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Standard Guide for Application of ASTM Evaluated Cross Section Data File[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.

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

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 nonproprietary. 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.*

1.7 *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
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
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
E844 Guide for Sensor Set Design and Irradiation for Reactor Surveiillance
E853 Practice for Analysis and Interpretation of Light-Water Reactor Surveillance 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
E944 Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance
E1005 Test Method for Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance
E2005 Guide for Benchmark Testing of Reactor Dosimetry in Standard and Reference Neutron Fields

[1] This guide is under the jurisdiction of ASTM Committee E10 on Nuclear Technology and Applicationsand is the direct responsibility of Subcommittee E10.05 on Nuclear Radiation Metrology.

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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.

[3] The last approved version of this historical standard is referenced on www.astm.org.

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.

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 (**15**), 201 (**16**), and 216 (**17**)] 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) (**18**), the Japanese Evaluated Nuclear Data Library (JENDL) (**19**), BROND (USSR) (**20**) 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 or fission rates in a fission neutron spectrum. Differential data

[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 (**3**)	1.68 MeV	2.13 MeV	100 keV–8 MeV	Ref **3** Designation XCF-5-N1
^{235}U Thermal Fission	NIST (**3**) Mol-χ_{25} (**4**, **5**)	1.57 MeV	1.97 MeV	250 keV–3 MeV	Ref **3** Designation XU5-5-N1
ISNF	NIST (**6**) NISUS (**7**) Mol-$\Sigma\Sigma$ (**8**)	0.56 MeV	~1.0 MeV	10 keV–3.5 MeV	Ref **3** Designation ISNF(5)-1-L1
		Reference Fields			
BIG TEN	LANL (**9**, **10**)	0.33 MeV	0.58 MeV	10 keV–3 MeV	Ref **9** Fast Reactor Benchmark 20
CFRMF	EGG-Idaho (**9**, **11**)	0.375 MeV	0.76 MeV	4 keV–2.5 MeV	Ref **9** Dosimetry Benchmark 1
		Controlled Environments			
PCA-PV	ORNL (**12**)	...	...	100 keV–10 MeV	Ref **12**
EBR-II	ANL-West (**13**)	...	...	1 keV–10 MeV	Ref **13**
FFTF	HEDL (**14**)	...	...	1 keV–10 MeV	Ref **14**

^{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.

are measurements at single energy points or over a relatively small energy range. Examples are time-of-flight measurements, proton recoil spectrometry, etc. (**21**).

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 (**22**), JENDL (**19**), and BROND (**20**), 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 (**17, 23**), consisting of activation cross sections important for dosimetry applications. This file was made available worldwide. Later, other" Special Purpose" files were introduced (**24**). In the ENDF/B-VI compilation (**25**), dosimetry files were identified, but they no longer appeared as separate evaluation files. The ENDF/V-VII compilation (**26**) 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) (**18**), 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 (**16**). The second format type consists of multigroup data in the 640 group SAND-II (**27,28**) 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* (**29**), 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(**30, 31**) and the 2005 *Nuclear Wallet Cards* (**32**) distributed by the National Nuclear Data Center (NNDC).

5.5.3 *gamma branching ratios*—The community standard source of branching ratios is the ENSDF (**33**).

5.5.4 *fission yields*—Within the U.S. community, the best data on fission yields is reflected in the ENDF/B-VII library (**26**)). 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 (**34**) and the 2005 *Nuclear Wallet Cards* (**32**) 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* (**32**).

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 (**16**). 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 suitable caution. Use shall be in accordance with other standards referenced in Section 2. Table 2 shows the current contents of the ASTM evaluated nuclear data file and specifies

TABLE 2 Recommended Sources for Several Useful Dosimetry Cross Sections

NOTE 1—P = Primary source of recommended evaluation.
• = Identical to primary source.

Dosimetry Reaction	Material ID in Primary Library	Cross Section Library					Comment
		ENDF/B-VI.8 (16)	JENDL/D-99 (35)	JEFF 3.1 (22)	RRDF-2002 (36)	IRDF-2002 (18)	
^{6}Li(n,X)^{4}He	325	P				•	A,B,C,D
^{10}B(n,X)^{4}He	525	P				•	B,C,D,E
^{23}Na(n,γ)^{24}Na	1125	P				•	F
^{24}Mg(n,p)^{24}Na	1225					•	G
^{27}Al(n,p)^{27}Mg	1325				P	•	H
^{27}Al(n,α)^{24}Na	1325				P	•	H
^{32}S(n,p)^{32}P	1625					•	I,J
^{45}Sc(n,γ)^{46}Sc	2126					•	F,K
^{46}Ti(n,p)^{46}Sc	2225				P	•	H,L
^{47}Ti(n,p)^{47}Sc	2228				P	•	H,L
^{48}Ti(n,p)^{48}Sc	2231				P	•	H
^{55}Mn(n,γ)^{54}Mn	2525	P				•	F
^{55}Mn(n,2n)^{54}Mn	2525	P					M
^{54}Fe(n,p)^{54}Mn	2625	P				•	J
^{56}Fe(n,p)^{56}Mn	2631				P	•	
^{58}Fe(n,γ)^{58}Fe	2637		P			•	F,N
^{nat}Fe(n,X)dpa	2600					•	O,P
^{59}Co(n,p^{59}Fe	2725	P					M
^{59}Co(n,γ)^{60}Co	2726/2725					•	J,Q
^{59}Co(n,α)^{56}Mn	2712				P		J
^{59}Co(n,2n)^{58}Co	2726/2725					•	J
^{58}Ni(n,p)^{58}Co	6433/2825				P	•	
^{58}Ni(n,2n)^{57}Ni	2825			P		•	
^{60}Ni(n,p)^{60}Co	2831	P				•	
^{63}Cu(n,γ)^{64}Cu	2925	P				•	
^{63}Cu(n,2n)^{62}Cu	2925	P				•	
^{63}Cu(n,α)^{60}Co	6435/2925				P	•	
^{65}Cu(n,2n)^{64}Cu	2931	P				•	
^{64}Zn(n,p)^{64}Cu	3025				P		
^{90}Zr(n,2n)^{89}Zr	4025		P				
^{93}Nb(n,γ)^{94}Nb	4125	P				•	
^{93}Nb(n,2n)^{92m}Nb	4112				P	•	
^{93}Nb(n,n')^{93m}Nb	4112				P	•	P
^{103}Rh(n,n')^{103m}Rh	4511				P	•	P
^{109}Ag(n,γ)^{110m}Ag	4731					•	K
^{115}In(n,γ)^{116m}In	4931	P				•	
^{115}In(n,n')^{115}In	4932/4931				P	•	P
^{197}Au(n,γ)^{198}Au	7925	P				•	C
^{197}Au(n,2n)^{196}Au	7925					•	
^{232}Th(n,f)F.P.	9040	P				•	P
^{235}U(n,f)F.P.	9228	P				•	C,D
^{238}U(n,f)F.P.	9237		P			•	C,D,P
^{237}Np(n,f)F.P.	9346				P	•	P
^{239}Pu(n,f)F.P.	9437		P			•	

[A] The ^{6}Li ^{4}He production is obtained from the ENDF/B-VI cross sections by summing the MT=105 and the MT=4 cross sections and subtracting the MT=57 cross section.
[B] This cross section is a combination of several reaction components. The recommended covariance matrix is taken from the covariance of the predominant reaction component, which is typically the (n,α) or (n,t) component.
[C] Use of the ENDF/B-VII.0 standards sub-library is under consideration. This transition is pending treatment of the cross section in energy regions outside the region for which covariance data is given in the standards sub-library.
[D] The covariance data is taken from IRDF-2002 instead of from ENDF/B-VI because the ENDF covariance data was deliberately eliminated from ENDF/B-VI.8 pending further analysis of correlations in the experimental data base that may not have been adequately taken into account. The covariance still reflects the evaluation data.
[E] The ^{10}B ^{4}He production is obtained from the ENDF/B-VI cross sections by summing the MT=107 and twice the MT=113 cross section.
[F] Experience suggests that this sensor may not be consistent with other dosimetry sensors for spectra where the majority of the sensor response comes from neutrons with energies above 10 keV. For fast neutron applications this sensor shoud be used with caution while the community examines the issue in more detail.
[G] From an IRK evaluation found in IRDF-90(37).
[H] From an update to the RRDF-98 library used in IRDF-2002.
[I] The latest GLUCS-3 cross section (38) is the same as that found in the IRDF-2002 except for a small difference in the reaction threshold energy and a different covariance representation.
[J] The literature has a conflict in the pedigree/source of the IRDF-2002 evaluation since it does not originate from ENDF/B-VI released libraries and special purpose dosimetry libraries were eliminated from the ENDF/B-VI release process. The IRDF-2002 documentation states that this cross section comes from the IRDF-90 library, but it does not use exactly the same representation as is found in the IRDF-90 library.
[K] From a CNDC evaluation.
[L] You must consider the (n, np) interference reactions on other titanium isotopes for neutron energies above 10 MeV. An alternative approach is to use a cross section that combines the appropriate titanium (n,p) and (n, np) reactions. This cross section has a target of the natural element and includes all reaction channels that result in the same primary residual nucleus. This type of combined reaction isoften denoted as ^{nat}Ti(n,x)^{46}Sc, ^{nat} Ti(n,X)^{47}Sc.
[M] This reaction is not included in the IRDF-2002 library.

[N] The natural abundance of ^{58}Fe has changed considerably over the last 25 years. This makes it difficult to ensure that the abundance value used in the evaluation is the same as is used in the interpretation of the ^{58}Fe activation product. The ^{58}Fe natural abundance value consistent with the time of this evaluation and release were done in 0.282(4) %

[O] The iron dpa is taken from Practice E683-01.

[P] The importance of interference by photon-induced reactions should be considered.

[Q] The file ID number, MAT, has been changed in the ASTM library to avoid a conflict between evaluations taken from different libraries. In the case of ^{59}Co, the number 2725 was changed to 2726. In the case of ^{58}Ni, the number 2825 in RRDF-2002 was changed to the original RRDF-98 MAT of 6433.

the origin of the data for each dosimetry reaction. Modifications to the contents of Table 2 shall be made in accordance with Section 5.

NOTE 1—Section 4.3.2 indicates that the contents of this dosimetry cross section file can contain neutron attenuation cross sections. There are, currently, no recommended cover cross sections in Table 2. The boron and lithium cross sections that appear in Table 2 support helium accumulation fluence monitors (HAFMs), see Test Methods E910. If applications require the use of a cover cross section, users are not limited to cross sections that appear in this file and, per 1.5, can add their own cross sections while documenting the deviations from the requirements that apply to the selection of materials that appear in this recommended dosimetry file.

NOTE 2—The methodology for the treatment of the additional uncertainty introduced when applying covers to produce modified sensor response functions is not fully developed by the community. This standard only addresses the selection of the nuclear data used to support the use of covers and the characterization of its uncertainty. As the dosimetry community refines the methodology for the use of covers, the requirements for the specification of the uncertainty in the underlying cover cross sections may change. Current issues for community consideration include the use of a total or absorption cross section with an exponential attenuation model versus the use of all the cross sections through adjoint radiation transport approaches (**39**) as well as the role/necessity for energy-dependent cross reaction covariance matrices in the uncertainty quantification.

NOTE 3—The primary evaluation sources indicated in Table 2 do not include details on the branching ratios because these are built into the evaluated reaction channel that is characterized. This is made clear in the comments associated with the evaluation. The primary evaluations also do not include fission yields and atomic abundances. Sections 5.5.2 and 5.5.4 provide the recommended data to be used in association with these cross sections.

7.2 *Processing Code Requirements*—Processing code requirements have been kept minimal through the format specifications. A code for reducing the cross section data in the ENDF-6 format is required. The NJOY-99 (**28**) and the Mieke (**40**) codes are examples of available processing codes that will handle the ENDF-6 format specifications. Data specified in the tabular multigroup format should be usable directly in spectrum adjustment codes.

7.3 *Uncertainty File Usage*—The cross section uncertainty file shall be used as one input to the determination of the overall uncertainties of processed quantities such as fluences or dpa. It is expected that, using least squares adjustment codes such as FERRET (**41**), LSL-M2 (**42**), STAY'SL (**43**), or LEPRICON (**44**), a good statistical evaluation of the uncertainty of processed quantities can be obtained. The use of validated cross section and uncertainty files will provide the needed confidence to justify usage of derived exposure parameter values and uncertainties for defining neutron-induced material property change limits for LWR nuclear power plants.

8. Availability

8.1 The ASTM file shall be available to all users. The primary distribution channel for the individual cross section data in the ENDF-6 format, and as recommended in Table 2, is through the four nuclear data centers. These ENDF-6 format cross section files are available as part of the source dosimetry libraries indicated in Table 2 and can be requested from the nuclear data centers. The four nuclear data centers are:

8.1.1 USA National Nuclear Data Center at Brookhaven National Laboratory, USA.

8.1.2 USSR Nuclear Data Center at the Fiziko-Energeticheskij Institute, Obninsk, USSR.

8.1.3 NEA Data Bank at Saclay, France.

8.1.4 IAEA Nuclear Data Section at Vienna, Austria.

8.2 *Multigroup Representation:*

8.2.1 The Radiation Safety Information Computation Center (RSICC) operated by the Oak Ridge National Laboratory shall serve as a distribution center for cross sections in the multigroup format. A multigroup representation of each of the dosimetry reactions in Table 2 along with covariance data can be found as part of the E1018doslib package at RSICC.

8.2.2 In addition, for those cross sections that have been taken from the IRDF-2002 file, the IRDF file uses an ENDF-6 format that consists of a 640 energy group histogram representation for the cross sections. Thus this representation satisfies the requirements of the ENDF-6 format and of the multigroup representation. The IRDF-2002 cross sections are distributed through the nuclear data centers as detailed in 8.1.

8.3 Electronic files are available from ASTM that contain a compendium of all of the recommended individual files from Table 2 in ENDF-6 and multigroup format. These files are available from the ASTM E10 website at: http://www.astm.org/COMMIT/COMMITTEE/E10_pubs.htm.

9. Keywords

9.1 covariance matrix; cross section; dosimetry; ENDF; IRDF; JEF; JENDL; nuclear metrology

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