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Standard Guide for Selection and Use of Neutron Sensors for Determining Neutron Spectra Employed in Radiation-Hardness Testing of Electronics[1]

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This standard has been approved for use by agencies of the U.S. Department of Defense.

1. Scope

1.1 This guide covers the selection and use of neutron-activation detector materials to be employed in neutron spectra adjustment techniques used for radiation-hardness testing of electronic semiconductor devices. Sensors are described that have been used at many radiation hardness-testing facilities, and comments are offered in table footnotes concerning the appropriateness of each reaction as judged by its cross-section accuracy, ease of use as a sensor, and by past successful application. This guide also discusses the fluence-uniformity, neutron self-shielding, and fluence-depression corrections that need to be considered in choosing the sensor thickness, the sensor covers, and the sensor locations. These considerations are relevant for the determination of neutron spectra from assemblies such as TRIGA- and Godiva-type reactors and from Californium irradiators. This guide may also be applicable to other broad energy distribution sources up to 20 MeV.

NOTE 1—For definitions on terminology used in this guide, see Terminology E170.

1.2 This guide also covers the measurement of the gamma-ray or beta-ray emission rates from the activation foils and other sensors as well as the calculation of the absolute specific activities of these foils. The principal measurement technique is high-resolution gamma-ray spectrometry. The activities are used in the determination of the energy-fluence spectrum of the neutron source. See Guide E721.

1.3 Details of measurement and analysis are covered as follows:

1.3.1 Corrections involved in measuring the sensor activities include those for finite sensor size and thickness in the calibration of the gamma-ray detector, for pulse-height analyzer deadtime and pulse-pileup losses, and for background radioactivity.

1.3.2 The primary method for detector calibration that uses secondary standard gamma-ray emitting sources is considered in this guide and in Test Methods E181. In addition, an alternative method in which the sensors are activated in the known spectrum of a benchmark neutron field is discussed in Guide E1018.

1.3.3 A data analysis method is presented which accounts for the following: detector efficiency; background subtraction; irradiation, waiting, and counting times; fission yields and gamma-ray branching ratios; and self-absorption of gamma rays and neutrons in the sensors.

1.4 The values stated in SI units are to be regarded as standard. No other units of measurement are included in this standard.

1.5 *This standard does not purport to address all of the safety concerns, if any, associated with its use. It is the responsibility of the user of this standard to establish appropriate safety and health practices and determine the applicability of regulatory limitations prior to use.*

2. Referenced Documents

2.1 General considerations of neutron-activation detectors discussed in Practice E261, Test Method E262, and Guides E721 and E844 are applicable to this guide. Background information for applying this guide are given in these and other relevant standards as follows:

2.2 *ASTM Standards:*[2]

E170 Terminology Relating to Radiation Measurements and Dosimetry

E181 Test Methods for Detector Calibration and Analysis of Radionuclides

E261 Practice for Determining Neutron Fluence, Fluence Rate, and Spectra by Radioactivation Techniques

[1] This guide is under the jurisdiction of ASTM Committee E10 on Nuclear Technology and Applicationsand is the direct responsibility of Subcommittee E10.07 on Radiation Dosimetry for Radiation Effects on Materials and Devices.

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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.

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
E393 Test Method for Measuring Reaction Rates by Analysis of Barium-140 From Fission Dosimeters
E496 Test Method for Measuring Neutron Fluence and Average Energy from $^{3}H(d,n)^{4}He$ Neutron Generators by Radioactivation Techniques
E704 Test Method for Measuring Reaction Rates by Radioactivation of Uranium-238
E705 Test Method for Measuring Reaction Rates by Radioactivation of Neptunium-237
E721 Guide for Determining Neutron Energy Spectra from Neutron Sensors for Radiation-Hardness Testing of Electronics
E844 Guide for Sensor Set Design and Irradiation for Reactor Surveillance, E 706 (IIC)
E944 Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance, E 706 (IIA)
E1018 Guide for Application of ASTM Evaluated Cross Section Data File, Matrix E706 (IIB)
E1297 Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Niobium

3. Significance and Use

3.1 Because of the wide variety of materials being used in neutron-activation measurements, this guide is presented with the objective of bringing improved uniformity to the specific field of interest here: hardness testing of electronics primarily in critical assembly reactor environments.

NOTE 2—Some of the techniques discussed are useful for 14-MeV dosimetry. See Test Method E496 for activation detector materials suitable for 14-MeV neutron effects testing.

NOTE 3—The materials recommended in this guide are suitable for ^{252}Cf or other weak source effects testing provided the fluence is sufficient to generate countable activities.

3.2 This guide is organized into two overlapping subjects; the criteria used for sensor selection, and the procedures used to ensure the proper determination of activities for determination of neutron spectra. See Terminology E170 and Test Methods E181. Determination of neutron spectra with activation sensor data is discussed in Guides E721 and E944.

4. Foil Sets

4.1 *Reactions Considered:*

4.1.1 Neutron-induced reactions appropriate for this guide are listed in Table 1. The table includes most of the reactions used in this field. Those not marked with an asterisk are recommended because of their demonstrated compatibility with other reactions used in spectrum adjustment determinations. This compatibility is primarily based on experience with the ENDF/B-VI.1 (**1, 2**)[3], and IRDFF n1.05 (**3**) cross-sections. These recommendations may change modestly as revisions are made in the ENDF/B and IRDF dosimetry cross sections. Other reactions may be useful in particular circumstances with appropriate care. It is important that the user take full account of both the footnotes attached to each reaction and the discussions in the body of the text about individual reactions when implementing the foil-activation technique.

4.1.2 The four paired columns under the labels fast burst (**13**) and "TRIGA (**14**) Type" list the energy ranges within which 95 % of the response occurs for these two representative spectra. These limits are just a guide because the response often varies widely within each range. The response limits for an idealized fission spectrum with no $1/E$ tail can be much different (shifted toward higher energy) for resonance reactions. For example, in a Watt fission spectrum the $^{197}Au(n,\gamma)^{198}Au$ has a 95 % response between 5.0×10^{-2} and 2.7 MeV. The recommended foil mass column gives values that are designed to minimize self-absorption, self-shielding, and other corrections, provided the foils are 1.27 cm in diameter. The $E_t \cong 0$ fission foils, ^{235}U and ^{239}Pu, have similar cross-section shapes. However, the ^{235}U foil is preferred since it is less expensive and is much less of a health hazard than ^{239}Pu. In addition, when measuring soft (TRIGA) spectra, the ^{235}U foil is useful in determining the correction for the ^{235}U impurity in the ^{238}U foil (which is readily available with about 400 ppm or less ^{235}U impurity).

4.1.3 Although sulfur is listed and is used widely as a monitor foil, it is the only recommended sensor requiring beta particle detection and, therefore, requires a different calibration and counting technique. The $^{58}Ni(n,p)^{58}Co$ reaction has about the same threshold energy and, therefore, can be used instead of the $^{32}S(n,p)^{32}P$ if it acquires sufficient activity. Many facilities use sulfur as a routine monitor because its two-week half-life allows a convenient period for counting and permits reuse of the sensor after 6 to 9 months. Automated beta counters are commercially available. Neither nickel nor sulfur should be counted for the (n,p) reaction products immediately after irradiation because for nickel the ^{58}Co must build up through a metastable state, and for sulfur there are competing reactions. According to Test Method E264 the waiting period for ^{58}Co should be 4 days. For ^{32}P, Test Method E265 recommends waiting 24 h. Corrections can be made for shorter waiting periods.

4.1.4 In selecting dosimetry reactions one should consider the validation of the cross sections and associated uncertainty as demonstrated in the ^{235}U thermal fission and the ^{252}Cf spontaneous fission benchmark neutron fields. Ref (**15**) provides a comparison of the measured and calculated spectrum-averaged cross sections for these benchmark fields.

4.1.5 Some frequently used reactions have shown relatively consistent deviations of measured to calculated activity ratios in many different spectra determinations. For example, when ENDF/B-V cross sections are used in the reaction $^{63}Cu(n,\gamma)^{64}Cu$, the calculated activity is usually low, and an adjustment code will try to raise the spectrum in the vicinity of Cu

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

TABLE 1 Activation Foils

Reaction	Fast Burst[A]		TRIGA Type[A]		Eγ[B], (keV)	Gamma Emission Probability[B]	Fast Fission Yield,[C] %	$T_{1/2}$[B]	Recommended Foil Mass, g[D]	Footnotes
	E_L, MeV	E_H, MeV	E_L, MeV	E_H, MeV						
$^{197}Au(n,\gamma)^{198}Au$	4.00 – 6	7.20 – 4	3.80 – 6	9.20 – 6	411.80205	95.62		2.6943 days	0.06	E,F,G
$^{59}Co(n,\gamma)^{60}Co$	7.60 – 6	4.50 – 4	6.90 – 7	1.43 – 4	1173.228	99.85		5.2711 years	0.06	E,G
					1332.492	99.9826				
$^{*58}Fe(n,\gamma)^{59}Fe$	1.00 – 6	2.10 + 0	5.25 – 7	1.00 – 2	1099.245	56.51		44.494 days	0.15	E,H
					1291.590	43.23				
$^{55}Mn(n,\gamma)^{56}Mn$	5.25 – 7	6.60 – 1	4.75 – 7	1.10 – 3	846.7638	98.85		2.57878 h	0.05	E,F
					1810.726	26.9				
$^{*63}Cu(n,\gamma)^{64}Cu$	1.15 – 6	2.30 + 0	5.25 – 7	9.60 – 3	1345.77	0.4748		12.7004 h	0.15	E
$^{23}Na(n,\gamma)^{24}Na$	6.30 – 7	2.00 + 0	5.25 – 7	3.00 – 3	1368.630	99.9934		14.4958 h	0.10	E,I,J
					2754.049	99.862				
$^{45}Sc(n,\gamma)^{46}Sc$	4.25 – 7	1.00 + 0	4.00 – 7	4.75 – 4	889.271	99.98374		83.787 days	0.05	E
					1120.537	99.97				
$^{235}U(n,f)^{140}La$	9.20 – 2	4.70 + 0	6.30 – 4	3.80 + 0	1596.203	95.40	5.9599	1.67858 days	0.30	E,K,L
$^{235}U(n,f)^{95}Zr$	9.20 – 2	4.70 + 0	6.30 – 4	3.80 + 0	724.192	44.27	6.3488	64.032 days	0.60	E,L
					756.725	54.438				
$^{239}Pu(n,f)^{140}La$	1.43 – 1	4.80 + 0	8.80 – 4	4.30 + 0	1596.203	95.40	5.3244	1.67858 days	1.00	E,K,L
$^{239}Pu(n,f)^{95}Zr$	1.43 – 1	4.80 + 0	8.80 – 4	4.30 + 0	724.192	44.27	4.6825	64.032 days	0.60	E,L
					756.725	54.38				
$^{93}Nb(n,n')^{93m}Nb$	8.40 – 1	5.70 + 0	1.00 + 0	5.50 + 0	30.77	0.000591		16.12 years		M
$^{103}Rh(n,n')^{103m}Rh$	5.50 – 1	5.70 + 0	6.90 – 1	5.70 + 0	39.755	0.068		56.114 min		M
$^{237}Np(n,f)^{140}La$	5.75 – 1	5.60 + 0	6.60 – 1	5.50 + 0	1596.203	95.40	5.74440	1.67858 days	0.60	E,K,L,N
$^{237}Np(n,f)^{95}Zr$	5.75 – 1	5.60 + 0	6.60 – 1	5.50 + 0	724.192	44.27	5.61470	64.032 days	0.60	E,L
					756.725	54.38				
$^{*115}In(n,n')^{115m}In$	1.00 + 0	6.00 + 0	1.20 + 0	5.80 + 0	336.241	45.8		4.486 h	0.12	
$^{238}U(n,f)^{140}La$	1.50 + 0	6.90 + 0	1.50 + 0	6.60 + 0	1596.203	95.40	5.9718	40.28 h	1.00	E,K,L,O
$^{238}U(n,f)^{95}Zr$	1.50 + 0	6.90 + 0	1.50 + 0	6.60 + 0	724.192	44.27	5.1883	64.032 days	1.00	E,L
					756.725	54.38				
$^{232}Th(n,f)^{140}Ba$	1.50 + 0	7.40 + 0	1.50 + 0	7.10 + 0	537.303	24.39	7.7121	12.753 days	1.00	E,K,P
$^{232}Th(n,f)^{95}Zr$	1.50 + 0	7.40 + 0	1.50 + 0	7.10 + 0	724.192	44.27	5.5230	64.032 days	1.00	E,L
					756.725	54.38				
$^{54}Fe(n,p)^{54}Mn$	2.30 + 0	7.70 + 0	2.30 + 0	7.40 + 0	834.848	99.9752		312.19 days	0.15	E
$^{58}Ni(n,p)^{58}Co$	2.00 + 0	7.60 + 0	2.00 + 0	7.30 + 0	810.7602	99.44		70.85 days	0.30	E
$^{47}Ti(n,p)^{47}Sc$	1.90 + 0	7.60 + 0	1.90 + 0	7.30 + 0	159.373	68.1		3.3485 days	0.15	E,Q,R
$^{32}S(n,p)^{32}P$	2.40 + 0	7.50 + 0	2.30 + 0	7.30 + 0	1710.66	100. (beta)		14.284 days	...	S
$^{64}Zn(n,p)^{64}Cu$	2.60 + 0	7.70 + 0	2.60 + 0	7.40 + 0	1345.77	0.4748		12.7004 h	0.30	E
$^{27}Al(n,p)^{27}Mg$	3.50 + 0	9.40 + 0	3.40 + 0	9.20 + 0	843.76	71.800		9.458 min	0.30	E
					1014.4	28.0				
$^{46}Ti(n,p)^{46}Sc$	3.80 + 0	9.60 + 0	3.70 + 0	9.20 + 0	889.3	99.983		83.787 days	0.15	E,Q
					1120.5	99.986				
$^{56}Fe(n,p)^{56}Mn$	5.50 + 0	1.14 + 1	5.50 + 0	1.10 + 1	846.7	98.85		2.57878 h	0.15	E,T
					1810.7	26.9				
$^{24}Mg(n,p)^{24}Na$	6.50 + 0	1.17 + 1	6.50 + 0	1.13 + 1	1368.6	99.993		14.958 h	0.03	E,J
					2754.1	99.872				
$^{27}Al(n,\alpha)^{24}Na$	6.50 + 0	1.21 + 1	6.50 + 0	1.17 + 1	1368.6	99.993		14.958 h	0.30	E,J
					2754.1	99.872				
$^{48}Ti(n,p)^{48}Sc$	5.90 + 0	1.24 + 1	5.90 + 0	1.20 + 1	983.5	100.1		43.67 h	0.15	E
					1037.5	97.56				
					1312.1	100.1				
$^{93}Nb(n,2n)^{92m}Nb$	9.70 + 0	1.45 + 1	9.40 + 0	1.40 + 1	934.4	99.1		10.15 days		
$^{127}I(n,2n)^{126}I$	9.70 + 0	1.47 + 1	9.70 + 0	1.43 + 1	388.633	35.6		12.93 days	0.25	E
					666.331	32.9				
$^{65}Cu(n,2n)^{64}Cu$	1.08 + 1	1.57 + 1	1.07 + 1	1.53 + 1	1345.7	0.475		12.7004 h	0.15	E,M
$^{*63}Cu(n,2n)^{62}Cu$	1.19 + 1	1.66 + 1	1.19 + 1	1.63 + 1	875.7	0.150		9.67 min	0.15	E,H
$^{90}Zr(n,2n)^{89}Zr$	1.28 + 1	1.69 + 1	1.27 + 1	1.67 + 1	909.1	99.0		78.42 h	0.10	
$^{58}Ni(n,2n)^{57}Ni$	1.32 + 1	1.71 + 1	1.31 + 1	1.69 + 1	1377.6	81.2		35.9 h	0.30	

[A] Energy limits which describe the 5 – 95 % region of the detector response occurs for each reaction (see Practice E261 and Refs **(4, 5)**. The foils are assumed to have Cd covers as described in Footnote *E*.

[B] Data taken from Refs **(6-8)**. Ref **(8)** takes precedent, but it only addresses reactions used in detector calibration. In other cases, Ref **(6)** provides the half-life and Ref **(7)** provides the gamma yields. Many gamma-ray energies rounded to the nearest 0.1 keV. For uncertainties on values, see references. When the emission process is beta decay, the quoted energy is the maximum beta energy.

[C] Fission yields can be found in Ref **(9)**.

[D] Choice of mass is based on assumed foil diameter of 1.27 cm.

[E] Cd covers 0.5 to 1-mm thicknesses. Pairs of bare and Cd-covered foils are advantageous for resonance reactions.

[F] Use ^{59}Co instead of ^{197}Au and ^{55}Mn for very long irradiations.

[G] Use dilute aluminum-gold alloy (<0.2 % Au) when possible.

[H] Do not count the 0.511 line.

[I] Use in the form of NaCl.

[J] The 1986 edition of Ref **(10)** has a typographical error for the half-life of ^{24}Na. The correct number can be found in previous editions. The correct number can also be found in Ref **(6)**.

[K] This is the 1.67858 days daughter of 12.753-day ^{140}Ba. Wait 5 days for maximum decay rate (see Test Method E393).

[L] E_t = 0.01 MeV shielded with ^{10}B sphere. (Use of ^{10}B shield is important for soft (TRIGA) spectra where $\Phi(E < 0.01$ MeV) will otherwise dominate).

[M] Precautions must be taken in counting because of the low gamma-ray energy. See Test Method E1297 for details of ^{93m}Nb use. For ^{103m}Rh, X-rays are typically counted rather than listed gamma ray. See Ref **(7)**.

[N] If a ^{10}B sphere is used for the ^{239}Pu foil, then a ^{10}B sphere should also be used for the ^{237}Np foil so that correction for ^{239}Pu impurity in the ^{237}Np foil can be made.
[O] If a ^{10}B sphere is used for the ^{235}U foil, then a ^{10}B sphere should also be used for the ^{238}U foil so that correction for ^{235}U impurity in the ^{238}U foil can be made.
[P] Radioactivity of ^{232}Th interferes with the ^{140}La line.
[Q] At high energies (>10 MeV), account for (*n,np*) contributions from higher atomic number *Ti* isotopes.
[R] See Refs **(11)** and **(12)**.
[S] Requires β counting techniques, see Test Method E265.
[T] Maximum Mn impurity = 0.001 %, Cd covered. Do not use ^{56}Fe foil for long irradiations.
* Not recommended for use at this time either because of large uncertainties or because of conflicts with other reactions during spectrum adjustment procedures.

resonances. In fact, however, this consistent behavior indicates that the tabulated cross-section values in some important energy region are too small. The analyst must then choose one of the following alternatives: (*1*) leave out reactions which have demonstrated consistent deviations; (*2*) seek better cross-section sets; (*3*) assign wide error bars or low statistical weight to these reactions. It is recommended that the first option be chosen because a sufficient number of well-established cross sections do exist to satisfactorily determine fast reactor spectra. Furthermore, if the cross section for a particular reaction is not well established, and it is assigned too large a weight in the spectrum adjustment procedure, the final spectrum can be severely distorted. Other suspect reactions are noted in Table 1 with an asterisk.

NOTE 4—Some of the reactions not recommended at this time (on the basis of inconsistencies among recommended cross sections) may be upgraded when more recent evaluations are applied to a wide range of neutron spectra.

4.2 *Foil Impurities:*

4.2.1 Foil impurities are especially serious for a moderated source (TRIGA reactor) when an impurity leads to the same reaction product by way of thermal-neutron capture. Some examples of these foils, with impurities in parentheses, are ^{238}U (^{235}U), ^{27}Al (^{23}Na), ^{56}Fe (^{55}Mn), and ^{24}Mn (^{23}Na).

4.2.2 For a soft spectrum, such as the TRIGA J-tube spectrum [boral (boron-aluminum alloy) shielded], the number of fissions in the ^{235}U foil (Cd covered) is about 100 times the number occurring in the ^{238}U foil; therefore, the ^{238}U must have an impurity level of ^{235}U of no more than about 200 ppm for an uncertainty of 2 % or less in determining accurately the ^{238}U activity. Higher impurity levels of ^{235}U can be tolerated for Godiva-type reactors where the fluence below 10 keV is much lower, or with TRIGA-type reactors if the ^{235}U foil data are used for correcting the ^{238}U activity.

4.2.3 When the ^{56}Fe foil (Cd covered) is used in a TRIGA spectrum, it should have no more than 10 ppm ^{55}Mn impurity to keep the contribution from the $^{55}Mn(n,\gamma)^{56}Mn$ reaction to less than 2 %. Similarly, the ^{55}Mn impurity should be no more than 100 ppm when using the ^{56}Fe foil at 50 cm from a Godiva-type reactor (which is approximately 2 m above the concrete floor) in order to achieve the same level of accuracy. Data from a ^{55}Mn foil (Cd covered) can be used to correct the ^{56}Fe data if the impurity correction is ≤20 % of the total (*n,p*) activation, and the percent of manganese in iron is accurately known.

4.3 *Influence of Nuclear Data on Foil Selection:*

4.3.1 Since the total number of interactions is deduced from an absolute specific activity determination, that activity should be determined with good accuracy (of the order of 5 %), and the foils selected should have gamma-ray yields known to the same or better accuracy. Some of the factors involved in determining these yields include conversion-electron production, branching ratio to a given energy level, and fission yield.

4.3.2 The 1596.203-keV gamma-ray transition from ^{140}La produced by ^{232}Th fission is not usually useful because of interference from ^{232}Th radioactivity. This often has led to the use of the 537.303-keV transition from the ^{140}Ba precursor of ^{140}La, having a gamma-transition probability of 0.2439 per ^{140}Ba decay. The use of ^{140}Ba generally requires the chemical separation of this isotope from the rest of the fission products so that the 537.303-keV line can be seen above competing lines. See Test Method E393.

4.3.3 The choice of element, and hence the gamma-ray transition, directly influences the accuracy of determining the specific activity induced by neutron irradiation. It also influences the final choice of foil thickness, in that the selection of an element resulting in a low-energy gamma ray may lead to a large self-absorption correction. For example, the ^{232}Th foil of Table 1 has a maximum attenuation of 22 %, or an average correction of about 11 %, for the 537.303-keV transition. This represents an upper limit for the thickness of that foil. Therefore, the self-attenuation of gamma rays, as well as the neutron self-shielding discussed later, will influence the foil selection.

NOTE 5—For other considerations in the selection of specific foils, see Guide E844, Practice E261, and Test Methods E262, E263, E264, E265, E266, E704, and E705.

5. Apparatus

5.1 The gamma-ray detector should be a germanium-type detector (either Ge(Li) or intrinsic) with an energy resolution of 2.5 keV or better (full-width at half-maximum (FWHM) at 1173 keV). Associated equipment would include a multichannel pulse-height analyzer and a precision pulse generator with calibrated pulse-height and pulse-rate inputs into the detection system.

5.2 Foil and source holders should be used to provide precise positioning of a gamma-ray standard source and of each activated foil with respect to the detector. Required precision is about 0.2 mm or better in distance from the window of the detector or in lateral alignment.

5.3 National standard sources that are traceable to NIST (or their equivalent) should be used for calibration of the detection system.

6. Precautions

6.1 *Scattering Problems*—A sensor with a strong resonance absorption, such as a thick ^{235}U foil, should not be placed in front of a 1/v detector, and thick foils with covers should not be

stacked because accurate corrections for the resultant scattering are difficult to determine. With an isotropic neutron-fluence, Φ_o, incident on stacked foils, the reduction in the fluence rate caused by scattering at a given foil can be estimated by using the following equation:

$$\Phi = \Phi_o e^{-\Sigma_i \sigma_i X_i}$$

where Φ is the attenuated fluence, $\sum_i$ is a summation-over-i symbol, σ_i is the total macroscopic scattering cross section in cm^{-1}, and X_i is the thickness of the i^{th} foil in centimetres. The summation is up to the foil of interest, located at its appropriate depth (distance from source) in the foil stack. For best results, the reduction in fluence rate should be less than 10 % for the foil located at the maximum depth.

6.2 *Foil Self-Shielding*—For the thicknesses of the foils recommended, the correction for self-shielding is recommended for all (n,γ) and (n,f) reactions. A pure gold foil is an example of a self shielding foil with its highly absorbing resonance at about 5 eV. The correction for a 0.025-mm thick foil being about a factor of two for epicadmium neutrons (that is, neutrons with energies greater than 0.5 eV) (**16**).

NOTE 6—Dilute aluminum-gold alloys are available and do not generally require self-shielding corrections.

6.3 *Fluence Nonuniformity*—If all the foils cannot be located in a region of uniform fluence rate (as determined by symmetry considerations), they can be located at different positions (and, hence, with different fluence rates) as long as the neutron energy spectrum is constant. If the fluence varies by more than 3 % from point to point, fluence monitors should be used with each foil. Around a Godiva-type reactor, sulfur foils can serve as monitors near the individual foils. Where space is more limited, then nickel [^{58}Ni(n,p)^{58}Co], iron [^{54}Fe(n,p)^{54}Mn], or even aluminum [^{27}Al(n,α)^{24}Na] should be considered for monitors. (See Practice E261 for other relevant considerations.) Often a better solution is obtained by mounting all foils on a rotating disk or ring to ensure that they receive the same fluence.

6.4 *Fluence Rate Depression*—At low energies, fluence rate depression can be significant for bare thermal-neutron detectors near cadmium-covered foils if both are embedded in a moderator. This is because the cover on one foil can shadow adjacent bare foils. At high energies, depression can be significant for foils irradiated under the same conditions if the moderator contains reactor fuel. However, this is ordinarily not a problem, since in the sizable irradiation volumes normally used for radiation damage studies, the cadmium covers (as well as the foils) generally subtend a negligibly small solid angle at the point of any surrounding moderator or fuel. Fluence rate depression is usually insignificant for irradiation in a Godiva reactor glory hole.

7. General Handling Procedures

7.1 *Foil Encapsulation*—Fission foils should be encapsulated in sealed containers to avoid oxidation, loss of material, and for health-safety requirements. If a ^{239}Pu foil is used (instead of the much safer ^{235}U foil), it will require special encapsulation and periodic monitoring to check for leakage of the material. Copper encapsulation has been found satisfactory for ^{235}U, ^{238}U, ^{237}Np, and ^{232}Th foils. The thickness of the copper capsule should be about 0.1 to 0.25 mm at the flat surfaces and soldered at the periphery.

7.2 *Foil Covers:*

7.2.1 As noted in Table 1, cadmium covers of 0.5 to 1-mm thickness are prescribed for all fission foils and 1/v detectors. Cadmium covers also should be used for finite-threshold foils with trace impurities that yield the same reaction product by means of thermal-neutron capture. Examples are foils such as ^{238}U, ^{56}Fe, ^{58}Ni, and ^{27}Al with impurities of ^{235}U, ^{55}Mn, ^{59}Co, and ^{23}Na, respectively. Depending on the concentration, such impurities can lead to large correction factors. For large correction factors (that is, greater than 5 %), cadmium-covered foils made of the impurity materials should be irradiated. Then, corrections can be made with good accuracy if the impurity concentration in the primary threshold foil is accurately known. If the impurity concentration is not known, a thermal-neutron activation analysis of the foil can provide data for the necessary correction. Cadmium covers may not be required for foils irradiated in the empty "glory hole" of a fast-pulse reactor, a cavity in which little or no moderator material is normally present (that is, less than 0.5 g/cm^2).

7.2.2 Covers of ^{10}B for fission foils are useful when measuring a soft TRIGA spectrum. However, if a boral shield that provides good 4-π geometry surrounds the irradiation cavity, and if a negligible amount of moderator is contained within the shield, then the ^{10}B covers may not be required. The effect of the boral shielding should be accounted for properly when the neutron spectrum is adjusted with a proper computer code. More is said about boron cover corrections in 7.2.4.

NOTE 7—Spectra adjustment codes are discussed in Guides E721 and E944.

7.2.3 If no ^{10}B covers are used for the foils, and if the TRIGA irradiation cavity is only partially shielded by boral, then it will be difficult to determine the neutron spectrum from 10^{-2} MeV down to about 3×10^{-7} MeV. If the TRIGA irradiation cavity has only partial boral shielding, it is important that all the fission foils, all the 1/v foils, and the foils with important 1/v impurities be placed in a boral "box" or a ^{10}B cover. For best results, a ^{10}B cover of 1 to 1.8 g/cm^2 of (93 %) ^{10}B should be used. In this way, the fraction of activations arising from neutrons in the energy range from 3×10^{-7} MeV to 10^{-2} MeV will be reduced greatly. The effect of the cover thickness can be accounted for by a spectrum adjustment code provided that the effective attenuation cross section that accounts for scattering in the cover is available. See 7.2.5.

7.2.4 For a Godiva-type reactor, ^{10}B covers may not be required, and cadmium covers may be sufficient for irradiation distances of less than 1 m from the reactor when the reactor is located a few metres above the concrete floor. Cadmium covers also may be used in the glory hole where the number of low-energy neutrons is insignificant. If ^{10}B covers are used, activities may require correction for scattering by the ^{10}B. The correction can be determined either experimentally with pure finite-threshold fission foils (^{237}Np or ^{232}Th) that contain

negligible zero-threshold impurities, or with a neutron transport calculation that takes into account the thickness of the material (**17**).

7.2.5 The attenuation by a boron cover of the neutron fluence is not adequately treated by many of the spectrum adjustment codes (**18**). Some versions of the spectrum adjustment code, SAND II (**19**), for example, use a simple exponential attenuation function versus energy, and because most irradiations are conducted in wide-beam or isotropic configurations, scattered neutrons are not in general lost from the beam. As a result, the absorption cross section of the boron should generally be used to determine the attenuation. However, in many configurations (such as narrow-beam geometry or down scattering of the neutrons to lower energy) the scattering portion of the cross section can remove additional neutrons and the true effective removal cross section value will fall somewhere in between the total and the absorption cross section. This is especially noticeable if the response of the foil is concentrated above the 10-keV limit where the ^{10}B absorption ceases to dominate the cross section. Thus, for high-threshold fission foils such as ^{238}U and ^{237}Np or a normal threshold foil such as nickel, the additional scattering will result in additional attenuation. For example, some experiments and calculations indicate that these corrections are of the order of 10 % for a 1.65-g/cm^2 ^{10}B cover and a thin 12.7-mm diameter fission foil (**20**). Other work indicates that these scattering corrections may be somewhat larger (**21**). Strictly speaking, a calculation of the transport in the full-experiment geometry through the boron cover should be performed for each geometry(**18**). Measurements with a high-threshold foil, ^{58}Ni(n,p)^{58}Co, have shown a transmission factor of 0.9 in a Godiva-type exposure geometry (**15**). This compares with a calculated value (for which only the boron capture cross section is used) of 0.96.

NOTE 8—A monitor foil such as nickel used both inside and outside a boron ball can be used to normalize the boron-covered-fission-foil exposure to that of the rest of the foil set in case positioning errors are likely to be significant. The nickel ratio is not very sensitive to spectrum shape. The procedure is to multiply the fission foil activities by a factor that ensures that the ratio of nickel activities inside and outside the boron ball is about 0.9.

7.2.6 Another advantage of using covers (B, Cd) on broad energy-response foils is that it restricts that response and permits improved definition of the spectrum during the adjustment process. If both bare and Cd-covered resonance materials (such as Au and Na) are exposed, much better definition of the shape of the spectrum in the epithermal and thermal region can be obtained.

NOTE 9—Some versions of spectrum adjustment codes handle covers through the use of auxiliary codes that apply an energy-dependent-cover correction factor to the dosimetry cross section.

7.2.7 If the spectrum is to be well defined, then the foil set must contain a large fraction of the reactions from Table 1 and possess response functions spread as uniformly over energy as is possible. This is necessary to ensure that the spectrum adjustment codes can arrive at sufficiently restricted solutions. With broad response functions the calculated fluence at one energy can influence the calculated spectrum values at distant energies. If at all possible include ^{237}Np, and ^{239}Pu or ^{235}U to provide sensitivity between 10 keV and 1 MeV where few other reactions have significant response. Silicon devices are also sensitive in this energy region and can be used as spectrum sensors (**22**).

8. Certification of Foil Purity

8.1 The foil purity analysis results should be recorded permanently so that appropriate impurity corrections can be made. The acceptable uncertainty in the results mainly dictates what impurity concentrations are acceptable. It also depends on the nature and source of the neutron spectrum being measured (see 4.2). If, for example, the percentage impurity of ^{235}U in a foil of ^{238}U is known to be 400 ppm to an accuracy of 10 %, a separate ^{235}U foil can be irradiated in the same way as the primary foil to determine a proper correction factor. In this case, the impurity effect can be reduced to 10 % of its stated value (40 ppm) ^{235}U in ^{238}U by applying the correction factor. In determining the activity of a ^{238}U foil irradiated with a TRIGA spectrum to an uncertainty of 2 % or less, up to 2000 ppm of ^{235}U impurity could be tolerated (see 4.2).

9. Determination of Activities

9.1 A suitable set of sensors is placed in the neutron field under study. After irradiation, the specific activities of the sensor are determined by counting the gamma-ray emissions from each foil and applying appropriate corrections.

NOTE 10—Other energy response functions appropriate for spectrum adjustment procedures measured by detection of other effects, such as emulsion tracks or even displacement damage, can also be used successfully. See Guide E944, Section 4.1.

9.2 The measured specific activities of the activation foils are related to the incident neutron energy-fluence spectrum by the following equation:

$$R_j = \int_0^\infty \sigma_j(E)\Phi(E)\,dE \qquad 1 \le j \le n \tag{1}$$

where:

R_j = measured specific activity of an activated foil isotope j,
$\sigma_j(E)$ = neutron cross section at energy E for isotope j,
$\Phi(E)$ = incident neutron fluence differential in energy, and
n = number of reactions.

9.3 The differential neutron energy-fluence spectrum $\Phi(E)$ is calculated by means of a computer code that utilizes the specific activity data from the activation foil set. A number of these codes have been developed for this purpose and are available from the Oak Ridge National Laboratory Radiation Safety Information Computation Center (**23**).

10. Detector Calibration Procedures

10.1 Follow the general considerations in General Methods E181 and Test Method E265 on energy and efficiency calibration of the detector.

10.2 The germanium detector is usually operated at low temperatures (near the boiling point of liquid nitrogen). This requires the detector to be in a cryostat under vacuum. Normally, a thin window separates the detector's face from the

outside environment. In such an enclosure, the exact position of the effective center of the active volume of the detector with respect to the cryostat window may not be known precisely.

10.3 Very low-activity foils must be placed close to the detector window in order to achieve a reasonable count rate. For such close foil-detector spacing, two problems occur that can affect the detector efficiency. One concerns the effects of finite source size on the effective detector solid angle, and the other concerns coincidence photon summing. Coincidence summing occurs when a radionuclide emits two or more cascade photons within the resolving time of the detector system. These problems are considered in the following sections that deal with determining detector efficiency.

10.3.1 Measure the count rate under each energy peak from a small diameter (about 2 mm) standard source at some specified distance, d (100 mm or greater), from the detector window. Use a long-lived mixed radionuclide standard source or several single radionuclide standard sources (or their equivalents) for these measurements (see Note 11). Determine the detector efficiency, $\varepsilon(d)$, at this distance, d, from the source. The detector efficiency is defined as the ratio of the net count rate under the selected energy peak to the known gamma-ray emission rate of the standard source at that energy. A log-log plot of these data provides an efficiency-versus-energy curve for later use in estimating the efficiency for foils of larger diameter than the point calibration source.

NOTE 11—An example of a mixed radionuclide standard source suitable for this purpose is NBS SRM 4275B.[4] NBS is now NIST. An alternative method for calculating summing corrections is found in the documentation for this source. While this technique does not require the counting of foil materials in two locations, as discussed below, it does require that the detector's total efficiency curve be known. Experience has shown that a relatively crude knowledge of the total efficiency curve is sufficient to calculate summing corrections within a few percent for the foils in Table 1 except for ^{48}Ti.

10.3.2 To determine the detector efficiency for activated foil, j, select one of the higher-activity single-energy-transition foils, (for example, ^{197}Au(n,γ)^{198}Au with a 412-keV gamma ray), and measure the peak count rates at a position, c, close to the detector window and at the distant position, d. From the definition of detector efficiency, it can be seen for Foil j that the ratio of peak count rates is equal to the ratio of efficiencies at the respective positions as follows:

$$\frac{\dot{N}_p(c)_j}{\dot{N}_p(d)_j} = \frac{\varepsilon(c)_j}{\varepsilon(d)_j} \tag{2}$$

where $\dot{N}_p$ is the net count rate under the selected energy peak. It is important to note that the count rate $\dot{N}_p$ is actually defined as the average count rate during the count period,

$$\dot{N}_p = N_p/t_i,$$

where t_i is the count period.

10.3.3 Assume that the efficiency at position d is approximately the same for both the selected foil and the standard source. Then the efficiency for foil j at the position c is expressed as follows:

$$\varepsilon(c)_j = \frac{\dot{N}_p(c)_j}{\dot{N}_p(d)_j}\,\varepsilon\,(d)_s \tag{3}$$

where $\varepsilon(d)_s$ is obtained from the calibration curve determined in 10.3.1.

10.3.4 Repeat the procedures of 10.3.2 and 10.3.3 for several other high-activity single-energy-transition foils with gamma-ray energies covering the energy range of interest. Use these data to determine an efficiency-versus-energy curve for a foil at the distance of c from the detector.

10.3.5 The procedures given in 10.3.1 – 10.3.3 are valid for either single-energy transitions or cascade transitions (two or more photons in coincidence). However, the efficiency-versus-energy curve determined in 10.3.4 for single-energy-transition foils is not applicable to cascade-transition foils because of the coincidence-summing effect. The efficiency for each cascade-transition radionuclide should be determined individually in order to avoid the uncertainties and efforts associated with calculating the summing corrections. The difference between the efficiency of a cascade-transition nuclide and the single-energy-transition efficiency curve at the same photon energy can be large for such cascade-transition nuclides as ^{60}Co, ^{56}Mn, ^{140}La, and ^{24}Na when these sources are counted very close to the face of the detector. For example, a difference of 27 % has been observed for the 1596-keV gamma ray from ^{140}La when counting this source near the face of a 65-cm^3 Ge(Li) detector (**24**).

10.3.6 Coincidence-summing effects are only important for very close detector-source geometries. Thus, the calibration efficiency curve determined with the standard source in 10.3.1 is valid even if there are cascade transitions from the standard source since the source-to-detector distance is 100 mm or greater. The additional uncertainty from summing for measurements with the source at 100 mm from the window of a 65-cm^3 Ge(Li) detector is 1 % or less (**24**).

10.3.7 In order to get reasonable counting rates at both the close and distant locations relative to the detector, some planning of the counting procedures may be required when determining the detector efficiency for the various foils. At the distant location (100 mm or greater from the detector face), the count rate should be high enough to achieve good counting statistics in a reasonable time period. At the close location, the count rate should be low enough so that large and complex corrections for random summing and pulse pileup are avoided (see 12.1). One possible method for meeting these conflicting requirements is to use isotopes with reasonably short half-lives so that the foil can be counted first at the distant location and then a few days later at the close location when it has decayed to an appropriate level.

10.3.8 As an example of the positioning reproducibility required when foils are counted close to the detector window, suppose that the effective center of a particular detector is 21 mm from the front face of the cryostat window. Then, the location of the center of the source (either the standard or the activation foil) relative to the cryostat face must be reproduced within 0.2 mm for the uncertainty in the distance to the detector center to be within 2 %.

[4] NIST SRM 4275B available from Office of Standard Reference Materials. National Institute of Standards and Technology, Gaithersburg, MD 20899.

10.4 Another detector calibration procedure may be used for activation foils with half-lives longer than about 1 day. It utilizes fission neutron sources in irradiation facilities (for example, at the National Institute of Standards and Technology) where direct free-field neutron calibrations (**18, 25**) are available. Such sources provide certified fluences of up to 10^{13} n/cm^2 in low-background environments.

10.4.1 The procedure involves irradiating a set of activation foils in calibrated neutron fields and then transporting the foils to the user's detector apparatus for counting. If the neutron fluence rate under investigation is similar to a fission spectrum, or if the detector response is energy independent over the energy range of interest, a direct neutron fluence transfer technique can be made. For details of this method, see 4.8.3 of Practice E261. The neutron fluence transfer technique relaxes the requirement to establish absolute detector efficiencies, and the uncertainties associated with absolute cross sections are significantly reduced because only ratios of the spectrum-averaged cross sections are required. It is important, however, for the spectrum-averaged cross section of the calibration source and of the reactor spectrum under consideration to be calculated from the same cross-section compilation.

10.5 If the activated foil has a decay scheme containing a significant number of low-energy gamma rays as well as the gamma ray of interest, insert a lead shield of an appropriate thickness between the foil and the detector. Choose a shield thickness that will significantly attenuate the low-energy gamma rays and avoid pulse pileup in the detector, but still allow a reasonable count rate for the desired gamma ray. For example, a lead shield of about 13-mm thickness is appropriate for counting some fission foils (^{237}Np, ^{235}U, ^{239}Pu, and ^{238}U) yielding the same fission product of interest, ^{140}La. If such a lead shield is required, then perform the calibration procedures of 10.3 or 10.4 with the lead shield in place and determine the detector efficiency for the resulting foil-shield-detector geometry. (For more details on ^{238}U and ^{237}Np, see Test Methods E704 and E705, respectively.)

11. Counting Procedures

11.1 *Pulse-Height Analyzer Deadtime and Pulse Pileup:*

11.1.1 Use of a precision pulse generator is recommended for determining the correction for the combined effects of multichannel analyzer deadtime and pulse-pileup losses. Use the pulser dynamically; that is, pulses from it are counted at the same time that the activated foils or standard source are measured. Adjust the pulser output (pulse amplitude) to place the peak in a low-background area of the spectrum being analyzed. Also, use a low repetition rate (about 60 Hz).

11.1.2 With a foil in the counting position and the pulser on, run the analyzer on "clock" time (as opposed to "live" time). The ratio of the number of pulses generated during the counting period to the number of counts recorded in the pulser peak in the analyzer gives the correction factor for the combined deadtime and pulse-pileup losses (**26**).

11.2 *Background Corrections:*

11.2.1 Minimize laboratory background counts by selecting a low-background counting area, by moving all radioactive sources other than the foil being counted from the counting area, and by placing a gamma shield around the detector. If the gamma shield is lead, it should be at least 50 mm thick.

11.2.2 Carefully determine the backgrounds for fission-foils since all such foils have some residual natural radioactivity and, also, because such foils often are reused due to their initial high cost. If the foils have been previously irradiated within a period of less than several half-lives of the gamma ray of interest, then measure the background at least twice. Allow sufficient time to elapse between the measurements so that gamma rays with relatively short half-lives can be distinguished from any long-lived components due to either natural radioactivity or to other fission fragments. Corrections can then be made for both short- and long-lived background components.

11.3 *Counting Redundancy*—It is recommended that each foil be counted on each of two or more calibrated counting systems. If there is disagreement by more than 5 %, repeat the count (and calibration if necessary). If only one counter is available, at least remove and replace the foil between readings.

12. Data Analysis

12.1 *Peak-Area Analysis:*

12.1.1 Use a consistent method of peak-area analysis for peaks originating from the precision pulser, the calibration source, and the activated foils.

12.1.2 In one basic method, plot the counts per channel around the peak. Subtract the baseline area (background) from the peak area by fitting a straight line through the baseline.

12.1.3 In counting fission foils, examine carefully the primary peak for the presence of a very close neighboring peak. If close neighboring peaks are present, use a peak-shape analysis technique. This analysis can be done either by hand or with a suitable computer code (for example, the SAMPO code (**27**)). Other peak analysis codes associated with commercial counting systems are also available. Good counting statistics are necessary for the peak-shape analysis to give reasonably accurate results. Accumulate at least 10 000 counts in the net peak area whenever possible.

12.2 *Peak Area Corrections*—Since in neutron effects testing of electronic parts, the usual interest is in the permanent damage from the integral fluence, the activities discussed in this section are determined in terms of detector counts (in the manner of Test Method E265) rather than in terms of count rates. The derivations of the activities are given in the appendix. Correct the net peak areas determined in 12.1 for analyzer deadtime and pulse-pileup losses by multiplying by the correction factor discussed in 11.1.1. The efficiency of the counting system must be accounted for and correction made for self-absorption of the gamma rays by thick foils. Thus, the gamma-ray emission, N_γ, (number of photons emitted by the daughter isotopes caused by the fluence Φ in the foil) is given as follows:

$$N_{\gamma(tc)} = \frac{N_p(t_c)C_{pp}e^{[(\mu_a/\rho)(z/2)]}}{\varepsilon(c)} \quad (4)$$

where:

N_p = net counts under the peak after counting for time t_c,
$\varepsilon(c)$ = detector efficiency for the foil at position c (counts per disintegration),
C_{pp} = correction factor for analyzer deadtime and pulse pileup (ratio of pulses generated to counts in pulser peak),
μ_d/ρ = mass absorption coefficient, cm^2/g, and
z = combined thickness of the foil and any encapsulation material, g/cm^2.

The exponential self-absorption correction is an approximation; however, it is reasonably accurate if the correction factor is less than 20 %.

12.3 *Calculation of Sensor Activity:*

12.3.1 To determine the activity of a sensor, correct for the decay of the activated sensor during the irradiation period, the waiting period, and the counting period. This requires the activity that would have existed if all of the fluence struck the foil in a time short compared to the half-life of the reaction of interest in the foil. This is because for fluence determination it is the total number of reactions that is needed. This activity is called A_o and is generally different from the activity A_i at the end of the irradiation. Thus, for a steady-state irradiation at a constant fluence rate, the foil activity for fluence determination is given as follows:

$$A_o = \frac{N_\gamma(t_c)\lambda\, {}^2 t_i\, e^{\lambda t_w}}{P_\gamma Y_f(1 - e^{-\lambda t_c})(1 - e^{-\lambda t_i})} \quad (5)$$

where:

A_o = activity of the foil, that is, disintegration rate of neutron-activated nuclide of interest, corrected for decay during the irradiation, number per second,
N_γ = gamma-ray emission (photons emitted during t_c from Eq 4),
λ = decay constant of product nuclide (= 0.693/half-life),
P_γ = probability per nuclear disintegration for the emission of the gamma ray being counted,
Y_f = fractional fission yield for the product of interest (for fission foils),
t_w = waiting time between end of irradiation period and beginning of counting period,
t_c = counting period, and
t_i = irradiation period.

12.3.2 Additional corrections to Eq 3 are required if significant neutron self-shielding or fluence depression occurs during irradiation. The value of A_o can be found from the activity at the end of the irradiation, A_i, by multiplying A_i by $\lambda t_i/(1 - e^{-\lambda t_i})$ provided the fluence rate during t_i was a constant. For short irradiation times, $t_i << 1/\lambda$, the correction approaches 1.0 and A_i approaches A_o. See the appendix for the derivations.

12.3.3 The absolute-specific-decay-corrected activity of a foil of Isotope j is given by the following equation:

$$R_j = \frac{A_o}{N_o} \quad (6)$$

where:

R_j = specific activity of isotope j (disintegrations per second per atom available for activation) assuming correction has been made for decay during t_i, and
N_o = number of atoms of isotope j in the foil available for activation.

The factor N_o can be expressed as follows:

$$N_o = \frac{N_A f m}{M} \quad (7)$$

where:

N_A = Avogadro's number, 6.0225×10^{23},
f = fractional mass abundance of Isotope j in the foil,
m = total mass of the foil, and
M = atomic mass of activated nuclide.

12.3.4 The specific activities of all the activation foils are used as input data for an appropriate computer code to determine the energy-fluence spectrum of the neutron source. See Guide E721.

13. Precision and Bias

13.1 The factors that determine the uncertainty of the measured sensor specific activities are as follows:

13.1.1 Counting statistics,

13.1.2 Reproducibility of the location of the standard source or foil with respect to the detector,

13.1.3 Reproducibility in the determination of net full-energy peak counts (peak-area analysis, background subtraction, and coincidence photon summing correction),

13.1.4 Systematic uncertainties, which are considered separately in 13.3,

13.1.5 Error associated with the positioning of foils in a nonuniform field,

13.1.6 Monitor normalization between separate runs, and

13.1.7 Uncertainties in the cross sections of the reactions and of the covers.

13.2 For example, a typical foil-counting system could have magnitudes of random errors (each equaling one standard deviation, σ) as follows:

Source of Random Error	Uncertainties % (Non-Fission Foil)	Uncertainties % (Fission Foil)
Counting statistics, σ	1.0	1.0
Source or foil location, σ	1.0	2.0
Peak counts, σ	1.0	2.0
Total, σ_τ	1.7	3.0

The total random error, σ_τ, in this example is obtained by combining the individual values in quadrature (that is, the square root of the sum of the squares).

13.3 Sources of systematic (nonrandom) uncertainties (along with examples of typical estimated values of these uncertainties) are listed as follows:

Source of Systematic Error	Uncertainties % (Non-Fission Foil)	Uncertainties % (Fission Foil)
Calibration of gamma detector	3.0	3.0
Correction for finite source size	0.5	0.5
Correction for analyzer deadtime and pileup	1.0	1.0
Correction for self-shielding and fluence depression	2.0	2.0
Nuclear data:		

Branching ratios of gamma ray scheme	2.0	2.0
Fission yields	...	3.0
Half-lives	1.0	1.0
Total, errors combined in quadrature	4.3	5.2

13.4 An estimate of the overall uncertainty in determining the specific activity of a foil is obtained by combining the random and systematic uncertainties in quadrature. Thus, for the example in 13.2 and 13.3, the overall uncertainty for nonfission foils is $[(1.7)^2 + (4.3)^2]^{0.5} = 4.6$ %, and for the fission foils it is $[(3.0)^2 + (5.2)^2]^{0.5} = 6.0$ %. Of course, these values are meant to approximate the errors in counting a typical foil and would not necessarily be representative of any specific case.

13.5 Perform a detailed error analysis for each foil measured and give details in a report of the results.

14. Keywords

14.1 activation foils; neutron activation; neutron spectra; radiation-hardness testing

APPENDIX

(Nonmandatory Information)

X1. DETERMINATION OF ACTIVITIES FROM COUNTING DATA

X1.1 *Activity at the End of the Irradiation Period:*

X1.1.1 The first objective is to determine the activity at the end of the irradiation, A_i, from counts recorded during the count period, t_c, which starts a time t_w after the end of the irradiation period. Note that this relationship does not depend on the irradiation history before t_i. All that is sought here is the emission rate caused by the daughter isotopes, N_d, that are present at t_i. Their decay rate is λN_d, and it does not matter how the quantity N_d came to be.

X1.1.2 The activity at the beginning of the count period is as follows:

$$A(t_w) = A_i e^{-\lambda t_w} \tag{X1.1}$$

Then the number of gammas emitted during the count time, t_c, is as follows:

$$N_\gamma(t_w, t_c) = A_i e^{-\lambda t_w} \int_o^{t_c} e^{-\lambda t} dt,$$

$$= \frac{A_i}{\lambda} e^{-\lambda t_w} \left(1 - e^{-\lambda t_c}\right),$$

which leads to

$$A_i = \frac{N_\gamma \lambda e^{\lambda t_w}}{1 - e^{-\lambda t_c}} \tag{X1.2}$$

Notice that this relationship does not depend on the time history of the fluence or on t_i. Another relationship must be found to connect this activity to the activity A_o (Eq 3), and the fluence, Φ. But before that is done, the saturated activity, A_s, can be established. This is the activity that would exist after a steady-state irradiation for a time $t_i >> 1/\lambda$.

X1.2 *Saturated Activity:*

X1.2.1 The total rate of change of daughter isotopes during irradiation is as follows:

$$\frac{dN_d(\text{total})}{dt} = \bar{\sigma}\dot{\Phi}N - \lambda N_d(\text{total}) \tag{X1.3}$$

where:

$\bar{\sigma}$ = spectrum averaged cross section for the foil reaction,
$\dot{\Phi}$ = neutron fluence rate, and
N = number of target nuclei in the foil.

These radioactive nuclei are being created at a rate $\bar{\sigma}\dot{\varphi}N$ and decay at a rate λN_d(total). Only the second term on the right is responsible for the gamma rays that are to be counted.

X1.2.2 The solution to Eq X1.3 for the number of daughter nuclei present at the end of a steady-state irradiation at rate $\dot{\varphi}$ starting at $t = o$ and ending at $t = t_i$ is as follows:

$$N_d(\text{total}, t_i) = \frac{\bar{\sigma}\dot{\Phi}N}{\lambda}\left(1 - e^{-\lambda t_i}\right) \tag{X1.4}$$

The decay rate is then λN_d(total, t_i), the second term on the right of Eq X1.3.

X1.2.3 The decay rate is related to the gamma-ray emission rate needed to calculate the activity by the relation as follows:

$$A_i = \lambda P_\gamma Y_f N_d(\text{total}, t_i) \tag{X1.5}$$

where P_γ and Y_f are the probabilities defined in Eq 5. (From this direction of derivation (from fluence rate back to activity at t_i), the activity does depend on t_i.) After a period of irradiation long compared to 1/λ, the decay rate will equal the creation rate, from Eq X1.3 (where dN_d (total)/dt = 0), as follows:

$$P_\gamma Y_f\left(\lambda N_d = \bar{\sigma}\dot{\Phi}N\right) = A_s$$

Thus

$$A_s = \frac{A_i}{\left(1 - e^{-\lambda t_i}\right)}$$

Again the condition was that A_i was established by a steady-state irradiation for a period t_i. The conversion factor from A_i to A_s is very large if the irradiation time is <<1/λ.

X1.3 *Relationship of A_o and A_i:*

X1.3.1 Since the activity is proportional to the number of daughter isotopes, the relationship between A_o (the activity that would have been induced by a Φ deposited during a $t_i << 1/\lambda$) and A_i (the activity at the end of the irradiation at t_I) by comparing the number of daughter isotopes created by Φ with

those remaining at the end of the irradiation at t_i. This is because the multiplying factors ($\lambda P_\gamma Y_f$) are the same.

$$N_{do} = \bar{\sigma}\Phi N = \text{total daughters generated} \quad \text{(X1.6)}$$

From Eq X1.4 calculate as follows for a steady fluence rate:

$$N_d(\text{total}, t_i) = \frac{\bar{\sigma}\dot{\Phi}N}{\lambda}\left(1 - e^{-\lambda t_i}\right)$$

$$= \frac{\bar{\sigma}\dot{\Phi}N}{\lambda t_i}\left(1 - e^{-\lambda t_i}\right)$$

Therefore

$$\frac{N_{do}}{N_d(\text{total}, t_i)} = \frac{A_o}{A_i} = \frac{\lambda t_i}{\left(1 - e^{-\lambda t_i}\right)} \quad \text{(X1.7)}$$

For short pulses $A_o = A_i$. It is important to note here that N_{do} is not really related to the N_d in Eq X1.5 even though they look similar. N_{do} is the number of daughter isotopes created by the fluence Φ. N_d in Eq X1.5 is the equilibrium population at constant fluence rate $\dot{\Phi}$.

X1.3.2 The procedure for finding the activity, A_o, to determine the fluence in the manner of Test Method E265, is to find the activity at the end of the irradiation with Eq X1.2 and then multiply it by the correction factor in Eq X1.7. This yields Eq 5. This is also the same activity used in determining a spectrum from a set of foils as is discussed in Guide E721.

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