Designation: E 261 – 03

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

This standard is issued under the fixed designation E 261; 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 the general 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 E 1005.

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: E 262, E 263, E 264, E 265, E 266, E 343, E 393, E 481, E 523, E 526, E 704, E 705, and E 854.

2. Referenced Documents

2.1 *ASTM Standards:*

E 170 Terminology Relating to Radiation Measurements and Dosimetry[2]

E 181 Test Methods for Detector Calibration and Analysis of Radionuclides[2]

E 262 Test Method for Determining Thermal Neutron Reaction and Fluence Rates by Radioactivation Techniques[2]

E 263 Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Iron[2]

E 264 Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Nickel[2]

E 265 Test Method for Measuring Reaction Rates and Fast-Neutron Fluences by Radioactivation of Sulfur-32[2]

E 266 Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Aluminum[2]

E 343 Test Method for Measuring Reaction Rates by Analysis of Molybdenum-99 Radioactivity from Fission Dosimeters[2]

E 393 Test Method for Measuring Reaction Rates by Analysis of Barium-140 from Fission Dosimeters[2]

E 481 Test Method for Measuring Neutron Fluence Rate by Radioactivation of Cobalt and Silver[2]

E 523 Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Copper[2]

E 526 Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Titanium[2]

E 704 Test Method for Measuring Reaction Rates by Radioactivation of Uranium-238[2]

E 705 Test Method for Measuring Reaction Rates by Radioactivation of Neptunium-237[2]

E 844 Guide for Sensor Set Design and Irradiation for Reactor Surveillance, E 706(IIC)[2]

E 854 Test Method for Application and Analysis of Solid State Track Recorder (SSTR) Monitors for Reactor Surveillance, E 706(IIIB)[2]

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

E 1005 Test Method for Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance, E 706 (IIIA)

E 1018 Guide for Application of ASTM Evaluated Cross Section Data File, Matrix E 706 (IIB)[2]

2.2 *ISO Standard:*

Guide in the Expression of Uncertainty in Measurement

3. Terminology

3.1 Descriptions of terms relating to dosimetry are found in Terminology E 170.

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

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 General Methods E 181.

7. Requirements for Activation-Detector Materials

7.1 The general considerations concerning the suitability of a material for use as an activation detector are found in Guide E 844.

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 E 343, E 393, and E 854, and Guide E 844.

8. Irradiation Procedures

8.1 The irradiations are carried out in two general 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)**.

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/λ', 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 9.11 and 9.12, the use of cadmium-shielded detectors is convenient in separating contributions to the measured activity from thermal and epithermal 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 The activity of the sample, A, at the end of the exposure period is calculated as follows:

$$A = \lambda D/[(1 - \exp(-\lambda t_c))\exp(-\lambda t_w)] \quad (1)$$

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.

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

$$A = D/[t_c \exp(-\lambda t_w)] \quad (2)$$

9.2 For irradiations at constant fluence rate, the saturation activity, A_s, is calculated as follows:

$$A_s = A/(1 - \exp - \lambda' t_i) \quad (3)$$

where:

t_i = exposure duration, and
λ' = effective decay constant during the irradiation.

NOTE 2—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.

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

$$\lambda' = \lambda + \int_0^\infty \sigma_a(E)\phi(E)\,dE \quad (4)$$

where:

$\sigma_a(E)$ = neutron absorption cross section for the product nuclide, and
$\phi(E)$ = neutron fluence rate per unit energy.

9.2.2 Application of the effective decay constant for irradiations under varying fluence rates is discussed in this section and in the detailed methods for individual detectors.

9.3 The reaction rate is calculated as follows:

$$R_s = A_s\lambda'/N\lambda \quad (5)$$

where:

N = number of target nuclei in the detector at time of irradiation.

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

$$N_o = N_A F m/M \quad (6)$$

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.

9.3.2 Calculations of the isotopic concentration after irradiation is discussed in 9.6.6 and in the detailed methods for individual detectors.

9.4 The neutron fluence rate, ϕ, is calculated as follows:

$$\phi = R_s/\bar{\sigma} \quad (7)$$

where:
$\bar{\sigma}$ = the spectral weighted neutron activation cross section.

9.4.1 Cross sections should be processed from an appropriate cross-section library that includes covariance data. Guide E 1018 provides information and recommendations on how to select the cross section library. The International Reactor Dosimetry File (IRDF-90) **(38)** is one good source for cross sections. The SNLRML cross section compendium **(25)** provides a processed fine-group representation of recommended dosimetry cross sections and covariance matrices.

9.4.2 If spectral-averaged cross-section or spectrum data are not available, one of the alternative procedures discussed in 9.10 to 9.13 may be used to calculate an approximate neutron fluence rate from the saturation activity.

9.5 The neutron fluence, Φ, is related to the time varying differential neutron fluence rate $\phi(E,t)$ by the following expression:

$$\Phi = \int_0^\infty \int_{t_1}^{t_2} \phi\,(E,t)\,\mathrm{d}t\,\mathrm{d}E \quad (8)$$

where:
$t_2 - t_1$ = duration of the irradiation period.

9.5.1 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.

9.5.2 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:

$$\Phi = \left(\frac{\phi}{P}\right) \sum_{i=1}^{n} P_i t_i \quad (9)$$

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.

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

9.6 *Transmutation Processes*:

9.6.1 The neutron fluence rate spectrum, $\phi(E)$, can be determined by computer calculations using neutron transport codes, and adjustment techniques using radioactivation data from multiple foil irradiations.

9.6.2 The reaction rate is related to the fluence rate by the following equation:

$$R_s = \int_0^\infty \sigma(E)\phi\,(E)\mathrm{d}E \quad (10)$$

where:
$\sigma(E)$ = activation cross section at energy E, and
$\phi(E)$ = differential neutron fluence rate, that is the fluence per unit energy per unit time for neutrons with energies between E and E + dE.

9.6.3 The production rate of a radioactive nuclide is related to the reaction rate by the following equation:

$$\mathrm{d}n/\mathrm{d}t = N R_s - n\lambda' \quad (11)$$

9.6.4 Solution of Eq 11, for the case where R_s and N are constant, yields the following expression for the activity of a foil:

$$A = (\lambda/\lambda')\,N R_s\,(1 - (\exp - \lambda' t)) \quad (12)$$

9.6.5 The saturation activity of a foil is defined as the activity when dn/dt = 0; thus Eq 11 yields the following relationship for the saturation activity:

$$A_s = (\lambda/\lambda')\,N R_s \quad (13)$$

9.6.6 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:

$$\mathrm{d}N/\mathrm{d}t = N\,(R_s + \sum_{t=1}^{n} R_i) - \sum_{j=1}^{m} N_j R_j \quad (14)$$

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 10, using the appropriate cross sections.

9.6.6.1 The R_s term may predominate and, if R_s is constant, Eq 14 can be solved as $N = N_o \exp(-R_s t)$. The change in the target composition may be negligible and N may be approximated by N_o.

9.6.7 During irradiation, the effective decay rate is increased by transmutations of the product isotope (see Eq 4).

9.7 *Long Term Irradiations*:

9.7.1 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.

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

$$A_i = [\lambda N_i (R_s/\lambda')_i](1 - \exp(-\lambda'_i t_i)) \quad (15)$$

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

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

$$(A_n)_i = A_i \exp\left(-\sum_{j=i+1}^{n} \lambda'_j t_j\right) \quad (16)$$

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

9.7.3 If the product of $(\lambda'_i t_i)$ is very small for all irradiation periods, the values of A_i calculated from Eq 15 are proportional to $(R_s)_i$ and t_i.

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

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

9.7.4 Under the conditions assumed in 9.7.3, Eq 15 can be written as:

$$A_i = A_s (P_i/P)\,(1\,(-\exp(-\lambda' t_i)), \quad (17)$$

and Eq 16 can be written as:

$$(A_n)_i = A_s \left(\frac{N_i}{N_o}\right) K_i\,(1 - \exp(-\lambda' t_i)) \quad (18)$$

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 = $N_o \exp\left(-R_s \sum_{j=1}^{i-1} \frac{P_j}{P} t_j\right)$.

NOTE 3—For a single irradiation period at reference power, $K_i = 1.000$ and Eq 18 reduces to Eq 3.

9.7.5 In some cases radioactive products are also produced from radioactive nuclei that build 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 15 used to determine the additional activity to be added to that from the original target nuclide.

9.8 *Spectral-Averaged Cross Sections*:

9.8.1 As a general practice, the spectral-averaged cross sections will be used in these calculations. Since a spectral-averaged cross section is defined as follows:

$$\bar{\sigma} = \frac{\int_0^\infty \sigma(E)\,\phi(E)\,dE}{\int_0^\infty \phi(E)\,dE} \quad (19)$$

the differential cross section of the nuclide and the neutron spectrum over the neutron energy range for which the nuclide has an effective cross section must be known. When cross-section and spectrum information are not available alternative procedures may be used; suggested alternatives are discussed in 9.11-9.13, and in the methods for individual detectors.

9.9 *Lethargy*:

9.9.1 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:

$$U = 1n(E_0/E) \quad (20)$$

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:

$$\Phi(U) = \Phi(E)E \quad (21)$$

9.10 *Neutron Spectra*:

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

9.10.1.1 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 general 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.

9.10.1.2 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 resonance or $1/E$ portion of the spectrum.

9.10.1.3 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 closely Maxwellian (except when a neutron absorber is present).

9.10.2 The thermal-neutron component overlaps the resonance-neutron component somewhat while the resonance-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 assumed for this procedure are 0 to 0.50 eV for thermal neutrons, 0.50 eV to 0.10 MeV for resonance neutrons, and 0.10 MeV to ∞ for fast neutrons.

9.11 *Thermal-Neutron Fluence Rate*:

9.11.1 A solution of the activation equation, Eq 11, leads to the following result:

$$R_s = (nv)\sigma_{\text{eff}} \tag{22}$$

where:

(nv) = true thermal-neutron fluence rate;
n = neutron density, and
σ_{eff} = effective cross-section value that will give the correct activation.

It has become conventional to tabulate cross sections for thermal neutrons as the value for a neutron velocity of V_o = 2200-m/s (see Table 1). This is the most probable velocity of the Maxwellian distribution for a standard temperature whose value is 20.44°C (293.6 K). Therefore, since the 2200-m/s cross section is more readily available, it was adopted in the thermal-neutron fluence rate notations, $nv_{2200} = \phi_o$ = the 2200-m/s neutron fluence rate. Substituting in Eq 23 and solving for ϕ_o, we have the following modified equation:

$$\phi_o = nv_o = R_s/\sigma_o \tag{23}$$

9.11.2 A detailed procedure for the measurement of thermal-neutron fluence rate is given in Method E 262. There have been many misunderstandings among experimenters because various conventions for expressing thermal fluence are in use. The 2200 m/s fluence rate, ϕ_o, is not the thermal-neutron fluence rate as a physical quantity according to the following definition:

$$(nv)_{\text{th}} = \int_0^\infty n(v)\, v\, dv = \frac{2}{\sqrt{\pi}} nv_T \tag{24}$$

$$(V_T = 2200 \text{ m/s}^{-1} \text{ at } T = 293.6 \text{ K}), \tag{25}$$

but a factor $2/\sqrt{\pi}$ smaller:

$$(nv)_{\text{th}} = 1.128\phi o \tag{26}$$

It is strongly recommended that Ref **(3)** 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.

9.11.3 In order to separate the activities due to thermal and resonance 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_c, but it passes neutrons of energies above E_c. E_c is known as the "effective cadmium cut-off energy." Its value depends upon the cadmium thickness and other factors **(4, 5)**. For a 1-mm thick cadmium shield in an isotropic neutron field, E_c may be taken to be about 0.50 eV. The cadmium ratio, *R*, for a given neutron flux is defined as follows:

$$R = R_B/R_{Cd} \tag{27}$$

where R_B and R_{Cd} = *the reaction rates for the bare and cadmium-covered configurations, respectively. When both resonance and thermal neutrons are present in the radiation field, the expression relating the thermal-fluence rate to the reaction rate, R_s, observed for a bare detector, Eq 23 is modified to read as follows:*

$$\phi_o = \frac{R_s}{\sigma_o} \times \frac{R-1}{R} \tag{28}$$

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	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)	3.5784(168)	1285(10)	83.(2)	[E]
				715.328(20)	0.5342(109)	1190.	15.(2)	
				1079.628(30)	0.0916(28)	305.(10)	1.7(2)	
Indium	$^{115}In(n,\gamma)^{116m}In$	166.413 ± 0.6 %	54.29(17) min	1293.54(15)	84.4(17)	1010.(4)	52.1(12)	[F]
				1097.3(2)	56.21(110)	872.(4)	33.8(15)	
				818.7(2)	11.48(42)	600.(4)	10.2(4)	
				2112.1(4)	15.53(42)			
Gold	$^{197}Au(n,\gamma)^{198}Au$	98.69 ± 0.14 %	2.69517(21) d	1087.684(3)	0.1590(20)	962.(1)	98.990(6)	[G][H]
				675.8836(7)	0.8038(29)			
				411.8044(11)	95.57(47)			
Cobalt	$^{59}Co(n,\gamma)^{60}Co$	37.233 ± 0.16 %	1925.1(5) d	1173.238(4)	99.857(22)	317.88(10)	99.883(21)	[G]
				1332.502(5)	99.983(6)	1492.16(13)	0.117(20)	
Manganese	$^{55}Mn(n,\gamma)^{56}Mn$	13.413 ± 1.5 %	2.5789(1) h	846.754(20)	98.87(30)	2848.9(9)	56.3(10)	
				1810.72(4)	27.19(79)	1038.2(9)	27.9(8)	
				2113.05(4)	14.34(40)	735.9(9)	14.6(4)	
Sodium	$^{23}Na(n,\gamma)^{24}Na$	0.528 ± 0.95 %	14.9512(32) d	1368.633(6)	99.9936(15)	1389.05(29)	99.9310(16)	[G]
				2754.030(5)	99.855(5)			

[A] 2200 ms cross section (E = 0.0253 eV, T = 20°C), taken the cross section files recommended in Reference (**25**). Uncertainty data is taken from Reference (**28**) for all thermal cross sections unless otherwise noted.
[B] Sources for half life and decay radiation data in this table are consistent with that from Reference (**25**) and with Standards E 1018 – 95.
[C] Original source is Reference (**26**).
[D] Original source is Reference (**27**).
[E] Source for cross section is Reference (**28**). This dosimetry reaction is not in Reference (**25**).
[F] This number represents an update of information in Reference (**25**) and represents an update in the original source data.
[G] Original source for half-life and decay radiations is Reference (**28**). This reference is a standard for detector calibration and takes precedent for isotopes used as calibration standards.
[H] Cross sections and uncertainty come from Reference (**30**).

9.11.4 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.

9.12 *Resonance-Neutron Fluence Rate*:

9.12.1 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 resonance neutrons or epithermal 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 resonance range of energies may be considered to be inversely proportional to the energy. In these cases, we may write the following:

$$\phi(E) = \phi_e/E \tag{29}$$

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

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

$$\sigma_{1/v}(E) = k_0/\sqrt{E} \tag{30}$$

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

$$R_{Cd} = \int_{E_c}^{\infty} \sigma_{1/v}(E)\,\phi(E)\,dE + \int_{E_c}^{\infty} \sigma_{res}(E)\phi(E)\,dE \tag{31}$$

The analysis of Dancoff **(6, 7)** makes use of the above expressions and shows that, if the fluence has a 1/E dependence

$$\phi_e = \frac{\phi_{th}}{2(1+\alpha)(R-1)}\sqrt{\frac{E_c}{E_{th}}} \tag{32}$$

where E_{th} 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:

$$\alpha = \int_{E_c}^{\infty} \sigma_{res}(E)\phi(E)\,dE \Big/ \int_{E_c}^{\infty} \sigma_{1/v}(E)\phi(E)\,dE \tag{33}$$

or from Eq 29 and Eq 30

$$\alpha = (\sqrt{E_c}/2k_0)\int_{E_c}^{\infty} \sigma_{res}(dE/E) \tag{34}$$

The validity of Eq 29 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 29 is not an appropriate assumption. The integral:

$$\int_{E_c}^{\infty} \sigma_{res}(dE/E) \tag{35}$$

in Eq 34 is known as the reduced resonance integral. Tabulations of the resonance integral are available **(8, 9, 10, 11)** for most resonance detectors. For thick detector foils, the "effective resonance integral" **(9)** 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.

9.13 *Fast-Neutron Fluence Rate*:

9.13.1 The energy at which to separate "fast neutrons" from "resonance energy neutrons" is rather arbitrarily chosen here to be 0.1 MeV. The spectral shape as given by the differential fluence rate, $\phi(E)$, 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 19 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 7. 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 $\phi(E)$ versus E. The method of obtaining the adjusted neutron energy spectrum, fluence rate, and fluence is discussed in Practice E 944, 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

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 resonstruction temperature of 300°C and a reconstruction accuracy of 0.1 %. The (n,γ) capture integrals were taked from Ref (**39**).

[B] Source for cross sections and covariance matrices is consistent with the recommendations in Ref (**25**).

calculations. The measured reaction rate data result in improved precision in the adjusted neutron spectrum.

9.13.2 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 is given as follows:

$$\bar{\sigma}_j(E > E_i) = [\int_0^\infty \sigma(E)\phi(E)\,dE] / [\int_{E_i}^\infty \phi(E)\,dE] \quad (36)$$

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

$$\phi(E > E_i) = R_s/\bar{\sigma}(E > E_i) \quad (37)$$

where, as before, R_s, the reaction rate, is determined experimentally from Eq 5. If $\phi(E > E_i)$ is determined for a number of detectors, that is for a number of values of E_i, the differential fluence rate, $\phi(E)$, can be deduced by differentiating the curve of $\phi(E > E_i)$ versus E_i.

9.13.3 Another procedure for obtaining a neutron fluence from a set of threshold detectors using Eq 7 is to calibrate each detector in a benchmark neutron field. Fission spectrum neutrons are available for this purpose, and for many applications, provide an appropriate energy spectrum **(12, 13, 14, 15)**. 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.

9.13.3.1 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 **(15, 16, 17)**.

9.13.3.2 The form of Eq 7 for establishing a neutron fluence from individual threshold detector responses based on fission spectrum calibration is as follows:

$$\Phi_s = \frac{R_s}{R_\chi} \times \frac{\bar{\sigma}_\chi}{\bar{\sigma}_s} \times \Phi_\chi \quad (38)$$

where:

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

R_s = 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_s) 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 9.13.1, 9.13.2, 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.

9.13.3.3 Spectrum-averaged cross sections calculated for the ^{235}U and ^{252}Cf benchmark fission standard fields using the SNLRML recommended cross sections **(25)** are given in Table 3. The 5 %, 50 %, and 95 % response ranges are also indicated in the table.

9.13.3.4 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.

9.13.3.5 In general, 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.

9.14 *Fission Threshold Detectors*:

9.14.1 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 effectiveness of neutrons in producing damage does not increase appreciably above a few MeV **(18, 19)**.

9.14.2 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 7 becomes:

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

[A] Cross section and covariance matrices are consistent with the sources detailed in Ref (**25**).
[B] Spectrum taken from Ref (**31**), MT = 9861, MF = 5, MT = 8. Uncertainty in the spectrum is taken from Ref (**31**), MT = 9861, MF = 35, MT = 18.
[C] Spectrum taken from Ref (**31**), MT = 9228, MF = 5, MT = 18. Uncertainty in the spectrum is taken from Ref (**32**).
[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 (**33**), (**34**), (**35**), and (**36**). 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.

$$\phi(E > E_i) = R_s/Y\bar{\sigma}_j(E > E_i) \quad (39)$$

Tabulations are available (see Test Methods E 704 and E 705) for the fission yields of various fission products for several fissionable nuclides and for thermal and fast neutrons.

9.14.3 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 E 704 and E 705), (*2*) an aluminum catcher technique in which ^{237}Np fission recoils are deposited on a thin aluminum catcher foil and then counted **(20, 21, 22)**, and (*3*) the counting of ^{140}Ba, ^{137}Cs or other fission products, following chemical separation from the fission foils (refer to Test Methods E 343 and E 393), and (*4*) radiation damage techniques such as Test Method E 854.

9.14.4 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 23) for the 2200 m/s fluence rate, ϕ_e(Eq 32) for the epithermal fluence rate, $\phi(E > E_i)$ (Eq 37) for the threshold detectors, and Φ_s (Eq 38) 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 **(37)**). 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 E 944 for general 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

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