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Designation: E261 – 15

StandardPractice 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E343 Test Method for Measuring Reaction Rates by Analysis of Molybdenum-99 Radioactivity From Fission Dosimeters (Withdrawn 2002)[3]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, E 706 (IIIA)

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.

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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 *Fleunce:*

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 *Spectrrum-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, Np, 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).

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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.

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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. In the absence of a strong absorber, the neutron temperature T_0 corresponds to 293.6°K.

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 Eq 4 may be expressed as follows:

$$R_{R,th} = (nv)_{th}\sigma_{eff} \tag{24}$$

where:

$R_{R,th}$ = the reaction rate attributable to thermal neutrons only,
$(nv)_{th}$ = true thermal-neutron fluence rate;
n = neutron density, neutrons per unit volume,
v = neutron speed, and
σ_{eff} = effective cross-section value chosen to give the correct activation.

The conventional thermal neutron fluence rate, φ_0, is the product of the neutron density times a single neutron velocity, $\varphi_0 = n_{th}v_0$. v_0 is usually 2200 m/s (E170). The thermal neutron reaction rate may be calculated in terms of the conventional neutron fluence rate idealized detector, called a $1/v$-detector, whose cross section is exactly proportional to $1/v$.

$$R_{R,th} = \int \sigma(E)n_{th}(E)\ vdE = \int \frac{\sigma_o v_o}{v} n_{th}(E)vdE = \ \sigma_0\ v_0 n_{th} = \sigma_0\varphi_0 \tag{25}$$

It has become conventional to tabulate cross sections for thermal neutrons as the value for a neutron speed of v_o = 2200-m/s (see Table 1). This is the most probable speed of the Maxwellian distribution for a standard temperature whose value is 20.44°C (293.6 K). For a neutron activation detector with a cross section proportional to $1/v$, the reaction rate is proportional to the neutron density, irrespective of the neutron spectrum. Conventions for thermal neutron fluence rate (see Annex A1) use the neutron density multiplied by the standard speed of 2200 m/s. The "conventional" in the conventional thermal neutron fluence rate is not directly related to any one of the neutron fluence conventions in Annex A1.

9.10.2 A detailed procedure for the measurement of thermal-neutron fluence rate is given in Test Method E262. See also Test Method E481. There have been many misunderstandings among experimenters because various conventions for expressing thermal fluence are in use. See Annex A1. The conventional 2200 m/s thermal-neutron fluence rate, φ_o, is not the thermal-neutron fluence rate as a physical quantity according to the following definition:

$$(nv)_{th} = \int_0^\infty n(v)\ v\ \mathrm{d}v = \frac{2}{\sqrt{\pi}}\, nv_T \tag{26}$$

$$\left(v_{\mathrm{T}} = v_0\sqrt{\frac{T}{T_0}} = 2200\ \mathrm{m/s\ at}\ T = T_0 = 293.6\ \mathrm{K}\right) \tag{27}$$

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

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**).

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$$(nv)_{th} = 1.128\sqrt{\frac{T}{T_0}}\varphi_0 \quad (28)$$

where, in Eq 27 and Eq 28:

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

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 28, is not usually done, nor is that necessary for the calculation of other reaction rates whose cross sections are approximately proportional to $1/v$.

9.10.3 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.

9.10.4 When the Westcott g-factor is used, Eq 25 becomes:

$$\varphi_0 = n_{th}v_0 = R_{R,th}/g\sigma_0 \quad (29)$$

9.10.5 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:

$$CR = \frac{R_B}{R_{Cd}} \quad (30)$$

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:

$$\varphi_{sc} = \frac{R}{g\sigma_o}\frac{CR-1}{CR} \quad (31)$$

where φ_{sc} is the conventional 2200 m/s subcadmium-nuetron 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.

9.10.6 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