlled environments.

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or fission rates in a fission neutron spectrum. Differential data are measurements at single energy points or over a relatively small energy range. Examples are time-of-flight measurements, proton recoil spectrometry, etc. (**38**).

3.1.6 *uncertainty file*—the uncertainty in cross section data has been included with evaluated cross section libraries that are used for dosimetry applications. Because of the correlations between the data points or cross section parameters, these uncertainties, in general, cannot be expressed as variances, but rather a covariance matrix must be specified. Through the use of the covariance matrix, uncertainties in derived quantities, such as average cross sections, can be calculated more accurately.

4. Significance and Use

4.1 The ENDF/B library in the United States and similar libraries elsewhere, such as JEF (**9**), JENDL (**7**), and BROND (**8**), provide a compilation of neutron cross section and other nuclear data for use by the nuclear community. The availability of these excellent and consistent evaluations makes possible standardized usage, thereby allowing easy referencing and intercomparisons of calculations. However, as the first ENDF/B files were developed it became apparent that they were not adequate for all applications. This need resulted in the development of the ENDF/B Dosimetry File (**5**, **10**), consisting of activation cross sections important for dosimetry applications. This file was made available worldwide. Later, other "Special Purpose" files were introduced (**23**). In the ENDF/B-VI compilation (**41**), dosimetry files were identified, but they no longer appeared as separate evaluation files. The ENDF/V-VII compilation (**37**) removed most of the covariance files used by the dosimetry community. It kept the covariance files for the "standard cross sections" in a special sub-library, but the covariance data in this sub-library is only provided over the energy range in which each reaction is considered to be a "standard", and does not include the full energy range required for LWR PVS dosimetry applications.

4.2 Another file of evaluated neutron cross section data has been established by the International Atomic Energy Agency (IAEA) for reactor dosimetry applications. This file, the International Reactor Dosimetry File (IRDF-2002) (**6**), draws upon the ENDF/B files and supplements these evaluations with a set of reactions evaluated by groups often outside of the United States. Some of the IRDF-2002 supplemental reactions represent material evaluations that are currently being examined by the CSEWG. The supplemental IRDF-2002 evaluations only include the specific reactions of interest to the dosimetry community and not a full material evaluation. The ENDF community requires a complete evaluation before including it in the main ENDF/B evaluated library.

4.3 The application to LWR surveillance dosimetry may introduce new data needs that can best be satisfied by the creation of a dedicated cross section file. This file shall be in a form designed for easy application by users (minimal processing). The file shall consist of the following types of information or indicate the sources of the following type of data that should be used to supplement the file contents:

4.3.1 Dosimetry cross sections for fission, activation, helium production sensor reactions in LWR environments in support of radiometric, solid state track recorder, helium accumulation dosimetry methods (see Test Methods E853, E854, E910, and E1005).

4.3.2 Other cross sections or sensor response functions useful for active or passive dosimetry measurements, for example, the use of neutron absorption cross sections to represent attenuation corrections due to covers or self-shielding.

4.3.3 Cross sections for damage evaluation, such as displacements per atom (dpa) in iron.

4.3.4 Related nuclear data needed for dosimetry, such as branching ratios, fission yields, and atomic abundances.

4.4 The ASTM-recommended cross sections and uncertainties are based mostly on the ENDF/B-VI and IRDF-2002 dosimetry files. Damage cross sections for materials such as iron have been added in order to promote standardization of reported dpa measurements within the dosimetry community. Integral measurements from benchmark fields and reactor test regions shall be used to ensure self-consistency and establish correlations between cross sections. The total file is intended to be as self-consistent as possible with respect to both differential and integral measurements as applied in LWR environments. This self-consistency of the data file is mandatory for LWR-pressure vessel surveillance applications, where only very limited dosimetry data are available. Where modifications to an existing evaluated cross section have been made to obtain this self-consistence in LWR environments, the modifications shall be detailed in the associated documentation (see 5.6).

5. Establishment of Cross Section File

5.1 *Committee*—The cross section and uncertainty file shall be established and maintained under a responsible task group appointed by Subcommittee E10.05 on Nuclear Radiation Metrology. The task group shall review, and approve all data before insertion of the file and ensure the adequate testing has been performed on the file contents. The task group shall establish requirements, data formats, etc.

5.2 *Formats*—Formats shall generally conform to one of two types. The first format type is that referred to as the ENDF-6 format and is specified in ENDF-201 (**4**). The second format type consists of multigroup data in the 640 group SAND-II (**11,12**) energy structure (see Practice E693 for SAND-II energy group structure). The multigroup data format is the preferred form since it is more compatible with the forms typically used to represent facility neutron spectra. The spectrum weighting function used to collapse the point cross section data onto the multigroup energy grid should be generic in nature and shall be completely specified in the associated documentation.

5.3 *Cross Section Evaluation*—Most evaluations generally shall be based on the IRDF-2002 Dosimetry File. Cross sections shall be consistent within error bounds for selected benchmark fields (see 5.4 and Table 1). Dosimetry cross sections presently not in ENDF/B or IRDF-2002 shall be obtained from other sources or new evaluations. Other cross

sections may be obtained from other sources, for example, the dpa cross section for iron may be obtained from Practice E693.

5.4 *Cross Section Validation*—The cross section file will be validated for LWR applications using dosimetry measurements made in benchmark fields. Such validation may result in necessary modifications to cross sections to eliminate significant biases. Modification of ENDF/B and IRDF-2002 files shall be done in a manner consistent with the uncertainties specified for the differential data, using a least squares methodology.

5.5 *Related Nuclear Data for Dosimetry Application*—All necessary related data shall be specified in the documentation associated with the specific dosimetry application. These data include isotopic abundances, gamma branching ratios, fission yields, half-lives, etc., as appropriate. Updates of these data shall require, in general, a revalidation of the cross section (see 5.4). In the ENDF-6 format this data can be specified as comment cards in the File 1 General Information section. The evaluation file or associated documentation may cite a comprehensive dosimetry-quality source, such as the *Nuclear Data Guide for Reactor Neutron Metrology* (**13**), for the related nuclear data.

5.5.1 If the related data is not explicitly provided in the cross section evaluation itself or a reference is not cited, then the related data shall be taken from sources specified in 5.5.2-5.5.7. These sources represent the latest dosimetry-quality community-evaluated databases.

5.5.2 *isotopic abundances*—The most recent comprehensive listing of isotopic abundances is given in Ref(**14, 40**) and the 2005 *Nuclear Wallet Cards* (**15**) distributed by the National Nuclear Data Center (NNDC).

5.5.3 *gamma branching ratios*—The community standard source of branching ratios is the ENSDF (**16**).

5.5.4 *fission yields*—Within the U.S. community, the best data on fission yields is reflected in the ENDF/B-VII library (**37**)). The release date for the latest fission yield data is December 2006.

5.5.5 *half-life*—The most recent comprehensive listing of half-lives is given in Ref (**17**) and the 2005 *Nuclear Wallet Cards* (**15**) distributed by the NNDC.

5.5.6 *atomic weights*—The cross section evaluation shall specify the atomic weight of the target atom. If the atomic weight is not specified, the atomic weight of the product nucleus shall be determined from the mass excess data in the NNDC *Nuclear Wallet Cards* (**15**).

5.5.7 *Q-value*—The reaction Q-value is typically specified in the cross section evaluation. For some dosimetry sensor response functions, such as dpa, a Q-value may not be relevant. In this case a zero entry shall be recorded for the Q-value in the cross section evaluation. If a Q-value is not given in the cross section evaluation for a dosimetry reaction, then the cross section format must provide a numerical recipe for calculating the cross section down to a zero energy for the incident particle.

5.6 *Documentation*—ENDF/B and IRDF-2002 evaluations are documented by CSEWG and IAEA, respectively, and will be referenced. Cross sections re-evaluated for incorporation in the ASTM file must be completely documented. Documentation must reference all data used, including versions of all standard cross sections (ENDF/B-VI or other) to which data is normalized, and complete details of all benchmark spectra used. This documentation is typically provided or referenced in the File I portion of the cross section evaluated ENDF-6 format file.

5.7 *Updates*—Updates shall be issued periodically. Updates may consist of file modifications or complete replacement releases.

6. Establishment of Cross Section Uncertainty File

6.1 *Requirements*—All cross section data in the ASTM file, except damage functions which are given for the purpose of standardization and cover cross sections, must have uncertainties specified. Since these data tend to be highly correlated, to be meaningful, the uncertainty shall include correlations. Therefore, the uncertainties must be specified in the form of a covariance matrix. If the data is truly uncorrelated, this will result in a diagonal covariance matrix. This matrix should include correlations between cross section data for the same dosimetry reaction (autocorrelations) when it is available. Correlations with other cross sections also may be specified, and should at least be addressed in the primary file documentation.

6.2 *Format*—The uncertainty matrix must be associated directly with the cross section file. Two format types are acceptable. The first format is the ENDF-6 File 33/32 format. The File 32 portion of the second format captures the "long-range" and "short-range" correlations in the resonance parameters. This format style allows several functional representations as specified in ENDF-201 (**4**). The second format consists of a 640-energy-group representation of the cross section and a separate multi-group tabular representation of normalized triangular-covariance matrix (upper or lower triangular form). If the covariance matrix is expressed as a relative correlation matrix with quantities in a percentage, ranging from − 100 % to 100 %, then a tabular representation of the standard deviation shall be provided in the same energy group representation as is used for the normalized covariance matrix. In both the ENDF-6 and multigroup formats, the energy grid for the uncertainty matrix will be explicitly stated in the file and will be chosen to be consistent with maintaining the detail of the covariance information for the data while minimizing energy groups.

6.3 *Evaluation*—The uncertainty file shall be evaluated, validated, and documented in a manner similar to the cross section data. In this case, however, the benchmark testing is expected to provide a major contribution towards establishing realistic uncertainty estimates and correlations between cross sections.

7. Application of ASTM Evaluated Nuclear Data File

7.1 *Area of Applicability*—The ASTM file is established specifically for application to LWR Pressure Vessel Surveillance dosimetry and damage analysis. See Guide E844. It shall be validated and may be adjusted for this purpose and, therefore, should not be used for other applications without