INTERNATIONAL

**Designation: ~~E1018 – 09 (Reapproved 2013)$^{\varepsilon 1}$~~ E1018 – 20**

# 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.

~~$\varepsilon^1$ 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~~ASTM-recommended 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~~ASTM-recommended 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,~~ standard are used, the deviations should be noted to the customer ~~ofr~~of 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~~ safety, health, and ~~health~~environmental 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 (DPA)
E844 Guide for Sensor Set Design and Irradiation for Reactor ~~Surveillance~~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

[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.

Current edition approved ~~June 1, 2013~~March 1, 2020. Published ~~July 2013~~April 2020. Originally published as E1018 – 84. Last previous edition approved in ~~2009~~2013 as E1018~~-09.~~ – 09 (2013)$^{\varepsilon 1}$. DOI: ~~10.1520/E1018-09R13E01.~~10.1520/E1018-20.

[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.

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

## 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~~were permanent facilities in which experiments ~~can~~could 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. Many of these fields may not be available any longer, but new nuclear data files are still validated against the historical measurements made in these fields. Other benchmark fields used for testing LWR calculations are described in Guide 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 national 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 3.1.5~~section 3.1.5.2).~~).



**TABLE 1 Partial List of Characterized Neutron Fields Used 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**)<br>Designation XCF-5-N1 |
| $^{235}$U Thermal Fission | NIST (**3**)<br>Mol-$\chi_{25}$ (**4**,**5**) | 1.57 MeV | 1.97 MeV | 250 keV–3 MeV | Ref (**3**)<br>Designation XU5-5-N1 |
| ISNF | NIST (**6**)<br>NISUS (**7**)<br>Mol-ΣΣ (**8**) | 0.56 MeV | ~1.0 MeV | 10 keV–3.5 MeV | Ref (**3**)<br>Designation ISNF(5)-1-L1 |
| | | **Reference Fields** | | | |
| BIG TEN | LANL (**9**,**10**) | 0.33 MeV | 0.58 MeV | 10 keV–3 MeV | Ref (**9**)<br>Fast Reactor Benchmark 20 |
| ~~CFRMF~~ | ~~EGG Idaho (9, 11)~~ | ~~0.375 MeV~~ | ~~0.76 MeV~~ | ~~4 keV–2.5 MeV~~ | ~~Ref 9~~ |
| CFRMF | INL (**9**, **11**) | 0.375 MeV | 0.76 MeV | 4 keV–2.5 MeV | Ref (**9**)<br>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~~ |
| EBR-II | INL (**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~~ stricter 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.

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

3.1.4 *Evaluated Nuclear Data File (ENDF)*—consists of neutron cross sections and other nuclear data evaluated from available experimental measurements (15) 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~~ format document ENDF-102 available at the CSEWG website, (~~15~~16~~),~~) ~~201~~after ~~(~~suitable~~16), and 216~~ review and testing. The latest version of the ENDF/B nuclear data evaluations is ENDF/B-VIII.0 (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 International Reactor Dosimetry and Fusion File (IRDFF) (19, 20), the Japanese Evaluated Nuclear Data Library (JENDL) (~~19~~21), BROND (USSR) (~~20~~22~~)~~), JEFF 23 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 are measurements at single energy points or over a relatively small energy range. Examples are time-of-flight measurements, proton recoil spectrometry, etc. (~~21~~24).

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~~JEFF (~~22~~23), JENDL (~~19~~21), and BROND (~~20~~22), 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 specialized ENDF/B Dosimetry File (17, ~~23~~25), consisting of activation cross sections important for dosimetry applications. This file was made available worldwide. Later, ~~other" Special~~other "Special Purpose" files were introduced (~~24~~26). In the ENDF/B-VI compilation (~~25~~27), dosimetry files ~~were identified, but they~~ no longer appeared as separate evaluation files. The ~~ENDF/V-VII~~ENDF/B-VII.0 compilation (~~26~~28) removed most of the reaction-specific 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~~are 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. Later updates to the ENDF/B releases added covariance files for some reaction channels but these covariance files were often based solely on calculations and were not representative of the methodology used to derive the underlying ENDF/B cross section. In response to the need for a dosimetry-specific library, the International Atomic Energy Agency convened a Coordinated Research Project (CRP) that drew upon the set of international experts to provide a recommended set of dosimetry cross sections and to compile a set of validation evidence that supported the use of this recommended dataset. This file, the International Reactor Dosimetry and Fusion File (IRDFF) (19, 20), draws upon other national nuclear evaluations and supplements these evaluations with a set of reactions evaluated by expert international groups. The IRDFF library was developed to support the LWR dosimetry application as well as other dosimetry applications that go beyond the scope of this standard and, as part of its development process, it incorporates validation data acquired in reference and standard benchmark neutron fields. Some of the IRDFF supplemental reactions represent material evaluations that are currently being examined by the CSEWG for inclusion within updated ENDF/B evaluations. The supplemental IRDFF 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.

~~4.2 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.~~

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

4.2.1 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).

4.2.2 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.

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

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

4.3 The ASTM-recommended cross sections and uncertainties are based mostly on the ~~ENDF/B-VI and IRDF-2002~~IRDFF (version 1.05) 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~~ have been considered in order to ensure self-consistency **and**(**29**). ~~establish correlations between cross sections.~~ The total dosimetry 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~~(**19**, **29**)).

## 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~~30**)~~.)~~. The second format type consists of multigroup data in the 640 group SAND-II (**~~27~~31**, **~~28~~32**) 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.~~nature. Unless specified differently in associated file documentation, the weighting function used in the collapse of the point cross sections is assumed to be equivalent to the NJOY-2012 GROUPR module iwt=8 weight function option for "thermal—1/Ee—fast reactor—fission +_ fusion" weight function using the default energy breakpoints (**32**).

5.3 *Cross Section Evaluation*—~~Most evaluations generally shall be based on the IRDF-2002 Dosimetry File.~~ The source(s) for the recommended cross section for each dosimetry reaction is/are identified in Table 2. Currently, the recommended cross section for most dosimetry reactions are found in the IRDFF-1.05 library. For the dosimetry reactions of interest for LWR applications and addressed in Table 2, users should use the IRDFF version 1.05 library as the recommended source for cross sections up to 20 MeV. Cross sections shall be consistent within error bounds for selected benchmark fields (see 5.4 and Table 1) and this consistency has been demonstrated for the IRDFF version 1.05 library (**29**). Dosimetry cross sections presently not in ~~ENDF/B or IRDF-2002~~ IRDFF-1.05 shall be obtained from ~~other~~ the designated alternate sources or from 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.

NOTE 1—The IRDFF library includes several iron dpa cross sections that are endorsed for use by the international reactor dosimetry community for application to LWR pressure vessel surveillance. The iron dpa from Practice E693 is one of these IRDFF dpa damage functions and it [MAT=2600, MF=3, MT=900] is the ASTM recommended response for this damage metric.

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~~IRDFF and ~~IRDF-2002~~ENDF/B 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~~should 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~~33**) or the SNLRMI, (**26**, **34**, **35**), 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~~ principal reference for isotopic abundance in dosimetry applications should be from the international collaboration Decay Data Evaluation Project (DDEP) as documented in BIPM-5 (**36**).~~of isotopic abundances is given in~~ This is the principal source of isotopic abundance data used in the IRDFF cross sections. When the data is not available in the BIPM-5, Ref (**~~30~~37~~, 31~~**) ~~and the 2005~~gives isotopic abundances ~~Nuclear Wallet Cards~~suitable for use(~~32~~) ~~distributed by the National Nuclear Data Center (NNDC).~~with this standard.

### 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~~ ~~ENDF/B-VI.8 (16)~~ | ~~JENDL/D-99 (35)~~ | ~~JEFF 3.1 (22)~~ | ~~RRDF-2002 (36)~~ | ~~IRDF-2002 (18)~~ | Comment |
|---|---|---|---|---|---|---|---|
| ~~$^{6}$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~~ | | | ~~•~~ | |

### TABLE 2 Recommended Sources for Several Useful Dosimetry Cross Sections

| ID # | Reaction Label | Original Source(s) for Recommended Cross Section | Library Material ID[I] | Consistent with IRDFF Library(19, 20) | Target Atom Natural Abundance (%)[H] | Residual Nuclei Half-life | Comment |
|---|---|---|---|---|---|---|---|
| 1 | $^{6}$Li(n,t)$^{4}$He | ENDF/B-VII.1 | 325 | Yes | 7.589 (24) | He, stable; $^{3}$He, 12.312 (25) a | A,B,C |
| 7 | $^{10}$B(n,α)$^{7}$Li | CENDL-3 | 525 | No | 19.82 (2) | stable | B,C,E,K |
| 22 | $^{23}$Na(n,γ)$^{24}$Na | JENDL-4.0/IRDFF | 1125 | No/Yes | 100. | 14.958 (2) h | F |
| 23 | $^{24}$Mg(n,p)$^{24}$Na | RRDF | 1225 | Yes | 78.951 (12) | 14.958 (2) h | G |
| 24 | $^{27}$Al(n,α)$^{24}$Na | RRDF | 1325 | Yes | 100. | 14.958 (2) h | G |
| 25 | $^{27}$Al(n,p)$^{27}$Mg | RRDF | 1325 | Yes | 100. | 9.458 (12) m | G |
| 30 | $^{32}$S(n,p)$^{32}$P | RRDF | 1625 | Yes | 95.04074 (88) | 14.284 (36) d | G |
| 33 | $^{45}$Sc(n,γ)$^{46}$Sc | FENDL-D/IRDFF | 2126 | No | 100. | 83.787 (16) d | D,F |
| | | | 2125 | Yes | | | |
| 34 | $^{46}$Ti(n,p)$^{46}$Sc | RRDF | 2225 | Yes | 8.249 (21) | 83.787 (16) d | I,G,J |
| 36 | $^{47}$Ti(n,p)$^{47}$Sc | RRDF | 2228 | Yes | 7.437 (14) | 3.3485 (9) d | I,G,J |
| 39 | $^{48}$Ti(n,p)$^{48}$Sc | RRDF | 2231 | Yes | 73.720 (22) | 43.67 (9) h | G |
| 47 | $^{55}$Mn(n,γ)$^{54}$Mn | IRDFF | 2525 | Yes | 100. | 2.57878 (46) h | F |
| 48 | $^{55}$Mn(n,2n)$^{54}$Mn | RRDF | 2525 | Yes | 100. | 2.57878 (46) h | G |
| 51 | $^{54}$Fe(n,p)$^{54}$Mn | GLUCS-3/RRDF(44) | 2625 | No/Yes | 5.8459 (230) | 312.19 (3) d | D,G |

**TABLE 2** *Continued*

| ID # | Reaction Label | Original Source(s) for Recommended Cross Section | Library Material ID[I] | Consistent with IRDFF Library(19, 20) | Target Atom Natural Abundance (%)[H] | Residual Nuclei Half-life | Comment |
|---|---|---|---|---|---|---|---|
| 52 | $^{56}$Fe(n,p)$^{56}$Mn | RRDF | 2631 | Yes | 91.7540 (240) | 2.57878 (46) h | G |
| 54 | $^{58}$Fe(n,$\gamma$)$^{59}$Fe | JENDL/D99(45)/ JEFF-3.1 | 2637 | No/Yes | 0.282 (4) | 44.495 (9) d | D,F,L |
| 55 | $^{nat}$Fe(n,X)dpa | ENDF/B-VI.1 | NA | Yes | NA | stable | M,N |
| 56 | $^{59}$Co(n,p$^{59}$Fe | RRDF | 2725 | Yes | 100. | 44.494 (12) d | G |
| 57 | $^{59}$Co(n,$\gamma$)$^{60}$Co | JENDL 4.0/IRDFF | 2725 | No/Yes | 100. | 5.2711 (8) a | |
| 58 | $^{59}$Co(n,2n)$^{58}$Co | RRDF | 2725 | Yes | 100. | 70.85 (3) d | G |
| 59 | $^{59}$Co(n,$\alpha$)$^{56}$Mn | RRDF | 2712 | Yes[O] | 100. | 2.57878 (46) h | G |
| 60 | $^{58}$Ni(n,2n)$^{57}$Ni | ENDF/B-VI | 2825 | Yes | 68.0769 (59) | 35.9 (3) h | |
| 61 | $^{58}$Ni(n,p)$^{58}$Co | RRDF | 2825 | Yes | 68.0769 (59) | 70.85 (3) d | G |
| 63 | $^{60}$Ni(n,p)$^{60}$Co | RRDF | 2831 | Yes | 26.2231 (51) | 5.2711 (8) a | G |
| 64 | $^{63}$Cu(n,2n)$^{62}$Cu | RRDF | 2825 | Yes | 69.174 (20) | 9.67 (3) m | G |
| 65 | $^{63}$Cu(n,$\gamma$)$^{64}$Cu | ENDF/B-VI | 2925 | Yes[O] | 69.174 (20) | 12.701 (2) h | |
| 66 | $^{63}$Cu(n,$\alpha$)$^{60}$Co | RRDF | 2925 | Yes | 69.174 (20) | 5.2711 (8) a | G |
| 67 | $^{65}$Cu(n,2n)$^{64}$Cu | RRDF | 2931 | Yes | 30.826 (20) | 12.7004 (20) h | G |
| 68 | $^{64}$Zn(n,p)$^{64}$Cu | RRDF | 3025 | Yes | 49.1704 (83) | 12.7004 (20) h | G |
| 73 | $^{64}$Zr(n,2n)$^{89}$Zr | RRDF | 4025 | Yes | 51.452 (9) | 78.42 (13) h | G |
| 77 | $^{93}$Nb(n,$\gamma$)$^{94[g+(m\rightarrow g)]}$Nb | ENDF/B-VI | 4125 | Yes | 100. | 2.03E4 (16) | |
| 78 | $^{93}$Nb(n,2n)$^{92m}$Nb | RRDF | 4112 | Yes | 100. | 10.15 (2) d | G |
| 79 | $^{93}$Nb(n,n')$^{93m}$Nb | RRDF | 4112 | Yes | 100. | 16.12 (15) a | N,G |
| 87 | $^{103}$Rh(n,n')$^{103m}$Rh | RRDF | 4525 | Yes | 100. | 56.114 (20) m | N,G |
| 88 | $^{109}$Ag(n,$\gamma$)$^{110m}$Ag | CNDC | 4731 | Yes | 48.1608 (51) | 249.78 (2) d | G |
| 92 | $^{115}$In(n,$\gamma$)$^{116m}$In | ENDF/B-VI | 4931 | Yes[O] | 95.719 (17) | 54.29 (17) m | |
| 93 | $^{115}$In(n,n')$^{115m}$In | RRDF | 4931 | Yes | 95.719 (17) | 4.486 (4) h | N,G |
| 110 | $^{197}$Au(n,$\gamma$)$^{198}$Au | IPPE | 7925 | Yes | 100. | 2.6943 (3) d | C |
| 112 | $^{197}$Au(n,2n)$^{196}$Au | 7925 | 7925 | Yes | 100. | 6.1669 (6) d | G |
| 119 | $^{232}$Th(n,f)FP | JENDL 4.0/IRDFF | 9040 | No/Yes | 100. | stable | N |
| 121 | $^{235}$U(n,f)FP | RRDF | 9228 | Yes | 0.72041 (36) | stable | C,G |
| 122 | $^{238}$U(n,f)FP | IRDFF | 9237 | Yes | 99.27417 (36) | stable | C,N |
| 123 | $^{237}$Np(n,f)FP | RRDF | 9346 | Yes | NA | stable | N,G |
| 124 | $^{239}$Pu(n,f)FP | IRDFF | 9437 | Yes | NA | stable | N |

[A] The total ~~$^{6}$~~$^{nat}$Li $^{4}$He production is ~~obtained from~~ typically the desired metric. It is obtained by combining the $^{6}$~~ENDF/B-VI~~ Li(n,t) ENDF/B-Viii.0 cross sections ~~by summing~~(MT=105) with the $^{6}$~~MT=105~~ Li(n,d) (MT=32) and $^{6}$~~the MT=4 cross sections~~ Li(n,2np) (MT=41) cross sections. If the sample is not isotopically pure $^{6}$Li, then the $^{7}$Li reactions need to be considered, for example, $^{7}$Li(n,2n$\alpha$) (MT=24) and $^{7}$~~subtracting the MT=57 cross section.~~Li(n.3n$\alpha$) (MT=25) cross sections.

[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,$\alpha$) 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.~~ Multiple recommended libraries are given. Existing experimental data and analysis of the evaluation methodology are not sufficient to identify a single recommended nuclear data evaluation.

[E] The $^{10}$B $^{4}$He production is ~~obtained from~~ typically the desired metric. It is obtained by combining the $^{10}$~~ENDF/B-VI cross sections by summing the MT=107 and twice the MT=113 cross section.~~B(n,$\alpha$) (MT=107) with twice the 10B(n,$\alpha$) (MT=113) cross section. If the sample is not isotopically pure $^{10}$B, then the $^{11}$B reactions need to be considered, for example, $^{11}$B(n,$\alpha$) (MT=107) and $^{11}$B(n,n$\alpha$) (MT=22) cross sections

[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~~should be used with caution while the community examines the issue in more detail.

[G] ~~From an IRK evaluation found in IRDF-90(~~ Various versions/releases of the Russian Reactor Dosimetry File (RRDF) (~~37~~46)~~).~~ set of evaluations exist but some of the later releases are not readily available to the public. Since many of the evaluations in this library were funded by the IAEA with the intention of providing a dosimetry-oriented evaluation for the IRDFF, the most accessible source of these evaluations are from the ENDF-6 format IRDFF file itself.

[H] ~~From an update to the~~ Natural abundance data from Ref 37~~RRDF-98 library used in IRDF-2002.~~.

[I] The ~~latest GLUCS-3 cross section (~~material ID ~~38)~~ is the ~~same as that~~ENDF-6 format 30 ~~found in the IRDF-2002 except for a small difference in the reaction threshold energy and a different covariance representation.~~ MAT identifier and is used in both the source evaluation and in the IRDFF library to identify the target nuclei.

[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.~~

[J] ~~You must consider the (n, np)~~ 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~~ is often denoted as $^{nat}$~~Ti(n,x)~~Ti(n,X) $^{46}$Sc, $^{nat}$Ti(n,X)$^{47}$Sc.

[K] This ~~reaction is not included in the IRDF-2002 library.~~evaluation deviated from the IRDFF 1.05 selection and was selected in order to be consistent with the ENDF/B-VII.1 Standards version of this reaction cross section.

[L] 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) %

[M] The iron dpa is taken from Practice E693~~E683-01.~~-17.

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

[O] ~~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~~ These evaluations in the IRDFF library reflect insignificant format changes from the cited source $^{59}$~~Co, the number 2725 was changed to 2726. In the case of~~ evaluation.$^{58}$~~Ni, the number 2825 in RRDF-2002 was changed to the original RRDF-98 MAT of 6433.~~