Designation: E261 – 16

Standard Practice for Determining Neutron Fluence, Fluence Rate, and Spectra by Radioactivation Techniques[1]

This standard is issued under the fixed designation E261; the number immediately following the designation indicates the year of original adoption or, in the case of revision, the year of last revision. A number in parentheses indicates the year of last reapproval. A superscript epsilon (ε) indicates an editorial change since the last revision or reapproval.

1. Scope

1.1 This practice describes procedures for the determination of neutron fluence rate, fluence, and energy spectra from the radioactivity that is induced in a detector specimen.

1.2 The practice is directed toward the determination of these quantities in connection with radiation effects on materials.

1.3 For application of these techniques to reactor vessel surveillance, see also Test Methods E1005.

1.4 *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.*

NOTE 1—Detailed methods for individual detectors are given in the following ASTM test methods: E262, E263, E264, E265, E266, E343, E393, E481, E523, E526, E704, E705, and E854.

2. Referenced Documents

2.1 *ASTM Standards:*[2]

E170 Terminology Relating to Radiation Measurements and Dosimetry

E181 Test Methods for Detector Calibration and Analysis of Radionuclides

E262 Test Method for Determining Thermal Neutron Reaction Rates and Thermal Neutron Fluence Rates by Radioactivation Techniques

E263 Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Iron

E264 Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Nickel

E265 Test Method for Measuring Reaction Rates and Fast-Neutron Fluences by Radioactivation of Sulfur-32

E266 Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Aluminum

E343 Test Method for Measuring Reaction Rates by Analysis of Molybdenum-99 Radioactivity From Fission Dosimeters (Withdrawn 2002)[3]

E393 Test Method for Measuring Reaction Rates by Analysis of Barium-140 From Fission Dosimeters

E481 Test Method for Measuring Neutron Fluence Rates by Radioactivation of Cobalt and Silver

E523 Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Copper

E526 Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Titanium

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

E704 Test Method for Measuring Reaction Rates by Radioactivation of Uranium-238

E705 Test Method for Measuring Reaction Rates by Radioactivation of Neptunium-237

E722 Practice for Characterizing Neutron Fluence Spectra in Terms of an Equivalent Monoenergetic Neutron Fluence for Radiation-Hardness Testing of Electronics

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

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

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

E1005 Test Method for Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance

E1018 Guide for Application of ASTM Evaluated Cross Section Data File, Matrix E706 (IIB)

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

[1] This practice 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.

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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.

2.2 *ISO Standard:*

JCGM 100:2008 Evaluation of measurement data—Guide to the expression of uncertainty in measurement

JCGM 104:2009 Evaluation of measurement data—An introduction to the "Guide to the expression of uncertainty in measurement" and related documents

JCGM 101:2008 Evaluation of measurement data—Supplement 1 to the "Guide to the expression of uncertainty in measurement" – Propogation of distributions using a Monte Carlo method

JCGM 102:2011 Evaluation of measurement data—Supplement 2 to the "Guide to the expression of uncertainty in measurement" – Extension to any number of output quantities

JCGM 106:2012 Evaluation of measurement data—The role of measurement uncertainty in conformity assessment

3. Terminology

3.1 Descriptions of terms relating to dosimetry are found in Terminology E170.

4. Summary of Practice

4.1 A sample containing a known amount of the nuclide to be activated is placed in the neutron field. The sample is removed after a measured period of time and the induced activity is determined.

5. Significance and Use

5.1 *Transmutation Processes*—The effect on materials of bombardment by neutrons depends on the energy of the neutrons; therefore, it is important that the energy distribution of the neutron fluence, as well as the total fluence, be determined.

6. Counting Apparatus

6.1 A number of instruments are used to determine the disintegration rate of the radioactive product of the neutron-induced reaction. These include the scintillation counters, ionization chambers, proportional counters, Geiger tubes, and solid state detectors. Recommendations of counters for particular applications are given in Test Methods E181.

7. Requirements for Activation-Detector Materials

7.1 Considerations concerning the suitability of a material for use as an activation detector are found in Guide E844.

7.2 The amounts of fissionable material needed for fission threshold detectors are rather small and the availability of the material is limited. Licenses from the U.S. Nuclear Regulatory Commission are required for possession.

7.3 A detailed description of procedures for the use of fission threshold detectors is given in Test Methods E343, E393, and E854, and Guide E844.

8. Irradiation Procedures

8.1 The irradiations are carried out in two ways depending upon whether the instantaneous fluence rate or the fluence is being determined. For fluence rate, irradiate the detector for a short period at sufficiently low power that handling difficulties and shielding requirements are minimized. Then extrapolate the resulting fluence rate value to the value anticipated for full reactor power. This technique is sometimes used for the fluence mapping of reactors (**1**, **2**).[4]

8.2 The determination of fluence is most often required in experiments on radiation effects on materials. Irradiate the detectors for the same duration as the experiment at a position in the reactor where, as closely as possible, they will experience the same fluence, or will bracket the fluence of the position of interest. When feasible, place the detectors in the experiment capsule. In this case, long-term irradiations are often required.

8.3 It is desirable, but not required, that the neutron detector be irradiated during the entire time period considered and that a measurable part of the activity generated during the initial period of irradiation be present in the detector at the end of the irradiation. Therefore, the effective half-life, $t'_{1/2} = 0.693/\lambda'$ (see Eq 6), of the reaction product should not be much less than the total elapsed time from the initial exposure to the final shutdown.

8.4 As mentioned in 9.10 through 9.11, the use of cadmium-shielded detectors is convenient in separating contributions to the measured activity from thermal (E170) and epithermal (E170) neutrons. Also, cadmium shielding is helpful in reducing activities due to impurities and the loss of the activated nuclide by thermal-neutron absorption. The recommended thicknesses of cadmium is 1 mm. When bare and cadmium-shielded samples are placed in the same vicinity, take care to avoid partial shielding of the bare detectors by the cadmium-shielded ones.

9. Calculation

9.1 *Fluence:*

9.1.1 $\varphi(E, t)$ is the differential neutron fluence rate; that is, the fluence rate per unit energy per unit time for neutrons with energies between E and $E + dE$. When focusing on the neutron spectrum, the notation $\varphi(E)$ is sometimes used. $\varphi(E)$ has an implicit dependence on time. In many cases, the neutron spectrum does not vary with time.

9.1.2 The neutron fluence rate φ is the integral over energy of the differential neutron fluence rate.

$$\varphi = \int \varphi(E)dE \tag{1}$$

φ has an implicit dependence on time.

9.1.3 $\varphi(E)$ may be determined by computer calculations using neutron transport codes or by adjustment techniques using radioactivation data from multiple-foil irradiations.

9.1.4 The neutron fluence, Φ, is related to the time varying differential neutron fluence rate by the following expression:

$$\Phi = \int_0^\infty \int_{t_1}^{t_2} \varphi(E,t)dt\ dE \tag{2}$$

where:

$t_2 - t_1$ = duration of the irradiation period

[4] The boldface numbers in parentheses refer to a list of references at the end of this standard.

9.2 *Spectrum-Averaged Cross Sections:*

9.2.1 Spectrum-averaged cross sections (E170) are used in reaction rate calculations. A spectrum-averaged cross section is defined as follows:

$$\bar{\sigma} = \frac{\int_0^\infty \sigma(E)\,\varphi(E)\,dE}{\int_0^\infty \varphi(E)\,dE} \tag{3}$$

where:

$\sigma(E)$ = microscopic cross section for the isotope and reaction of interest. $\bar{\sigma}$ has an implicit dependence on time and may change if the neutron spectrum changes.

9.2.2 In order to calculate the spectrum-averaged cross section, the differential cross section of the nuclide and the neutron spectrum over the neutron energy range for which the nuclide has a non-negligible cross section must be known. When cross-section and spectrum information are not available, alternative procedures may be used; suggested alternatives are discussed in 9.10 – 9.12, and in the methods for individual detectors.

9.3 *Reaction Rate:*

9.3.1 The reaction rate per nucleus, R_R, for a given reaction is related to the fluence rate by:

$$R_R = \int_0^\infty \sigma_R(E)\varphi(E)dE \tag{4}$$

where:

$\sigma_R(E)$ = microscopic cross section for the isotope and reaction of interest.

9.3.2 It follows that:

$$R_R = \bar{\sigma}_R\varphi \;\; or \;\; \varphi = \frac{R_R}{\bar{\sigma}_R} \tag{5}$$

9.4 *Effective Decay Constant:*

9.4.1 The effective decay constant, λ', which may be a function of time, is related to the decay constant λ as follows:

$$\lambda' = \lambda + \int_0^\infty \sigma_a(E)\varphi(E)\,dE \tag{6}$$

where:

$\sigma_a(E)$ = the neutron absorption cross section for the product nuclide.

9.4.2 The effective decay constant accounts for burnup of a product nuclide during irradiation. Application of the effective decay constant for irradiation under varying fluence rates is discussed in this section and in the detailed methods for individual detectors.

9.5 *Activity:*

9.5.1 The activity of the sample, A, is the decay rate of the product nuclei of interest, N_p.

$$A = N_p\lambda \tag{7}$$

The activity at the end of the exposure period is calculated from an activation foil count rate as follows:

$$A = \lambda D/[(1 - \exp(-\lambda\, t_c))\exp(-\lambda\, t_w)] \tag{8}$$

where:

λ = decay constant for the radioactive nuclide,

t_c = time interval for counting,

t_w = time elapsed between the end of the irradiation period and the start of the counting period, and

D = number of disintegrations (net number of counts corrected for background, random and true coincidence losses, efficiency of the counting system, and fraction of the sample counted) in the interval t_c.

9.5.2 If, as is often the case, the counting period is short compared to the half-life ($=0.693/\lambda$) of the radioactive nuclide, the activity is well approximated as follows:

$$A = D/[t_c\exp(-\lambda\, t_w)] \tag{9}$$

9.5.3 The number of radioactive product nuclei, N_p, is related to the reaction rate by the following equation:

$$dN_p/dt = NR_R - N_p\lambda' \tag{10}$$

9.5.4 Solution of Eq 10, for the case where the neutron spectrum and N are constant and N_p=0 at t=0, yields the following expression for the activity of a foil:

$$A = N_p\lambda = (\lambda/\lambda')NR_R(1 - \exp(-\lambda' t_i)) \tag{11}$$

9.5.5 For irradiations at constant fluence rate, the saturation activity (E170), A_s, is calculated as follows:

$$A_s = A/(1 - \exp(-\lambda' t_i)) \tag{12}$$

where:

t_i = exposure duration.

It follows from Eq 11 and Eq 12 that:

$$A_S = (\lambda/\lambda')\,NR_R \tag{13}$$

The saturation activity corresponds to the number of disintegrations per foil per unit time for the steady-state condition in which the rate of production of the radioactive nuclide is equal to the rate of loss by radioactive decay and transmutation. The activity A approaches the saturation activity, A_s, but does not surpass it, as the exposure duration increases ($\exp(-\lambda' t)\rightarrow 0$).

9.5.6 The isotopic content of the target nuclide may be reduced during the irradiation by more than one transmutation process and it may be increased by transmutation of other nuclides so that the rate of change of the number of target nuclei with time is described by a number of terms:

$$dN/dt = -N\left(R_R + \sum_{i=1}^{n} R_i\right) + \sum_{j=1}^{m} N_j R_j \tag{14}$$

where:

i = discrete transmutation path for removal of the target isotope, and

j = discrete transmutation reaction whereby the target isotope is produced from isotope N_j and each of the R_i and R_j terms could be calculated from equations similar to Eq 4, using the appropriate cross sections.

9.5.6.1 The R_R term may predominate and, if R_R is constant, Eq 14 can be solved as

$$N = N_0\exp(-R_R\, t) \tag{15}$$

using the approximation that the change in target composition is negligible and replacing N by N_0.

9.5.6.2 During irradiation, the effective decay rate may be increased by transmutations of the product isotope (see Eq 6).

9.6 *Long Term Irradiations:*

9.6.1 Long irradiations for materials testing programs and reactor pressure vessel surveillance are common. Long irradiations usually involve operation at various power levels, including extended zero-power periods; thus, appropriate corrections must be made for depletion of the target nuclide, decay and burnout of the radioactive nuclide, and variations in neutron fluence rate. Multiple irradiations and nuclide burnup must also be considered in short-irradiation calculations where reaction-product half-lives are relatively short and nuclide cross sections are high.

9.6.2 Long irradiations usually involve operation at various power levels, and changes in isotopic content of the system; under such conditions φ(E, *t*) can show large variations with time.

9.6.3 It is usual to assume, however, that the neutron fluence rate is directly proportional to reactor power; under these conditions, the fluence can be well approximated by:

$$\Phi = \left(\frac{\varphi}{P}\right) \sum_{i=1}^{n} P_i t_i \tag{16}$$

where:

φ/P = average value of the neutron fluence rate, φ, at a reference power level, P,
t_i = duration of the i^{th} operating period during which the reactor operated at approximately constant power, and
P_i = reactor power level during that operating period.

9.6.3.1 Alternate methods include measuring the power generation rate in a fraction of the reactor volume adjacent to the volume of interest.

9.6.4 In a manner similar to power, the activity may be summed over the reactor operating periods. The activity for each operating period decays in subsequent periods, however, making the summation more complex.

9.6.4.1 The total irradiation period can be divided into a continuous series of periods during each of which φ(*E*) is essentially constant. Then the activity generated during the i^{th} irradiation period is:

$$A_i = [\lambda N_i (R_R/\lambda')_i](1 - \exp(-\lambda'_i t_i)) \tag{17}$$

where:

N_i = number of target atoms during the i^{th} period, and
t_i = duration of the i^{th} period.

9.6.4.2 The activity remaining from the i^{th} period at the end of the n^{th} period can be calculated as the following equation:

$$(A_n)_i = A_i \exp\left(-\sum_{j=i+1}^{n} \lambda'_j t_j\right) \tag{18}$$

9.6.4.3 The total activity of the foil at the end of the irradiation duration is thus the sum of all the $(A_n)_i$ terms.

9.6.4.4 If the product of $(\lambda'_i t_i)$ is very small for all irradiation periods, the values of A_i calculated from Eq 17 are proportional to $(R_s)_i$ and t_i.

9.6.4.5 If the spectrum averaged cross section is also constant over all irradiation periods, $(R_R)_i$ is proportional to the magnitude of the neutron fluence rate.

9.6.4.6 It is normally assumed that the fluence rate is directly proportional to the power generation rate in the adjacent fuel.

9.6.5 Under the conditions assumed in 9.6.4.4, Eq 17 can be written as:

$$A_i = A_S (P_i/P)\,(1 - \exp(-\lambda' t_i)) \tag{19}$$

and Eq 18 can be written as:

$$(A_n)_i = A_s \left(\frac{N_i}{N_o}\right) K_i\,(1 - \exp(-\lambda' t_i)) \tag{20}$$

where:

A_s = the saturation activity corresponding to a reference power level, P,
P_i = actual power generation rate during the irradiation period,
K_i = $(P_i/P)\exp\left(-\lambda' \sum_{j=i+1}^{n} \left[1 + \frac{P_j}{P}\left(\frac{\lambda'}{\lambda} - 1\right)\right] t_j\right)$, and
N_i = $N_o \exp\left(-R_R \sum_{j=1}^{i-1} \frac{P_j}{P} t_j\right)$.

NOTE 2—For a single irradiation period at reference power, $K_i = 1.000$ and Eq 20 reduces to Eq 12.

9.6.6 In some cases radioactive products are also produced from radioactive nuclei that built in (for example, fission products produced from ^{239}Pu that arises from neutron capture in ^{238}U). In these cases the number of atoms of the new target isotope(s) must be calculated for each time interval and Eq 17 used to determine the additional activity to be added to that from the original target nuclide.

9.7 The number of target nuclei can often be assumed to be equal to N_0, the number prior to irradiation.

$$N_0 = N_A F m/M \tag{21}$$

where:

N_A = Avogadro's number
= 6.022×10^{23} mole $^{-1}$,
F = atom fraction of the target nuclide in the target element,
m = mass of target element, g, and
M = atomic mass of the target element.

9.7.1 Calculations of the isotopic concentration after irradiation is discussed in 9.5 and in the detailed methods for individual detectors.

9.7.2 Cross sections should be processed from an appropriate cross-section library that includes covariance data. Guide E1018 provides information and recommendations on how to select the cross section library. The International Reactor Dosimetry File (IRDF-2002) **(3)** is one good source for cross sections, as is its successor, the International Reactor Dosimetry and Fusion File **(4)**. The SNLRML cross section compendium **(5)** provides a processed fine-group representation of recommended dosimetry cross sections and covariance matrices.

9.7.3 If spectrum-averaged cross-section or spectrum data are not available, one of the alternative procedures discussed in 9.10 to 9.12 may be used to calculate an approximate neutron fluence rate from the saturation activity.

9.8 *Lethargy:*

9.8.1 For certain purposes it is more convenient to describe a neutron fluence spectrum in terms of fluence per unit lethargy, $\Phi(U)$, rather than in terms of fluence per unit energy, $\Phi(E)$. Lethargy, U, is defined as follows:

$$U = 1n(E_0/E) \tag{22}$$

where E_0 = an arbitrarily chosen upper energy limit; 10 MeV and 14.918 MeV (0.4 lethargy units above 10 MeV) are energies often chosen for E_0. The relationship between $\Phi(U)$ and $\Phi(E)$ is:

$$\Phi(U)dU = E{\cdot}\Phi(E)\mathrm{d}E \tag{23}$$

Neutron spectra are sometimes plotted as $E{\cdot}\Phi(E)$ versus energy. This allows the plotting of a wide range of fluence on a linear plot and shows $1/E$ portions of the spectrum as horizontal lines. If plotted with a linear lethargy axis and a logarithmic energy axis, equal areas have equal fluence.

9.9 *Neutron Spectra:*

9.9.1 A reactor neutron spectrum can be considered as being divided into three idealized energy ranges describing the neutrons as thermal, epithermal (or resonance), and fast. Since these ranges have distinctive distributions, they are a natural division of neutrons by energy for thermal reactor spectra.

9.9.1.1 The neutrons emitted by fission of ^{235}U have an average energy of approximately 2 MeV and the number of neutrons per unit lethargy interval decreases rapidly on either side of this average energy. The major portion of the neutrons with energies above 1 or 2 MeV are “first-flight” neutrons; that is, fission neutrons that have not lost any of their original energy through interaction with atoms. Thus, the fast-neutron fluence spectra have the shape of the ^{235}U fission spectrum, modified by the non-uniform removal of neutrons from some energy regions by interactions with atoms in the reactor materials.

9.9.1.2 Neutrons are slowed (lose energy) primarily by elastic interactions with atoms; the average energy lost per collision is proportional to the neutron energy before the interaction. Thus, at lower energies where the “slowing-down fluence” becomes much larger than the fluence due to first-flight fission neutrons, $\Phi(U)$ is approximately a constant over a large range of energies and $\Phi(E)$ is approximately inversely proportional to the energy. This is the epithermal or $1/E$ portion of the spectrum.

9.9.1.3 At still lower energies, the energy transfer between the neutrons and atoms is influenced by the thermal vibrations of the atoms. The thermalized neutrons have a distribution that is approximately Maxwellian (except when a strong neutron absorber is present). The Maxwellian distribution is characterized by a neutron temperature, T. The reference neutron temperature, T_0, is defined as 293.6 K. The neutron temperature T is the value which provides the best fit of the neutron spectrum to a Maxwellian form. See definition in paragraph 3.6 of Test Method E262. When the absorption is not strong the neutron temperature should approximate the physical temperature of the scattering medium, though it commonly exceeds it up to 10 %.

9.9.2 The thermal-neutron component overlaps the epithermal-neutron component somewhat while the epithermal-neutron component and the fast-neutron component also overlaps. The exact energy limits between the components are somewhat arbitrary but the choice is influenced by the cross-section characteristics of the isotopes used to detect the neutrons in each energy range. The energy limits adopted for this practice are 0 to 0.55 eV for thermal neutrons, $5\sqrt{T/T_0}(0.0253)\ eV$ to 0.10 MeV for epithermal neutrons, and 0.10 MeV to ∞ for fast neutrons.

9.10 *Thermal-Neutron Fluence Rate:*

9.10.1 The thermal-neutron fluence rate is designated as $(nv)_{th}$. This represents an integral:

$$(nv)_{th} = \int_0^\infty n(v)v\,dv \tag{24}$$

where $n(v)$ is the thermal neutron density as a function of velocity, and v is the velocity.

For a Maxwell-Boltzmann thermal neutron distribution of neutron temperature T:

$$(nv)_{th} = \frac{2}{\sqrt{\pi}} n_{th} v_T \tag{25}$$

$$\left(v_T = v_0\sqrt{\frac{T}{T_0}} = 2200\ \mathrm{m/s\ at}\ T = T_0 = 293.6\ \mathrm{K} \right) \tag{26}$$

9.10.2 Many of the reaction cross sections for thermal neutrons have a $1/v$ dependence. For those cases, the reaction rate is independent of the neutron temperature. This is because the reaction rate is proportional to the neutron velocity times the neutron cross section. For $1/v$ cross sections, these terms cancel. This allows the use of the conventional thermal neutron fluence rate φ_0:

$$\varphi_0 = n_{th} v_0 \tag{27}$$

Here, n_{th} is the thermal neutron density, and v_0 is an arbitrary neutron velocity, usually taken to be 2200 m/s, the most probable speed of the Maxwellian distribution for a standard temperature of 20.44°C (293.6 K). In this case, the reaction rate $R_{R,th}$ for thermal neutrons is:

$$R_{R,th} = \int \sigma(E) n_{th}(E)\ v dE = \int \frac{\sigma_o v_o}{v} n_{th}(E) v dE = \ \sigma_0\ v_0 n_{th} = \sigma_0 \varphi_0 \tag{28}$$

σ_0 is the neutron cross section corresponding to v_0, and is usually called the 2200 m/s cross section.

9.10.2.1 Thermal neutron cross sections are usually tabulated as the value for a neutron speed of v_0 = 2200-m/s (See Table 1). Conventions for thermal neutron fluence rate (see Annex A1) use the neutron density multiplied by the standard speed of 2200 m/s. The conventional thermal neutron fluence rate, φ_0, is the same as that used in the Stoughton and Halperin convention, see Eq A1.4.

NOTE 3—Using the neutron density times v_0 is not the same as using the thermal neutron density times v_0.

9.10.3 A detailed procedure for the measurement of thermal-neutron fluence rate is given in Test Method E262. See also Test Method E481.

9.10.3.1 There have been many misunderstandings among experimenters because various conventions for expressing thermal fluence are in use. See Annex A1. For example, the

TABLE 1 Thermal-Neutron Detectors

Element	Reaction	Thermal Cross Section[A] (b)	Product Nucleus[B]					Comments
			Half-Life[C]	E_γ[D] (keV)	Yield (%)[D] γ per Reaction	End Point E_β[D] (keV)	Yield (%)[D] β per Reaction	
Dysprosium	$^{164}Dy(n,\gamma)^{165}Dy$	2650. ± 3.8 %	2.334(1) h	94.700(3)	3.80(5)	1286.6(10)	83.(2)	E
				715.328(20)	0.578(9)	1191.9(9)	15.(2)	
				1079.63(3)	0.0999(17)	291.5(9)	1.7(2)	
Indium	$^{115}In(n,\gamma)^{116m}In$	166.413 ± 0.6 %	54.29(17) min	1293.56(2)	84.8(12)	1014(4)	54.2(6)	F
				1097.28(2)	58.5(8)	876.(4)	32.5(3)	
				818.68(2)	12.13(14)	604.(4)	10.3(14)	
				2112.29(2)	15.09(22)			
Gold	$^{197}Au(n,\gamma)^{198}Au$	98.69 ± 0.14 %	2.6948(12) d	1087.6842(7)	0.1589(18)	960.5(5)	98.990(9)	G,H
				675.8836(7)	0.805(5)			
				411.80205(17)	95.62(47)			
Cobalt	$^{59}Co(n,\gamma)^{60}Co$	37.233 ± 0.16 %	1925.28(14) d	1173.228(3)	99.85(3)	317.05(20)	99.88(3)	G
				1332.492(4)	99.9826(6)	1490.29(20)	0.12(3)	
Manganese	$^{55}Mn(n,\gamma)^{56}Mn$	13.413 ± 1.5 %	2.5789(1) h	846.7638(19)	98.85(3)	2848.86(21)	56.6(7)	G
				1810.726(4)	26.9(4)	1038.09(21)	27.5(4)	
				2113.092(6)	14.2(3)	735.70(21)	14.5(3)	
Sodium	$^{23}Na(n,\gamma)^{24}Na$	0.528 ± 0.95 %	14.997 (12) h	1368.626(5)	99.9935(5)	1392.56(8)	99.855(5)	G
				2754.007(11)	99.872(8)			

[A] 2200 ms cross section (E = 0.0253 eV, T = 20°C), taken from the cross section files recommended in Ref (**5**). Uncertainty data is taken from Ref (**6**) for all thermal cross sections unless otherwise noted.
[B] Sources for half life and gamma radiation data in this table are consistent with that from Ref (**5**).
[C] Original source is Ref (**7**).
[D] Original source is Ref (**6**).
[E] Source for cross section is Ref (**6**). This dosimetry reaction is not in Ref (**5**).
[F] This number represents an update of information in Ref (**5**) and represents an update in the original source data.
[G] Original source for decay radiations is Ref (**8**). This reference is a standard for detector calibration and takes precedence for isotopes used as calibration standards.
[H] Cross sections and uncertainty come from Ref (**9**).

conventional 2200 m/s thermal-neutron fluence rate, φ_0, is not the thermal-neutron fluence rate $(nv)_{th}$ from Eq 25

$$\text{but a factor } \frac{2}{\sqrt{\pi}}\sqrt{\frac{T}{T_0}} \text{ smaller:}$$

$$(nv)_{th} = 1.128\sqrt{\frac{T}{T_0}}\varphi_0 \tag{29}$$

where, in Eq 26 and Eq 29:

T = the thermal neutron temperature, a parameter of the Maxwellian distribution chosen to best fit the actual thermal neutron spectrum.

9.10.3.2 The thermal neutron temperature is typically larger than the physical temperature of the moderator because of the spectrum-hardening caused by absorption. Because the neutron temperature is often unknown, the conversion from the conventional 2200 m/s thermal-neutron fluence-rate to the true thermal-neutron fluence rate using Eq 29, is not usually done, nor is that necessary for the calculation of other reaction rates whose cross sections are approximately proportional to $1/v$.

9.10.4 It is strongly recommended that Ref (**10**) be studied, particularly with regard to the issue of corrections required for the Maxwellian temperature of the thermal neutrons and for the departure of activation detector cross section from a $1/v$ behavior. Westcott (**11**) and others subsequently (**12**) have tabulated correction factors, known as Westcott-g-factors, which correct the response of tabulated reactions for departures of their cross section from the ideal $1/v$ response in a Maxwellian thermal spectrum, at various neutron temperatures, T. The Wescott g-factor is used in all the thermal neutron fluence conventions discussed in Annex A1, not just the Westcott convention.

9.10.5 When the Westcott g-factor is used, Eq 28 becomes:

$$\varphi_0 = n_{th}v_0 = R_{R,th}/g\sigma_0 \tag{30}$$

9.10.6 In order to separate the activities due to thermal and epithermal neutrons, bare and cadmium-covered foils are exposed under identical conditions and the activities measured. The method, called the cadmium-difference method, is based on the fact that cadmium is an effective absorber of neutrons below some energy, E_{Cd}, but it passes neutrons of energies above E_{Cd}. E_{Cd} is known as the "effective cadmium cut-off energy" (see Terminology E170). Its value depends upon the cadmium thickness and other factors (**13, 14**). For a 1-mm thick cadmium shield in an isotropic neutron field, E_{Cd} may be taken to be about 0.55 eV. The cadmium ratio (E170), CR, for a given neutron flux is:

$$CR = \frac{R_B}{R_{Cd}} \tag{31}$$

where R_B and R_{Cd} = the reaction rates for the bare and cadmium-covered configurations, respectively. When both epithermal and thermal neutrons are present in the radiation field, an expression relating the subcadmium fluence rate due to neutron of energies below E_{Cd} to the reaction rate, R, observed for a bare detector, is as follows:

$$\varphi_{sc} = \frac{R}{g\sigma_o}\frac{CR-1}{CR} \tag{32}$$

where φ_{sc} is the conventional 2200 m/s subcadmium-neutron fluence rate, which is approximately equal to the 2200 m/s thermal-neutron fluence rate, φ_0, with small correction factors that are defined in A1.2.3 and A1.3.4.

9.10.7 A knowledge of the thermal-neutron fluence rate is often important in making fast-neutron fluence rate measurements because of interfering activities produced as a result of thermal-neutron absorption by the nuclide being activated, by its activation products, or by impurities in the test specimen. Also there may be a reduction in the measured activity because of the transmutation loss or "burn-up" of the activation product of the fast-neutron reaction due to thermal-neutron absorption. Furthermore, thermal-neutron measurements are necessary in connection with reactor physics analysis and in order to predict the radioactivity in reactor components. Finally, although thermal neutrons are not generally capable of producing radiation damage in materials by direct neutron collision, indirect mechanisms exist for thermal-neutron damage. One such mechanism is associated with the atomic displacements produced upon atomic recoil following thermal-neutron absorption and the emission of a capture gamma ray.

9.11 *Epithermal-Neutron Fluence Rate:*

9.11.1 In this section, we consider the detection of neutrons with energies extending from those of thermal neutrons to about 0.1 MeV. These neutrons are called epithermal neutrons or resonance neutrons. In this range of energies, the neutron absorption may be divided into two parts. For the first, the cross section varies as the reciprocal of the neutron velocity. The second is "resonance absorption," that is characterized by a large increase in cross section over a narrow energy range. For the slowing-down spectrum of certain types of nuclear reactors, the neutron fluence spectrum in the epithermal range of energies may be considered to be inversely proportional to the energy. In these cases, we may write the following:

$$\varphi(E) = \frac{\varphi_e}{E} \tag{33}$$

from which it can be shown that the epithermal fluence rate parameter, φ_e, is the fluence rate per unit interval in $\ln(E)$.

9.11.2 The cross section for 1/*v*-absorption is inversely proportional to the speed of the neutron or to the square-root of the neutron energy, so that we may write the following:

$$\sigma_{1/v} = k_0/\sqrt{E} = \sigma_0\sqrt{E_0/E} \tag{34}$$

where:

$E_0 = kT_0 = 0.0253$ eV

Also let the resonance absorption cross section be σ_{res}. Then the reaction rate for the cadmium-covered detector is given as follows:

$$R_{\text{Cd}} = \int_{E_{Cd}}^{\infty} \sigma_{1/v}(E)\,\varphi(E)\,dE + \int_{E_{Cd}}^{\infty} \sigma_{res}(E)\varphi(E)\,dE \tag{35}$$

The analysis of Dancoff (**15**, **16**) makes use of the above expressions and shows that, if the fluence has a 1/*E* dependence

$$\varphi_e = \frac{\varphi_{\text{th}}}{2(1+\alpha)(CR-1)}\sqrt{\frac{E_{Cd}}{E_{\text{th}}}} \tag{36}$$

where $E_{\text{th}} = kT$ is the energy of neutrons in thermal equilibrium with the environment (the energy corresponding to the most probable velocity in the Maxwellian distribution) and α is given as follows:

$$\alpha = \frac{\int_{E_{Cd}}^{\infty} \sigma_{res}(E)\varphi(E)\,dE}{\int_{E_{Cd}}^{\infty} \sigma_{1/v}(E)\varphi\,(E)\,dE} \tag{37}$$

or from Eq 33 and Eq 34

$$\alpha = \left(\frac{\sqrt{E_{Cd}}}{2k_0}\right)\int_{E_{Cd}}^{\infty} \frac{\sigma_{\text{res}}(E)}{E}\,dE \tag{38}$$

The validity of Eq 33 may be tested by determining φ_e with several detectors. If the values of φ_e are not equal, this is an indication that Eq 33 is not an appropriate assumption. The integral:

$$\int_{E_{Cd}}^{\infty} \frac{\sigma_{\text{res}}(E)}{E}dE \tag{39}$$

in Eq 38 is known as the reduced resonance integral. Tabulations of the resonance integral are available (**12**, **17**, **18**, **19**, **20**), for most resonance detectors. These are usually tabulated for either $E_{Cd} = 0.5$ eV or $E_{Cd} = 0.55$ eV. The difference is small unless there is a resonance in the neighborhood of the cadmium cut-off energy. For a 1/*v*-detector, the resonance integral is reduced by $0.0225\sigma_0$ if E_{Cd} is increased from 0.5 eV to 0.55 eV. For thick detector foils, the "effective resonance integral" (**18**) must be used, that includes corrections for self-shielding, Doppler broadening of the resonances, and resonance fluence depression. In some tabulations, the term "resonance integral" is taken to include the 1/*v*-absorption contribution. In Table 2, values of the resonance integral are given that include the 1/*v*-absorption contribution. Also, the values refer to infinitely thin foils.

9.12 *Fast-Neutron Fluence Rate:*

9.12.1 The energy at which to separate "fast neutrons" from "resonance energy neutrons" is arbitrarily chosen here to be 0.1 MeV. The spectral shape as given by the differential fluence

TABLE 2 Resonance Integrals for Various Detector Materials

Reaction	Cross Section[A,B]	
	(barn)	% Uncertainty
^{235}U(n,f)FP	269.27	0.27 %
^{238}U(n,f)FP	2.162 x 10^{-3}	9.2 %
^{239}Pu(n,f)FP	286.96	0.26 %
^{241}Am(n,f)FP	8.2550	2.1 %
^{237}Np(n,f)FP	0.21446	27.2 %
^{23}Na(n,γ)^{24}Na	0.311	3.2 %
^{45}Sc(n,γ)^{46}Sc	12.0	4.2 %
^{58}Fe(n,γ)^{59}Fe	1.7	5.9 %
^{59}Co(n,γ)^{60}Co	75.9	2.6 %
^{63}Cu(n,γ)^{64}Cu	4.97	1.6 %
^{115}In(n,γ)^{116m}In	3300	3.0 %
^{197}Au(n,γ)^{198}Au	1550	1.8 %
^{232}Th(n,γ)^{233}Th	85	3.5 %
^{238}U(n,γ)^{239}U	277	1.1 %
^{6}Li(n,X)^{4}He	422.45	0.14 %
^{10}B(n,X)^{4}He	1721.06	0.16 %

[A] Resonance integral uses a 1/E function for the source term and uses integration limits of 0.5 eV and 100 keV. The fission and (n, α) integrals were computed with the NJOY94 code using a resonance reconstruction temperature of 300°C and a reconstruction accuracy of 0.1 %. The (n,γ) capture integrals were taken from Ref (**12**).

[B] Source for cross sections and covariance matrices is consistent with the recommendations in Ref (**5**).

rate, $\varphi(E)$, can in principle be determined from the measured reaction rates of several detectors that are activated by different parts of the neutron energy spectrum. The effective cross section $\bar{\sigma}_j$ is calculated from Eq 3 for a known spectrum similar to the spectrum for the unknown field being measured; the integral fluence rate is then calculated from the measured reaction rate for the detector according to Eq 5. A number of detectors are exposed for which trial values of $\bar{\sigma}_j$ have been calculated. The resulting reaction rates can be analyzed, yielding a curve for $\varphi(E)$ versus E. The method of obtaining the adjusted neutron energy spectrum, fluence rate, and fluence is discussed in Practice E944, in which the applicable computer codes are reviewed. Note that proper application of the procedure requires prior information on the spectrum shape which should be obtained by means of neutron transport calculations. The measured reaction rate data result in improved precision in the adjusted neutron spectrum.

9.12.2 An alternate procedure is to consider the detectors as having threshold properties. For an ideal threshold detector, the cross section for activation is a step-function; that is, it is zero for neutrons with energies below a certain energy E_i (the "threshold energy") and constant for neutron energies above E_i. The constant value at energies above E_i is the "threshold cross section," $\bar{\sigma}_j(E > E_i)$. Then the effective threshold cross section for the assumed spectrum, obtained where possible from a transport calculation, is given as follows:

$$\bar{\sigma}_j(E>E_i) = \left[\int_0^\infty \sigma(E)\varphi(E)\,dE\right] / \left[\int_{E_i}^\infty \varphi(E)\,dE\right] \quad (40)$$

The integral in the denominator of this equation is the integral neutron fluence rate with energies above E_i, $\varphi(E > E_i)$. Hence, the integral fluence rate above energy E_i is given, in this ideal case, as follows:

$$\varphi(E>E_i) = \frac{R_R}{\bar{\sigma}\,(E>E_i)} \quad (41)$$

where, as before, R_R, the reaction rate, is determined experimentally from Eq 13. If $\varphi(E > E_i)$ is determined for a number of detectors, that is, for a number of values of E_i, the differential fluence rate, $\varphi(E)$, can be deduced by differentiating the curve of $\varphi(E > E_i)$ versus E_i.

9.12.3 Another procedure for obtaining a neutron fluence from a set of threshold detectors using Eq 5 is to calibrate each detector in a benchmark neutron field. Fission spectrum neutrons are available for this purpose, and for many applications, to provide an appropriate energy spectrum **(21, 22, 23, 24)**. Such a calibration is useful for reducing experimental and interpretational errors quite apart from the method employed to reduce counting data to a neutron fluence.

9.12.3.1 The fission spectrum is the most widely and consistently studied fast-neutron energy distribution. A host of documented differential measurements define the ^{252}Cf and ^{235}U fission neutron spectra to better than ±2.5 % and ±5 % respectively, over the primary energy range of 0.25 to 8 MeV **(24, 25, 26)**. The use of standard fission fields is described in Guide E2005.

9.12.3.2 The form of Eq 5 for establishing a neutron fluence from individual threshold detector responses based on fission spectrum calibration is as follows:

$$\Phi_s = \frac{R_R}{R_\chi} \times \frac{\bar{\sigma}_\chi}{\bar{\sigma}_s} \times \Phi_\chi \quad (42)$$

where:

Φ_s = calibrated neutron fluence for the neutron field under study,

R_R = measured detector response in the study spectrum (a reaction probability, disintegration rate, or any reproducible activation detector response quantity, for example, the gamma counting rate at a fixed time, corrected for irradiation time history),

R_χ = measured detector response (equivalent to R_R) for the fission spectrum calibration,

$\bar{\sigma}_s$ = spectrum-averaged cross section for the study spectrum (obtained with a spectrum from neutron transport calculation or from spectrum adjustment schemes based on the distinctive threshold responses of a set of integral detectors as described in 9.12.1, 9.12.2, and elsewhere),

$\bar{\sigma}_\chi$ = spectrum-averaged cross section for the fission spectrum (calculated with the same detector cross sections used for $\bar{\sigma}_s$, and with the fission spectrum shape associated with the spectrum assumed for the neutron field under study, for example, the fission source spectrum in a neutron transport calculation), and

Φ_χ = fission neutron fluence for the detector calibration exposure.

9.12.3.3 Spectrum-averaged cross sections calculated for the ^{235}U and ^{252}Cf benchmark fission standard fields using the SNLRML recommended cross sections **(5)** are given in Table 3. The 5 %, 50 %, and 95 % response ranges are also indicated in the table.

9.12.3.4 Neutron fluence measurements with activation detectors depend upon gamma detection efficiencies, number of detector atoms and isotopic abundance, decay constants, branching ratios, fission yields, and competing activities. Calibration in a benchmark neutron field can eliminate most of these detector response factors or reduce their error contribution. In addition, the uncertainty in absolute cross section scales, which is generally difficult to assess, is not involved in the cross section ratio and hence in the fluence determination. Furthermore, the effects of cross section shape errors are reduced to the extent that the benchmark and study spectra are similar.

9.12.3.5 The magnitude of cross section errors for activation detectors used in dosimetry may be judged from the disagreement between measurement and prediction for fission neutron spectra. An extensive data base for ^{252}Cf and ^{235}U fission sources exists and has been evaluated. The resulting C/E ratios are presented in Table 3.

9.13 *Fission Threshold Detectors:*

9.13.1 The fission threshold detectors are particularly important as fast-neutron detectors because their effective threshold energies lie in the low MeV range (see Table 3). The detection of neutrons in this range of energies is of special interest because the peak in the fission spectrum occurs at about 1 MeV. Also, in connection with radiation damage to materials, due to ionization effects at higher energies the

TABLE 3 Fission-Spectrum-Averaged Cross Sections and Related Parameters for Threshold Activation Detectors

Reaction[A]	^{252}Cf Spontaneous Fission Field[B]			^{235}U Thermal Fission Field[C]			^{252}Cf Response Range (MeV)[D]		
	Calculation[E] (mb)	Observation[F] (mb)	C/E[G]	Calculation[E] (mb)	Observation[F] (mb)	C/E[G]	Low E_{05}	Median E_{50}	High E_{95}
^{237}Np(n,f)FP	1335.046 (9.2 %, 0.23 %)	1361.0 (1.58 %)	0.981 (9.43 %)	1330.114 (9.33 %, 4.31 %)	1344.0 (4.0 %)	0.9897 (11.0 %)	0.69	2.03	6.06
^{238}U(n,f)FP	315.39 (0.53 %, 0.4 %)	325.0 (1.63 %)	0.970 (1.76 %)	306.23 (0.53 %, 4.21 %)	309.0 (2.6 %)	0.991 (4.98 %)	1.45	2.73	7.12
^{103}Rh(n,n')^{103m}Rh	714.45 (3.08 %, 0.27 %)	757.0 (4.0 %)	0.944 (5.06 %)	706.02 (3.1 %, 4.14 %)	733.0 (5.2 %)	0.963 (7.33 %)	0.74	2.34	6.05
^{93}Nb(n,n')^{93m}Nb	142.65 (3.04 %, 0.36 %)	149.0 (7.0 %)	0.957 (7.64 %)	139.97 (3.06 %, 4.14 %)	146.2 (8.6 %)	0.957 (10.02 %)	0.97	2.67	6.08
^{115}In(n,n')^{115m}In	189.8 (2.16 %, 0.38 %)	197.6 (1.3 %)	0.961 (2.55 %)	186.35 (2.17 %, 4.17 %)	190.3 (3.84 %)	0.979 (6.07 %)	1.13	2.63	6.15
^{47}Ti(n,p)^{47}Sc	19.38 (3.76 %, 0.63 %)	19.29 (1.66 %)	1.005 (4.16 %)	17.95 (3.7 %, 4.26 %)	19.0 (7.4 %)	0.946 (9.3 %)	1.75	3.80	8.17
^{32}S(n,p)^{32}P	70.44 (4.01 %, 0.75 %)	72.62 (3.5 %)	0.970 (5.38 %)	64.69 (4.0 %, 4.86 %)	66.8 (5.54 %)	0.968 (8.39 %)	2.31	4.03	7.74
^{58}Ni(n,p)^{58}Co	115.31 (2.40 %, 0.73 %)	117.6 (1.3 %)	0.981 (2.83 %)	105.69 (2.43 %, 4.52 %)	108.5 (5.0 %)	0.974 (7.16 %)	2.05	4.08	7.90
^{54}Fe(n,p)^{54}Mn	88.12 (2.14 %, 0.79 %)	86.92 (1.34 %)	1.014 (2.65 %)	80.18 (2.17 %, 4.69 %)	80.5 (2.86 %)	0.996 (5.91 %)	2.32	4.23	7.93
^{46}Ti(n,p)^{46}Sc	12.56 (2.45 %, 1.18 %)	14.09 (1.76 %)	0.891 (3.24 %)	10.43 (2.46 %, 5.4 %)	11.6 (3.45 %)	0.899 (6.86 %)	3.76	5.90	9.92
^{56}Fe(n,p)^{56}Mn	1.370 (2.23 %, 1.45 %)	1.466 (1.77 %)	0.934 (3.20 %)	1.029 (2.33 %, 6.58 %)	1.09 (3.67 %)	0.944 (7.89 %)	5.51	7.49	11.91
^{63}Cu(n,α)^{60}Co	0.678 (2.83 %, 1.38 %)	0.689 (1.98 %)	0.984 (3.72 %)	0.521 (2.85 %, 6.05 %)	0.50 (11.0 %)	1.042 (12.87 %)	4.65	7.24	11.52
^{27}Al(n,α)^{23}Na	1.04 (1.36 %, 1.61 %)	1.017 (1.47 %)	1.019 (2.57 %)	0.727 (1.40 %, 6.95 %)	0.706 (3.97 %)	1.030 (8.13 %)	6.53	8.61	12.44

[A] Cross section and covariance matrices are consistent with the sources detailed in Ref (**5**).
[B] Spectrum taken from Ref (**27**), MT = 9861, MF = 5, MT = 8. Uncertainty in the spectrum is taken from Ref (**27**), MT = 9861, MF = 35, MT = 18.
[C] Spectrum taken from Ref (**27**), MT = 9228, MF = 5, MT = 18. Uncertainty in the spectrum is taken from Ref (**28**).
[D] One half of the detector response occurs below an energy given by E_{50}; 95 % of the detector response occurs below E_{95} and 5 % below E_{05}.
[E] The cross section and spectrum components of uncertainty, respectively, appear in parentheses.
[F] Observed cross sections are taken from Refs (**29**), (**30**), (**31**), and (**32**). The measurement uncertainty appears in parentheses.
[G] The uncertainty in the ratio represents a sum in quadrature of the experimental uncertainty and the calculated uncertainty.

effectiveness of neutrons in producing damage does not increase appreciably above a few MeV **(33, 34)**. See also Practices E693 and E722.

9.13.2 The techniques for the use of fission threshold detectors depend upon the measurement of the activities of one or more of the fission products. The expressions given previously apply as well as to the fission threshold detectors except that the fission yield, *Y*, must be inserted, for example, Eq 41 becomes:

$$\varphi(E>E_i) = \frac{R_R}{Y\cdot\bar{\sigma}(E>E_i)} \quad (43)$$

Tabulations are available (see Test Methods E704 and E705) for the fission yields of various fission products for several fissionable nuclides and for thermal and fast neutrons.

9.13.3 Several methods have been used for the collection and counting of the fission products; these include (*1*) direct counting of the fission foil, (see Test Methods E704 and E705), (*2*) an aluminum catcher technique in which ^{237}Np fission recoils are deposited on a thin aluminum catcher foil and then counted **(35, 36, 37)**, and (*3*) the counting of ^{140}Ba, ^{137}Cs or other fission products, following chemical separation from the fission foils (refer to Test Methods E343 and E393), and (*4*) radiation damage techniques such as Test Method E854.

9.13.4 The use of ^{239}Pu or ^{235}U as a fission threshold detector represents a special case. Since ^{239} Pu and ^{235}U have high thermal cross section, bare ^{239}Pu or ^{235}U samples will not function as threshold detectors. However, when the ^{239}Pu or ^{235}U is shielded with ^{10}B, a threshold is introduced whose threshold energy increases with the thickness of ^{10}B. A thickness of ^{10}B of 10 mm produces a threshold at about 1 keV.

10. Report

10.1 Since it may be anticipated that new information concerning cross-section values and other parameters will become available in the course of time, it is important to report more than the mere fluence values resulting from a given fluence determination. If sufficient detail is given in reporting fluence measurements, subsequent revisions in the results may be made on the basis of the newer information. The following is a check list of items to be included, if applicable:

10.1.1 The weight, shape, and chemical and isotopic analysis of the sample,

10.1.2 A description or drawing of the irradiation geometry and capsules, including information concerning the thickness and the shape of thermal neutron shields,

10.1.3 Information regarding duration of exposure, time from termination of exposure until start of counting, elapsed time during counting, description of reactor power variation during exposure, and the temperatures of the detector materials and the moderator in the vicinity of the test,

10.1.4 A description of the counting apparatus, including calibration methods and counter efficiency,

10.1.5 The activity at the time of measurement and the calculated reaction rates including values of the decay constants used in the calculations,

10.1.6 Measured or estimated amounts of impurity-induced activities,

10.1.7 Values of cadmium ratios,

10.1.8 Calculated values of φ_o (Eq 30) for the 2200 m/s fluence rate, φ_e (Eq 36) for the epithermal fluence rate, $\varphi(E > E_i)$ (Eq 41) for the threshold detectors, and Φ_s (Eq 42) for fission-spectrum-related neutron fields. Values of the cross sections used to relate the fluence rates to the measured saturation activities and, in the case of threshold detectors, the assumed effective threshold energies must also be given, and

10.1.9 Error values associated with the various measured and calculated quantities.

11. Precision and Bias

NOTE 4—Measurement uncertainty is described by a precision and bias statement in this practice. Another acceptable approach is to use Type A and B uncertainty components (see ISO *Guide in the Expression of Uncertainty in Measurement* and Ref (**38**)). This Type A/B uncertainty specification is now used in International Organization for Standardization (ISO) standards, and this approach can be expected to play a more prominent role in future uncertainty analyses.

11.1 Reference should be made to the methods for individual detectors for guidance on the precision and bias of fluence rate and fluence measurements.

11.2 Reference should be made to Practice E944 for information on the problems associated with the determination of neutron spectra and the precision and bias obtainable with the specific computer codes discussed therein.

12. Keywords

12.1 activation; fluence; neutron spectrum; radioactivation

ANNEX

A1. NEUTRON FLUENCE CONVENTIONS FOR THERMAL AND EPITHERMAL NEUTRONS

NOTE A1.1—Material in the annex is based on lecture notes by J.G. Williams, University of Arizona, and is reproduced with permission. See also Ref (**39**).

A1.1 Background

A1.1.1 Three distinct conventions are used for characterizing thermal and epithermal neutron fluencies in moderated systems. In each case the main goal is to provide a prediction, based on test exposures of dosimetry foils or wires, of non-threshold reaction rates induced in materials exposed in such fields.

A1.1.2 The formulae presented in this annex do not include the corrections that may be needed to account for thermal or epithermal self-shielding or external fluence-depression. Therefore, these formulae are applicable only for thin or dilute materials in which these effects are negligible, or to reaction rates that have been appropriately corrected.

A1.1.3 The conventions are the Westcott convention (**11, 40**), the Stoughton and Halperin convention (**41**) and the Høgdahl convention (**42, 43**). The use of each of them is permissible and the choice depends on the particular purpose and methods of each test. All three are used in ASTM standards: E261 is based on the Høgdahl convention, Test Method E262 is based on the Stoughton and Halperin convention, and Test Method E481 is based on the Westcott convention.

A1.1.4 In each case some modeling assumptions are made about the shapes of the thermal and epithermal spectra. Westcott (**11**) specified some restrictive limitations on the nature of the fields to which his convention is applicable. These restrictions ought to apply similarly to the other conventions, though deviations in the epithermal spectrum shape have only a slight effect in estimation of the thermal fluence part, and vice versa. Severe errors due to modeling bias in the thermal or subcadmium neutron spectrum can arise for reactions, including ^{239}Pu fission or ^{115}In radiative capture, in which the cross section deviates greatly from $1/v$-dependence, at energies close to, or below, the cadmium filter cut-off. Bias errors that may arise from deviations from the idealized spectra should be considered in estimation of uncertainties.

A1.1.5 The Westcott convention is applicable when no cadmium filter is used, but may be modified to include the effect of cadmium filters.

A1.1.6 The Stoughton and Halperin convetion may be used with or without cadmium filters.

A1.1.7 The Høgdahl convention requires the use of cadmium filters to determine the respective fluence components.

A1.1.8 The purpose of this annex is to clarify the distinctions between the conventions, to define the relevant quantities and notations, to emphasize the requirement to report which convention is used in a particular test, and to provide formulae for conversion of results from one convention to another. In some cases the conversion requires the Maxwellian neutron temperature to be estimated.

A1.1.9 In each case the conventional fluence rate is defined as the product of a neutron density and the standard speed of 2200 m/s. For the Westcott convention the neutron density used is the total of the thermal and epithermal neutron density. For the Stoughton and Halperin convention, the conventional thermal neutron fluence rate is the product of the Maxwellian thermal neutron density and the standard neutron speed of 2200 m/s. For the Høgdahl convention, the conventional subcadmium fluence rate is the product of the neutron density below the cadmium filter threshold (at approximately 0.55 eV) and the standard neutron speed of 2200 m/s. The subcadmium fluence is not the same as the thermal fluence, because part of the epithermal neutron component extends below the cadmium

cut-off, though the distinction has often been incorrectly ignored. The two may differ, typically by up to 2-3 %, in reactor spectra.

A1.1.10 In each convention also, the epithermal component of the differential neutron fluence rate per unit energy is modeled as proportional to 1/*E*:

$$\varphi_{epi}(E) = \frac{\varphi_e}{E}, \quad E > \mu kT \tag{A1.1}$$

where:

φ_e = is the epithermal fluence rate parameter,
μ = a number usually taken equal to 5 for water moderated reactors (**11**), and
T = the neutron temperature of the Maxwellian distribution.

A1.1.11 It has been pointed out that in reactor spectra the epithermal neutron may deviate from 1/*E* (**44, 45**). Versions of each of the three conventions (W, S&H, H) have been used (**46, 47, 48**, respectively) in which Eq Eq A1.1 is modified as follows:

$$\varphi_{epi}(E) = \frac{E_1 \varphi_e}{E^{1+\alpha}}, \quad E > \mu kT \tag{A1.2}$$

where:

α = a non-dimensional parameter that is charcteristic of a particular environment,
E_1 = a reference energy, usually designated as 1 eV, that is needed to ensure dimensional consistency.

A1.2 Calculation of Reaction Rates

A1.2.1 In a bare, thin detector exposed in a thermal and epithermal neutron field, the reaction rate per atom R_B, according to the Westcott convention, is:

$$R_B = \varphi_w \sigma_0 (g + rs) = \varphi_w \sigma_0 \left(g + r\sqrt{\frac{T}{T_0}} S_0 \right) \tag{A1.3}$$

where:

φ_w = the Westcott conventional 2200 m/s neutron fluence rate,
σ_0 = the reaction cross section for neutron speed 2200 m/s,
g = a temperature-dependent factor ("the Westcott *g*-factor") which accounts for the departure of the cross section from 1/*v*-dependence in a Maxwellian-thermal neutron spectrum at neutron temperature, *T*,
r = a factor that represents the proportion of the neutron fluence that is in the epithermal part of the spectrum,
s = a factor that accounts for the departure of the cross section from 1/*v*-dependence in the epithermal energy range.
T_0 = the standard temperature of 293.6 K,
S_0 =

$$\sqrt{\frac{4}{\pi}} \frac{I_0''}{\sigma_0} = \sqrt{\frac{4}{\pi}} \left(\frac{I_0}{\sigma_0} - 2g\sqrt{\frac{E_0}{E_{Cd}}} \right)$$

I_0 = the resonance integral,
I_0'' = the reduced resonance integral that does not include the 1/*v* part of the cross section.

A1.2.2 The parameters φ_w and $r\sqrt{T/T_0}$ may be determined by exposure of two activation monitors with different values of I_0/σ_0; see Practice E481, or by exposure of an activation monitor with and without cadmium or gadolinium filters.

A1.2.3 In a bare, thin detector exposed in a thermal and epithermal neutron field, the reaction rate per atom R_B, according to the Stoughton and Halperin convention, is:

$$R_B = g\sigma_0 \left(\varphi_0 + \varphi_e \left(\frac{I_0}{g\sigma_0} + f_1 + \frac{w'}{g} \right) \right) \tag{A1.4}$$

where:

φ_0 = the conventional 2200 m/s thermal neutron fluence rate,
σ_0 = the reaction cross section for neutron speed 2200 m/s,
g = the Westcott *g*-factor, which accounts for the departure of the cross section from 1/*v*-dependence in a Maxwellian-thermal neutron spectrum at neutron temperature, *T*,
φ_e = the epithermal fluence-rate parameter, as defined in Eq A1.1,
f_1 = a number that represents the 1/*v* response to the portion of the neutron fluence that is in the epithermal part of the spectrum, above the energy μkT, but below the cadmium cut-off energy, E_{Cd}.
w' = a factor that accounts for the departure of the cross section from 1/*v*-dependence in the epithermal energy range, below the cadmium cut-off.
I_0 = the resonance integral, defined as

$$\int_{E_c}^{\infty} (\sigma(E)/E) dE$$

In practice this integral is measured in a standard 1/*E*-field or calculated using an upper energy limit of, typically, 0.1 MeV.

A1.2.4 The parameters φ_0 and φ_e may be determined by exposure of an activation monitor with and without cadmium covers, see Practice E262, or by exposure of two activation monitors with different values of I_0/σ_0. See Practice E481.

A1.2.5 In a bare, thin detector exposed in a thermal and epithermal neutron field, the reaction rate per atom, R_B, according to the Stoughton and Halperin convention, after modification using Eq A1.2 to account for departure of the epithermal spectrum from 1/*E* (**47**) is:

$$R_B = g\sigma_0 \left(\varphi_0 + \varphi_e \left(\frac{I_0}{g\sigma_0} + f_1(\alpha) + \frac{w'}{g} + E_r^{-\alpha} \left[\frac{I_0}{g\sigma_0} - f_2(0) \right] + f_2(\alpha) \right) \right) \tag{A1.5}$$

where:

φ_e = the epithermal fluence rate parameter, as defined in Eq A1.2,

$f_1(\alpha)$ = a number that represents the 1/*v* response to the portion of the neutron fluence that is in the epithermal part of the spectrum, above the energy μkT, but below the cadmium cut-off energy, E_{Cd},

=

$$\frac{2}{1+2\alpha} \left(\sqrt{\frac{E_0}{E_{\mu kT}}} \left(\frac{E_1}{E_{\mu kT}} \right)^{\alpha} - \sqrt{\frac{E_0}{E_{Cd}}} \left(\frac{E_1}{E_{Cd}} \right)^{\alpha} \right)$$

E_r = the effective resonance energy, as defined by Ryves (**49**) and tabulated by Moens et al. (**50**),

$f_2(\alpha)=\frac{2}{1+2\alpha}\sqrt{\frac{E_0}{E_{Cd}}}\left(\frac{E_1}{E_{Cd}}\right)^{\alpha}$ = a number that represents the 1/*v* response to the portion of the neutron fluence that is in the epithermal part of the spectrum, above the cadmium cut-off energy, E_{Cd},

$f_2(0)=2\sqrt{\frac{E_0}{E_{Cd}}}$ = a number that represents the 1/*v* response to the portion of the neutron fluence that is in the epithermal part of the spectrum, above the cadmium cut-off energy, E_{Cd}, in a 1/*E* spectrum.

A1.2.6 Eq A1.5 does not include correction of the w'-factor for departure of the epithermal spectrum from 1/*E* in the part of the epithermal spectrum that is below the cadmium cut-off.

A1.2.7 The parameters, φ_0, φ_e and α may be determined by exposure of two or more activation monitors with and without cadmium covers, or by exposure of three or more activation monitors with different values of I_0/σ_0. The generalized least-squares method has been used **(51)** to simultaneously evaluate flux parameters and nuclear data constants from activation measurements with and without cadmium covers and from cadmium ratios in different neutron fields.

A1.2.8 In a bare, thin detector exposed in a thermal and epithermal neutron field, the reaction rate per atom, R_B, according to the Høgdahl convention, is:

$$R_B = g\sigma_0\,\varphi_{sc}+\varphi_e I_0 \tag{A1.6}$$

where:

φ_{sc} = the conventional 2200 m/s subcadmium neutron fluence rate,
σ_0 = the reaction cross section for neutron speed 2200 m/s,
g = the Westcott *g*-factor, which accounts for the departure of the cross section from 1/v-dependence in a Maxwellian-thermal neutron spectrum at neutron temperature, *T*,
φ_e = the epithermal fluence rate parameter, as defined in Eq A1.1, and
I_0 = the resonance integral, defined as

$$\int_{E_C}^{\infty}(\sigma(E)/E)dE$$

A1.2.9 As originally published **(42)**, the Høgdahl convention did not include the *g*-factor. It is included here, however, and elsewhere in E261 for consistency with the other conventions. The convention still contains a small inherent inconsistency, because the *g*-factor does not account for epithermal neutrons below the cadmium cut-off that are not included in the Maxwellian thermal spectrum. The necessary additional modification is usually ignored, however.

A1.2.10 The parameters φ_{sc} and φ_e may be determined by exposure of an activation monitor with and without cadmium covers. See 9.11 and 9.12.

A1.2.11 In a bare, thin detector exposed in a thermal and epithermal neutron field the reaction rate per atom, R_B, according to the Høgdahl convention, after modification using Eq A1.2 to account for departure of the epithermal spectrum for 1/*E*, is:

$$R_B = g\sigma_0\left(\varphi_{sc}+\varphi_e\left(\frac{I_0}{g\sigma_0}+E_r^{-\alpha}\left[\frac{I_0}{g\sigma_0}-f_2(0)\right]+f_2(\alpha)\right)\right) \tag{A1.7}$$

where:

φ_e = the epithermal fluence rate parameter, as defined in Eq A1.2,
E_r = the effective resonance energy, as defined by Ryves **(49)** and tabulated by Moens et al. **(50)**,
$f_2(\alpha)=\frac{2}{1+2\alpha}\sqrt{\frac{E_0}{E_{Cd}}}\left(\frac{E_1}{E_{Cd}}\right)^{\alpha}$ = a number that represents the 1/*v* response to the portion of the neutron fluence that is in the epithermal part of the spectrum, above the cadmium cut-off energy, E_{Cd},
$f_2(0)=2\sqrt{\frac{E_0}{E_{Cd}}}$ = a number that represents the 1/*v* response to the portion of the neutron fluence that is in the epithermal part of the spectrum, above the cadmium cut-off energy, E_{Cd}, in a 1/*E* specimen.

A1.2.12 Apart from the inclusion here of the *g*-factor, Eq A1.7 is equivalent to the formulae used by Moens et al. **(48)** in the single comparator k_0 method for neutron activation analysis. It does not include the (usually) small correction terms, f_1, for departure of the epithermal neutron spectrum from 1/*E* below the cadmium cut-off energy, or *w'*, for departure of the cross section from 1/*v* in the epithermal part of the spectrum in energy range from μ*kT* to E_{Cd}.

A1.2.13 The parameters φ_{sc}, φ_e and α may be determined by exposure of two or more activation monitors with and without cadmium covers, or by exposure of three or more activation monitors with different values of I_0/σ_0.

A1.2.14 In a cadmium covered, thin detector exposed in a thermal and epithermal neutron field, the reaction rate per atom, R_{Cd}, according to either the Stoughton and Halperin or the Høgdahl convention is:

$$R_{Cd} = \varphi_e I_0 \tag{A1.8}$$

A1.2.15 In a cadmium covered, thin detector exposed in a thermal and epithermal neutron field the reaction rate per atom, R_{Cd}, according to either the Stoughton and Halperin or the Høgdahl convention, after modification using Eq A1.2 to account for departure of the epithermal spectrum from 1/*E* **(47)**, is:

$$R_{Cd} = g\sigma_0\left(\varphi_e\left(\frac{I_0}{g\sigma_0}+E_r^{-\alpha}\left[\frac{I_0}{g\sigma_0}-f_2(0)\right]+f_2(\alpha)\right)\right) \tag{A1.9}$$

A1.3 Conversion Formulae Between Fluence Conventions

A1.3.1 The following formulae may be used to convert values of the Westcott fluence rate parameters to the Stoughton and Halperin convention:

$$\varphi_0 = \left(1-\frac{4r}{\sqrt{\pi\mu}}\right)\varphi_w \tag{A1.10}$$

$$\varphi_e = \frac{2r}{\sqrt{\pi}}\sqrt{\frac{T}{T_0}}\varphi_w \tag{A1.11}$$

A1.3.2 The following formulae may be used to convert values of the Stoughton and Halperin fluence rate parameters to the Westcott convention:

$$\sqrt{\frac{T}{T_0}}r = \frac{\varphi_e}{\varphi_0}\frac{\sqrt{\pi}}{2}\left(1+\frac{2}{\sqrt{\mu}}\sqrt{\frac{T_0}{T}}\frac{\varphi_e}{\varphi_0}\right)^{-1} \tag{A1.12}$$

$$\varphi_w = \varphi_0\left(1+\frac{2}{\sqrt{\mu}}\sqrt{\frac{T_0}{T}}\frac{\varphi_e}{\varphi_0}\right) \tag{A1.13}$$

A1.3.3 The following formula may be used to convert values of the Stoughton and Halperin fluence rate parameters to the Høgdahl convention:

$$\varphi_{sc} = \varphi_0\left(1+\frac{\varphi_e}{\varphi_0}\left(f_1+\frac{w'}{g}\right)\right) \tag{A1.14}$$

A1.3.4 The following formula may be used to convert values of the Høgdahl fluence rate parameters to the Stoughton and Halperin convention:

$$\varphi_0 = \varphi_{sc}\left(1-\frac{\varphi_e}{\varphi_{sc}}\left(f_1+\frac{w'}{g}\right)\right) \tag{A1.15}$$

where the epithermal fluence rate parameter, φ_e, is the same in both conventions.

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