INTERNATIONAL

~~Designation:E 261–98~~ Designation: E261 – 10

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

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

1. Scope

1.1 This practice describes ~~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~~E1005.

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

NOTE ~~1—Detailed methods for individual detectors are given in the following ASTM test methods: E 262E 262, E 263E 263, E 264E 264, E 265E 265, E 266E 266, E 343E 343, E 393E 393, E 418E 418, E 481E 481, E 523E 523, E 526E 526, E 704E 704, E 705E 705, and E 854E 844~~ 1—Detailed methods for individual detectors are given in the following ASTM test methods: E262, E263, E264, E265, E266, E343, E393, E481, E523, E526, E704, E705, and E854.

2. Referenced Documents

2.1 *ASTM Standards:*[2]

E170 Terminology Relating to Radiation Measurements and Dosimetry
E181 Test Methods for Detector Calibration and Analysis of Radionuclides
E262 Test Method for Determining Thermal Neutron Reaction Rates and Thermal Neutron Fluence Rates by Radioactivation Techniques
E263 Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Iron
E264 Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Nickel
E265 Test Method for Measuring Reaction Rates and Fast-Neutron Fluences by Radioactivation of Sulfur-32
E266 Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Aluminum
E343 Test Method for Measuring Reaction Rates by Analysis of Molybdenum-99 Radioactivity ~~from~~From Fission Dosimeters
E393 ~~Test Method for Measuring Reaction Rates by Analysis of Barium-140 from Fission Dosimeters~~[2]
~~E418Method for Measuring Fast-Neutron Flux by Track-Etch Technique~~ Test Method for Measuring Reaction Rates by Analysis of Barium-140 From Fission Dosimeters
E481 Test Method for Measuring Neutron Fluence Rates by Radioactivation of Cobalt and Silver
E523 Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Copper
E526 ~~Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Titanium~~[2] Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Titanium
E693 Practice for Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacements Per Atom (DPA), E 706(ID)
E704 Test Method for Measuring Reaction Rates by Radioactivation of Uranium-238
E705 ~~Test Method for Measuring Reaction Rates by Radioactivation of Neptunium-237~~[2] Test Method for Measuring Reaction Rates by Radioactivation of Neptunium-237
E722 Practice for Characterizing Neutron Fluence Spectra in Terms of an Equivalent Monoenergetic Neutron Fluence for Radiation-Hardness Testing of Electronics

[1] This practice is under the jurisdiction of ASTM Committee ~~E-10~~ E10 on Nuclear Technology and Applications and is the direct responsibility of Subcommittee E10.05 on Nuclear Radiation Metrology.

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[2] For referenced ASTM standards, visit the ASTM website, www.astm.org, or contact ASTM Customer Service at service@astm.org. For *Annual Book of ASTM Standards* ~~, Vol 12.02.~~volume information, refer to the standard's Document Summary page on the ASTM website.

E844 Guide for Sensor Set Design and Irradiation for Reactor Surveillance, ~~E 706(IIC)[2]~~E 706 (IIC)
E854 Test Method for Application and Analysis of Solid State Track Recorder (SSTR) Monitors for Reactor Surveillance, ~~E 706(IIIB)[2]~~E706(IIIB)
E944 Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance, E 706 (IIA)
E1005 Test Method for Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance, E 706 (IIIA)
E1018 ~~Guide for Application of ASTM Evaluated Cross Section Data File, Matrix E 706 (IIB)[2]~~ Guide for Application of ASTM Evaluated Cross Section Data File, Matrix E706 (IIB)
E2005 Guide for Benchmark Testing of Reactor Dosimetry in Standard and Reference Neutron Fields

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~~E170.

4. Summary of Practice

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

5. Significance and Use

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

6. Counting Apparatus

6.1 A number of instruments are used to determine the disintegration rate of the radioactive product of the neutron-induced reaction. These include the scintillation counters, ionization chambers, proportional counters, Geiger tubes, and solid state detectors. Recommendations of counters for particular applications are given in General Methods ~~E 181~~E181.

7. Requirements for Activation-Detector Materials

~~7.1The general considerations~~ 7.1 Considerations concerning the suitability of a material for use as an activation detector are found in Guide ~~E 844~~E844.

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

7.3 A detailed description of procedures for the use of fission threshold detectors is given in Test Methods ~~E 343, E 393, and E 854~~E343, E393, and E854, and Guide ~~E 844~~E844.

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)**.[3]

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

8.3 It is desirable, but not required, that the neutron detector be irradiated during the entire time period considered and that a measurable part of the activity generated during the initial period of irradiation be present in the detector at the end of the irradiation. Therefore, the effective half-life, $t'_{1/2} = 0.693/\lambda'$, 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:

~~[3] Discontinued—see *1984 Annual Book of ASTM Standards*, Vol 12.02.~~
[3] The boldface numbers in parentheses refer to a list of references at the end of this standard.

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

E0261-10_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:

E0261-10_2

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

)) E0261-10_3

where:

t_i = exposure duration, and
$\lambda`$ = 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:

E0261-10_4

where:

$\sigma_a(E)$ = neutron absorption cross section for the product nuclide, and
$\varphi(E)$ = 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:

E0261-10_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.

E0261-10_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:

E0261-10_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~~E1018 provides information and recommendations on how to select the cross section library. The International Reactor Dosimetry File ~~(IRDF-90)~~(IRDF-2002) **(~~3~~83)** is one good source for cross sections. The SNLRML cross section compendium **~~(25~~(4)** 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 $\varphi(E,t)$ by the following expression:

E0261-10_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:

E0261-10_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, φ(*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:

E0261-10_10

where:

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

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

− Npλ` E0261-10_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:

~~9.6.5The saturation activity of a foil is defined as the activity when d~~*~~n~~* $(1 - \exp(-\lambda' t))$ *E*0261-10_12

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

E0261-10_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: ~~d*N*/d*t*=*N*~~

E0261-10_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~~ *may, in that case, 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 φ(*E*) is essentially constant. Then the activity generated during the i^{th} irradiation period is:

E0261-10_15

where:

N_i = ~~number of target atoms, and~~ number of target atoms during the i^{th} period, 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:

E0261-10_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:

)) E0261-10_17

and Eq 16 can be written as:

E0261-10_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 ~~E0261-98_19~~

E0261-10_19

N_i =

$$N_o\exp\left(-R_s\sum_{j=1}^{i-1}\frac{P_j}{P}t_j\right).$$ ~~E0261-98_20~~

E0261-10_20

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~~built in (for example, fission products produced from ^{239}Pu that arises from neutron capture in ^{238}U). In these cases the number of atoms of the new target isotope(s) must be calculated for each time interval and Eq 15 used to determine the additional activity to be added to that from the original target nuclide.

9.8 *Spectral-Averaged Cross Sections*:

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

9.8.1 Spectral-averaged cross sections are used in reaction rate calculations. A spectral-averaged cross section is defined as follows:

E0261-10_21

~~the~~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, in order to calculate the spectral-averaged cross section. 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:

$$U = 1n(E_0/E) \tag{20}$$

E0261-10_22

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

E0261-10_23

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,~~epithermal (or resonance), and fast. Since these ranges have distinctive distributions, they are a natural division of neutrons by energy for thermal reactor spectra.

9.10.1.1 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~~epithermal 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~~approximately Maxwellian (except when a strong neutron absorber is present).

9.10.2 The thermal-neutron component overlaps the ~~resonance-neutron~~epithermal-neutron component somewhat while the ~~resonance-neutron~~epithermal-neutron component and the fast-neutron component also overlaps. The exact energy limits between the components are somewhat arbitrary but the choice is influenced by the cross-section characteristics of the isotopes used to detect the neutrons in each energy range. The energy limits ~~assumed~~adopted for this ~~procedure~~practice are 0 to ~~0.50~~0.55 eV for thermal neutrons, ~~0.50 eV~~ $5\sqrt{T/T_0}\,(0.0253)\,eV$ to 0.10 MeV for ~~resonance~~epithermal neutrons, where T and T_0 are neutron temperatures, as defined in 9.11.2, 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:

E0261-10_25

where:

~~(nv)~~ $R_{s,th}$	=	the reaction rate attributable to thermal neutrons only,
$(nv)_{th}$	=	true thermal-neutron fluence rate;
n	=	~~neutron density, and~~ neutron density, neutrons per unit volume,
v	=	neutron speed, 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~~speed of ~~V~~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}~~). This is the most probable speed of the Maxwellian distribution for a standard temperature whose value is 20.44°C (293.6 K). For a neutron activation detector with a cross section proportional to $1/v$, the reaction rate is

TABLE 1 Thermal-Neutron Detectors

Element	Reaction	Thermal Cross Section[A] (b)	Product Nucleus[B]: Half-Life[C]	E_γ[D] (keV)	Yield (%)[D] γ per Reaction	E_β[D] (keV)	Yield (%)[D] β per Reaction	Comments
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 %~~	~~2.6943(8) d~~	~~1087.684(3)~~	~~0.1590(20)~~	~~962.(1)~~	~~98.990(6)~~	~~GH~~
Gold	$^{197}Au(n,\gamma)^{198}Au$	98.69 ± 0.14 %	2.69517(21) d	1087.6842(7)	0.159(3)	960.4(10)	98.986(10)	G,H
				~~675.8836(7)~~	~~0.8038(29)~~			
				675.8836(7)	0.806(7)			
				~~411.8044(11)~~	~~95.57(47)~~			
				411.802504(17)	95.54(47)			
~~Cobalt~~	~~$^{59}Co(n,\gamma)^{60}Co$~~	~~37.233 ± 0.16 %~~	~~1925.5(5) d~~	~~1173.238(4)~~	~~99.857(22)~~	~~317.88(10)~~	~~99.883(21)~~	~~G~~
Cobalt	$^{59}Co(n,\gamma)^{60}Co$	37.233 ± 0.16 %	1925.28(14) d	1173.22(3)	99.85(3)	317.32(21)	99.883(3)	G
				~~1332.502(5)~~	~~99.983(6)~~	~~1492.16(13)~~	~~0.117(20)~~	
				1332.49(4)	99.9826(6)	1490.56(21)	0.12(3)	
~~Manganese~~	~~$^{55}Mn(n,\gamma)^{56}Mn$~~	~~13.413 ± 1.5 %~~	~~2.5785(2) h~~	~~846.754(20)~~	~~98.87(30)~~	~~2848.9(9)~~	~~56.3(10)~~	
Manganese	$^{55}Mn(n,\gamma)^{56}Mn$	13.413 ± 1.5 %	0.107449(19) d	846.7538(19)	98.85(3)	2848.7(3)	56.6(7)	G
				~~1810.72(4)~~	~~27.19(79)~~	~~1038.2(9)~~	~~27.9(8)~~	
				1810.726(4)	26.9(4)	1037.9(3)	27.5(4)	
				~~2113.05(4)~~	~~14.34(40)~~	~~735.9(9)~~	~~14.6(4)~~	
				2113.092(6)	14.2(3)	735.6(3)	14.5(4)	
~~Sodium~~	~~$^{23}Na(n,\gamma)^{24}Na$~~	~~0.528 ± 0.95 %~~	~~0.62356(17) d~~	~~1368.633(6)~~	~~99.9936(15)~~	~~1389.05(29)~~	~~99.9310(16)~~	~~G~~
Sodium	$^{23}Na(n,\gamma)^{24}Na$	0.528 ± 0.95 %	0.62329(6) d	1368.626(5)	99.9935(5)	1392.94(16)	99.939(8)	G
				~~2754.030(5)~~	~~99.855(5)~~			
				2754.007(11)	99.872(8)			

[A] 2200 ms cross section (E = 0.0253 eV, T = 20°C), taken from the cross section files recommended in Ref~~erence~~ (**2~~5~~4**). Uncertainty data is taken from Ref~~erence~~ (**2~~8~~5**) for all thermal cross sections unless otherwise noted.

[B] Sources for half life and ~~dec~~ ga~~y~~mma radiation data in this table are consistent with that from Ref~~erence~~ (**2~~5~~4**)~~ and with Standards E 1018–95~~.

[C] Original source is Ref~~erence~~ (**26**).

[D] Original source is Ref~~erence~~ (**2~~7~~5**).

[E] Source for cross section is Ref~~erence~~ (**2~~8~~5**). This dosimetry reaction is not in Ref~~erence~~ (**2~~5~~4**).

[F] This number represents an update of information in Ref~~erence~~ (**2~~5~~4**) and represents an update in the original source data.

[G] Original source for ~~half life an~~ d-decay radiations is Ref~~erence~~ (**2~~8~~7**). This reference is a standard for detector calibration and takes preceden~~t~~ce for isotopes used as calibration standards.

[H] Cross sections and uncertainty come from Ref~~erence~~ (**3~~0~~8**).