Designation: E 1018 – 95

Standard Guide for Application of ASTM Evaluated Cross Section Data File, Matrix E 706 (IIB)[1]

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 r

elated to 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 E 482 and E 944, and Practices E 560, E 185, and E 693.

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.

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:*

E 185 Practice for Conducting Surveillance Tests for Light Water-Cooled Nuclear Power Reactor Vessels, E706 (IF)[2,3]

E 482 Guide for Application of Neutron Transport Methods for Reactor Vessel Surveillance, E706 (IID)[2,3]

E 560 Practice for Extrapolating Reactor Vessel Surveillance Dosimetry Results, E706 (IC)[2,3]

E 693 Practice for Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacements Per Atom (DPA), E706 (ID)[2,3]

E 706 Master Matrix for Light-Water Reactor Pressure Vessel Surveillance Standards, E706 (O)[2,3]

E 844 Guide for Sensor Set Design and Irradiation for Reactor Surveillance, E706 (IIC)[2,3]

E 853 Practice for Analysis and Interpretation of Light-Water Reactor Surveillance Results, E706 (IA)[2,3]

E 854 Test Method for Application and Analysis of Solid State Track Recorder (SSTR) Monitors for Reactor Surveillance, E706 (IIIB)[2,3]

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

E 944 Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance, (IIA)[2,3]

E 1005 Test Method for Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance, E706 (IIIA)[2,3]

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] These fields are permanent or semi-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

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[2] The reference in parentheses refers to Section 5 as well as Figs. 1 and 2 of Matrix E 706.

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

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

which are limited in flux intensity, reactor test regions for dosimetry method validation have also been defined, including both in-reactor and ex-vessel dosimetry positions. Table 1 lists the neutron fields that are being used for data development, testing, and evaluation.

3.1.1.1 *standard field*—these fields are produced by facilities and apparatus that are permanent and whose fields are reproducible with neutron flux intensity, energy spectra, and angular flux distributions characterized to state-of-the-art accuracy. 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. The current recommended set of ENDF/B files is ENDF/B-VI, revision 2, July 1993.

3.1.4.2 *ENDF/A files*—evaluated files including outdated versions of ENDF/B, the International Reactor Dosimetry File (IRDF-90) **(6)**, the Japanese Evaluated Nuclear Data Library (JENDL) **(7)**, and BROND (USSR) **(8)** evaluated cross section libraries.

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.

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

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

Neutron Field	Sample Facility Location	Energy: Median	Energy: Average	Useful Energy Range for Data Testing[A]	Reference Documentation
		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-ΣΣ (**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.

of activation cross sections important for dosimetry applications. This file was made available worldwide. Later, other" Special Purpose" files were introduced. In the latest ENDF/B-VI compilation, dosimetry files are identified, but do not typically appear as separate evaluation files.

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-90) **(6)**, draws upon the ENDF/B-VI files and supplements these evaluations with a set of reactions evaluated by groups often outside of the United States. Some of the IRDF-90 supplemental reactions represent material evaluations that are currently being examined by the CSEWG. The supplemental IRDF-90 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 ENDF/B evaluated library.

4.3 The application to LWR surveillance dosimetry introduces new data needs that can best be satisfied by the creation of a special 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:

4.3.1 Dosimetry cross sections for fission, activation, helium production, and damage sensor reactions in LWR environments in support of radiometric, solid state track recorder, helium accumulation, and damage monitor dosimetry methods (see Test Methods E 853, E 854, E 910, and E 1005 and Matrix E 706-IIID).

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 and IRDF dosimetry files. Damage cross sections for materials such as iron may be 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, test, and approve all data before insertion of the file. 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 E 693 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 spectrum. 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-90 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-90 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 E 693.

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-90 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. 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 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)** and the 1990 *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 fission yield tapes **(4)**, tape numbers 90 through 96 and 98 through 100. The release date for the latest fission yield data is October 1993.

5.5.5 *half-life*—The most recent comprehensive listing of half-lives is given in Ref **(17)** and the 1990 *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-90 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.

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, must have uncertainties specified. Since these data are highly correlated, to be meaningful, the uncertainty must include correlations. Therefore, the uncertainties must be specified in the form of a covariance matrix. This matrix should include correlations between all cross section data for the same dosimetry reaction (autocorrelations). Correlations with other cross sections also may be specified, and should at least be addressed in the 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 format. This format allows several functional representations as specified in ENDF-201 **(4)**. The second format consists of a tabular representation of a normalized triangular covariance matrix (upper or lower triangular form). If the covariance matrix is expressed 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 being established specifically for application to LWR Pressure Vessel Surveillance dosimetry and damage analysis. It shall be validated and 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 the origin of the data for each dosimetry reaction. Modifications to the contents of Table 2 shall be made in accordance with Section 5.

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-91 **(12)** and the Mieke **(18)** 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 **(19)**, LSL-M2 **(20)**, STAY'SL **(21)**, or LEPRICON **(22)**, 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 data in the ENDF-6 format is through the four nuclear data centers. These ENDF-6 format cross section files are generally available as part of the 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 Shielding Information Center (RSIC) 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 SNLRML package **(23)** at RSIC.

8.2.2 In addition, for those cross sections that have been taken from the IRDF-90 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-90 cross sections are distributed through the nuclear data centers as detailed in 8.1.

TABLE 2 Recommended Source for Several Useful Dosimetry Cross Sections

NOTE 1—P = Primary source of recommended evaluation.
• = Identical to primary source.
[oplus] = Very close to recommended evaluation. Some smoothing may have taken place in the recommended cross section.

Dosimetry Reaction	Material ID in Primary Library	Cross Section Library			Comment
		ENDF/B-VI **(4)**	IRDF-90 **(4)**	GLUCS **(36)**	
$^{10}B(n,X)^{4}He$	525				*A,B*
$^{6}Li(n,X)^{4}He$	325	P			*C,B*
$^{23}Na(n,\gamma)^{24}Na$	1125	P			*D,E,F*
$^{24}Mg(n,p)^{24}Na$	1225		P		
$^{27}Al(n,p)^{27}Mg$	6313		•	P	
$^{27}Al(n,\alpha)^{24}Na$	1325		P		
$^{32}S(n,p)^{32}P$	32		•	P	
$^{45}Sc(n,\gamma)^{46}Sc$	2125	P	•		*F*
$^{46}Ti(n,p)^{46}Sc$	3222		•	P	*G*
$^{47}Ti(n,p)^{47}Sc$	3222		P	[oplus]	*G*
$^{48}Ti(n,p)^{48}Sc$	2231		•	P	
$^{55}Mn(n,\gamma)^{56}Mn$	1325				*H,F*
$^{55}Mn(n,2n)^{54}Mn$	2525	P	•		
$^{54}Fe(n,p)^{54}Mn$	2625	P	•	[oplus]	
$^{56}Fe(n,p)^{56}Mn$	561	•	•	P	
$^{58}Fe(n,\gamma)^{59}Fe$	2637	P			*F,I*
$^{nat}Fe(n,X)dpa$	8000				*J*
$^{59}Co(n,p)^{59}Fe$	2725	P			
$^{59}Co(n,\gamma)^{60}Co$	2725	P	•		
$^{59}Co(n,\alpha)^{56}Mn$	2725	P	•		
$^{59}Co(n,2n)^{58}Co$	2725		P		
$^{58}Ni(n,p)^{58}Co$	6433	P	•	[oplus]	
$^{58}Ni(n,2n)^{57}Ni$	2825	P	•		
$^{60}Ni(n,p)^{60}Co$	2831	P	•		
$^{63}Cu(n,\gamma)^{64}Cu$	2925	P	•		
$^{63}Cu(n,2n)^{62}Cu$	2925		P		
$^{63}Cu(n,\alpha)^{60}Co$	6435	P	•	[oplus]	
$^{65}Cu(n,2n)^{64}Cu$	2931	P	•	[oplus]	
$^{64}Zn(n,p)^{64}Cu$	3025		P		
$^{90}Zr(n,2n)^{89}Zr$	4025		P		
$^{93}Nb(n,\gamma)^{94}Nb$	4125	P			
$^{93}Nb(n,2n)^{92m}Nb$	4125		P		
$^{93}Nb(n,n')^{93m}Nb$	4125		P		*I*
$^{98}Mo(n,\gamma)^{99}Mo$	3426				*D,E,K*
$^{103}Rh(n,n')^{103m}Rh$	4525		P		*E,I*
$^{109}Ag(n,\gamma)^{110m}Ag$	7479				*D,E,L*
$^{115}In(n,\gamma)^{116m}In$	4931	P			
$^{115}In(n,n')^{115}In$	4931		P		*I*
$^{197}Au(n,\gamma)^{198}Au$	7925	P	•		
$^{197}Au(n,2n)^{196}Au$	7925		P		
$^{232}Th(n,f)F.P.$	9040	P			*E,I*
$^{235}U(n,f)F.P.$	9228	P	•		*M*
$^{238}U(n,f)F.P.$	9237	P	•		*M,I*
$^{237}Np(n,f)F.P.$	9346	P	•		*I*
$^{239}Pu(n,f)F.P.$	9437	P	•		

A 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.
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 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.
D No covariance data available for this dosimetry sensor. It is included as a best recommendation despite the requirements in 6.1.
E No recent dosimetry-oriented cross sections are available for this reaction. This cross section represents the best available data.
F 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.
G You must consider the (n,np) interference reactions on other titanium isotopes for neutron energies above 10 MeV.
H This dosimetry cross section is taken from the ENDF/B-V dosimetry cross sections. The ENDF/B-VI file is specifically excluded due to inconsistencies of the prominent resonances with other reaction cross sections.
I The importance of interference by photon-induced reactions should be considered.
J The iron dpa is taken from Practice E 693-79.
K Data taken from the JENDL (**(7)**) cross section library.
L Data taken from DOSCROS84 library (**37**). The cross section corresponds to the ENDF/B-V Activation cross section with a branching ratio of 5.299×10^{-2}.
M The covariance data is taken from IRDF-90 instead of from ENDF/B-VI because the covariance data was deliberately eliminated from ENDF/B-VI pending further analysis.

9. Keywords

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

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