INTERNATIONAL

Designation: E496 – 14 (Reapproved 2022)

Standard Test Method for Measuring Neutron Fluence and Average Energy from $^{3}H(d,n)^{4}He$ Neutron Generators by Radioactivation Techniques[1]

This standard is issued under the fixed designation E496; 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 test method covers a general procedure for the measurement of the fast-neutron fluence rate produced by neutron generators utilizing the $^{3}H(d,n)^{4}He$ reaction. Neutrons so produced are usually referred to as 14-MeV neutrons, but range in energy depending on a number of factors. This test method does not adequately cover fusion sources where the velocity of the plasma may be an important consideration.

1.2 This test method uses threshold activation reactions to determine the average energy of the neutrons and the neutron fluence at that energy. At least three activities, chosen from an appropriate set of dosimetry reactions, are required to characterize the average energy and fluence. The required activities are typically measured by gamma-ray spectroscopy.

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

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, health, and environmental practices and determine the applicability of regulatory limitations prior to use.*

1.5 *This international standard was developed in accordance with internationally recognized principles on standardization established in the Decision on Principles for the Development of International Standards, Guides and Recommendations issued by the World Trade Organization Technical Barriers to Trade (TBT) Committee.*

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

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

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

E720 Guide for Selection and Use of Neutron Sensors for Determining Neutron Spectra Employed in Radiation-Hardness Testing of Electronics

2.2 *International Commission on Radiation Units and Measurements (ICRU) Reports:*[3]

ICRU Report 13 Neutron Fluence, Neutron Spectra and Kerma

ICRU Report 26 Neutron Dosimetry for Biology and Medicine

2.3 *ISO Standard:*[4]

Guide to the Expression of Uncertainty in Measurement

2.4 *NIST Document:*[5]

Technical Note 1297 Guidelines for Evaluating and Expressing the Uncertainty of NIST Measurement Results

3. Terminology

3.1 *Definitions*—Refer to Terminology E170.

4. Summary of Test Method

4.1 This test method describes the determination of the average neutron energy and fluence by use of three activities

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

Current edition approved July 1, 2022. Published July 2022. Originally approved in 1973. Last previous edition approved in 2014 as E496 – 14. DOI: 10.1520/E0496-14R22.

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

[3] Available from the International Commission on Radiation Units, 7910 Woodmont Ave., Washington, DC 20014.

[4] Available from American National Standards Institute (ANSI), 25 W. 43rd St., 4th Floor, New York, NY 10036, http://www.ansi.org.

[5] Available from National Institute of Standards and Technology (NIST), 100 Bureau Dr., Stop 1070, Gaithersburg, MD 20899-1070, http://www.nist.gov.

from a select list of dosimetry reactions. Three dosimetry reactions are chosen based on a number of factors including the intensity of the neutron field, the reaction half-lives, the slope of the dosimetry reaction cross section near 14 MeV, and the minimum time between sensor irradiation and the gamma counting. The activities from these selected reactions are measured. Two of the activities are used, in conjunction with the nuclear data for the dosimetry reactions, to determine the average neutron energy. The third activity is used, along with the neutron energy and nuclear data for the selected reaction, to determine the neutron fluence. The uncertainty of the neutron energy and the neutron fluence is determined from the activity measurement uncertainty and from the nuclear data.

5. Significance and Use

5.1 Refer to Practice E261 for a general discussion of the measurement of fast-neutron fluence rates with threshold detectors.

5.2 Refer to Test Method E265 for a general discussion of the measurement of fast-neutron fluence rates by radioactivation of sulfur-32.

5.3 Reactions used for the activity measurements can be chosen to provide a convenient means for determining the absolute fluence rates of 14-MeV neutrons obtained with ^{3}H(*d, n*)^{4}He neutron generators over a range of irradiation times from seconds to approximately 100 days. High-purity threshold sensors referenced in this test method are readily available.

5.4 The neutron-energy spectrum must be known in order to measure fast-neutron fluence using a single threshold detector. Neutrons produced by bombarding a tritiated target with deuterons are commonly referred to as 14-MeV neutrons; however, they can have a range of energies depending on: (*1*) the angle of neutron emission with respect to the deuteron beam, (*2*) the kinetic energy of the deuterons, and (*3*) the target thickness. In most available neutron generators of the

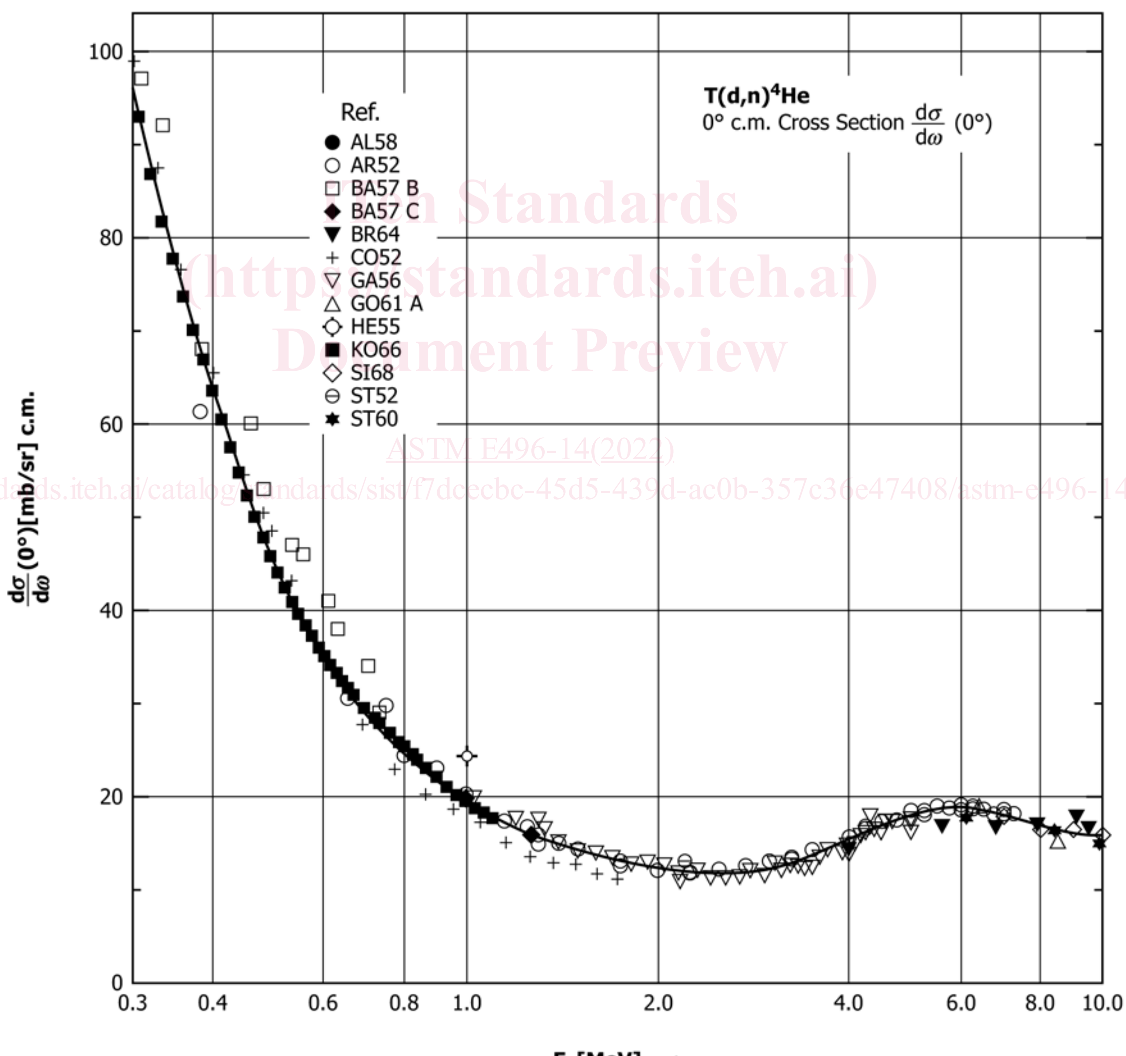

FIG. 1 Variation of 0 Degree ^{3}H(*d,n*)^{4}He Differential Cross Section with Incident Deuteron Energy (1)

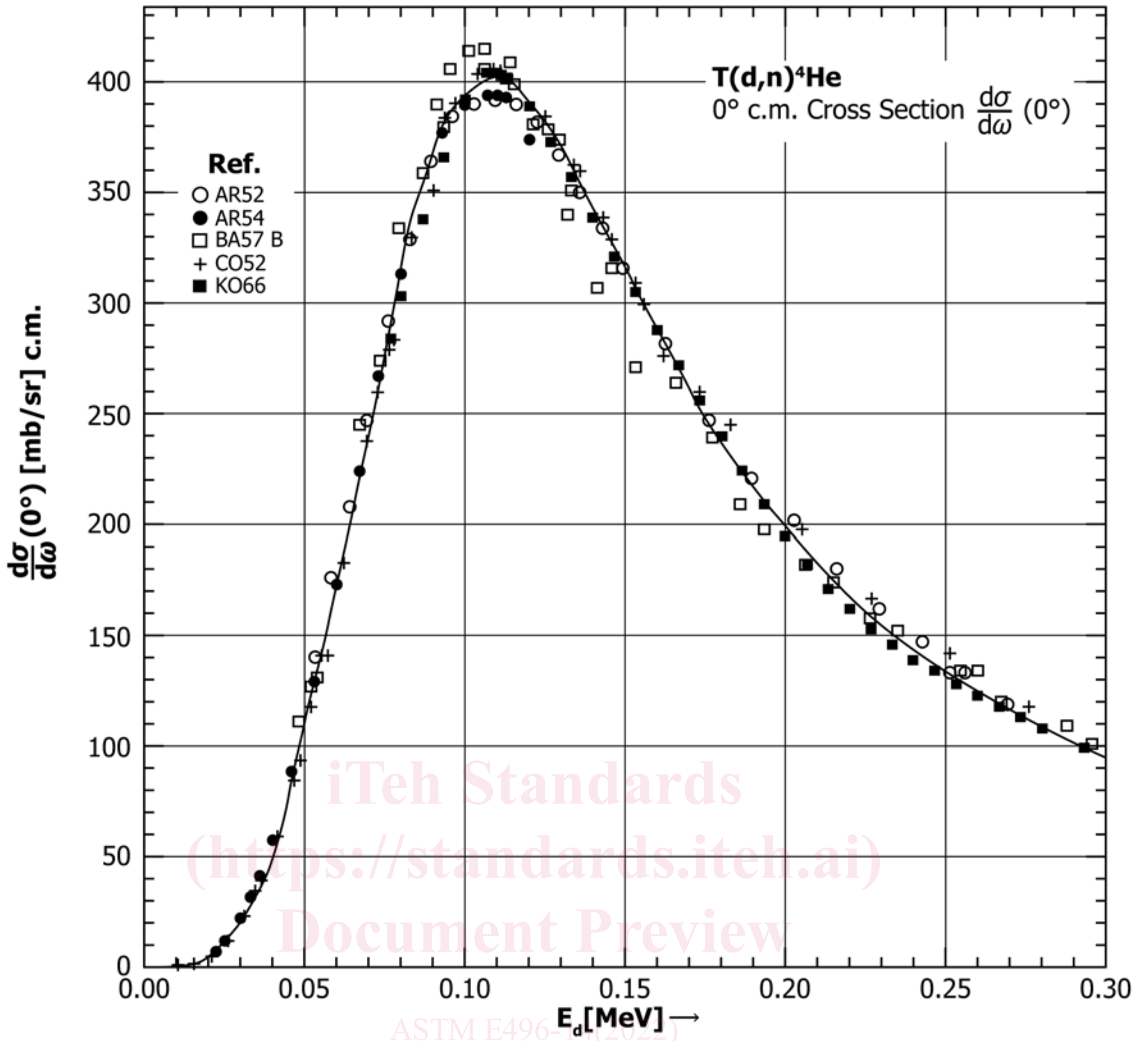

FIG. 2 Variation of 0 Degree $^3H(d,n)^4He$ Differential Cross Section with Incident Deuteron Energy (1)

Cockroft-Walton type, a thick target is used to obtain high-neutron yields. As deuterons penetrate through the surface and move into the bulk of the thick target, they lose energy, and interactions occurring deeper within the target produce neutrons with correspondingly lower energy.

5.5 Wide variations in neutron energy are not generally encountered in commercially available neutron generators of the Cockroft-Walton type. Figs. 1 and 2 (**1**)[6] show the variation of the zero degree $^3H(d,n)^4He$ neutron production cross section with energy, and clearly indicate that maximum neutron yield is obtained with deuterons having energies near the 107 keV resonance. Since most generators are designed for high yield, the deuteron energy is typically about 200 keV, giving a range of neutron energies from approximately 14 to 15 MeV. The differential center-of-mass cross section is typically parameterized as a summation of Legendre polynomials. Figs. 3 and 4 (**1, 2**) show how the neutron yield varies with the emission angle in the laboratory system. The insert in Fig. 4 shows how the magnitude, A_1, of the $P_1(\theta)$ term, and hence the asymmetry in the differential cross section grows with increasing energy of the incident deuteron. The nonrelativistic kinematics (valid for $E_d < 20$ MeV) for the $^3H(d,n)^4He$ reaction show that:

$$E_n^{1/2} = 0.28445 E_d^{1/2} \times \cos\theta + \frac{\left(2.031 E_d \times \cos\theta^2 + 352.64228 + 9.95998 E_d\right)^{1/2}}{5.01017} \quad (1)$$

where:

E_n = the neutron energy in MeV,
E_d = the incident deuteron energy in MeV, and
θ = the neutron emission angle with respect to the incident deuteron in the laboratory system.

5.5.1 Fig. 5 (**2**) shows how the neutron energy depends upon the angle of scattering in the laboratory coordinate system when the incident deuteron has an energy of 150 keV and is incident on a thick and a thin tritiated target. For thick targets, the incident deuteron loses energy as it penetrates the target and produces neutrons of lower energy. A thick target is defined as a target thick enough to completely stop the incident

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

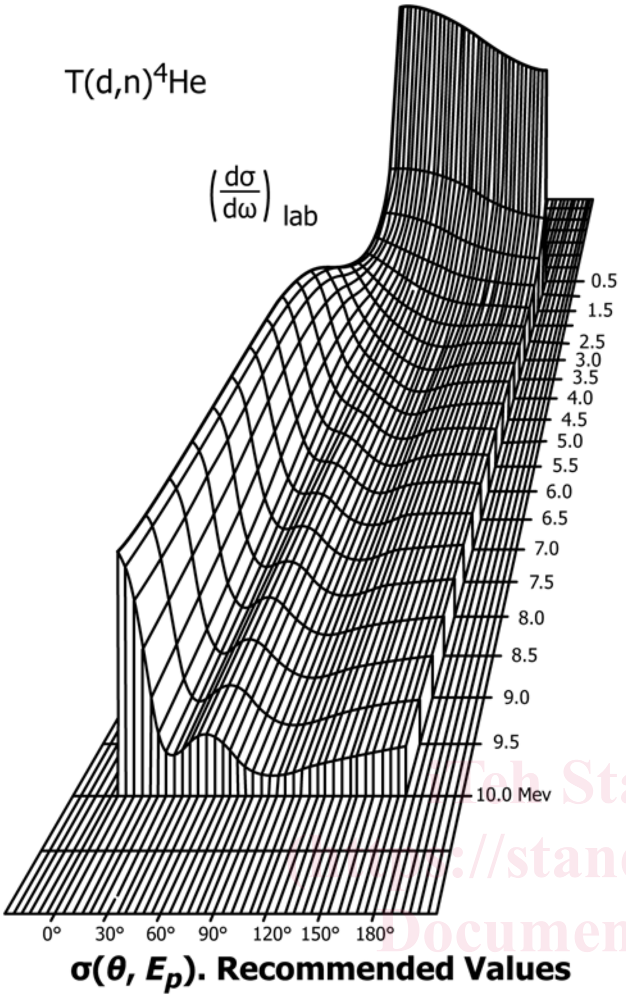

FIG. 3 Energy and Angle Dependence of the ^{3}H(d,n)^{4}He Differential Cross Section (1)

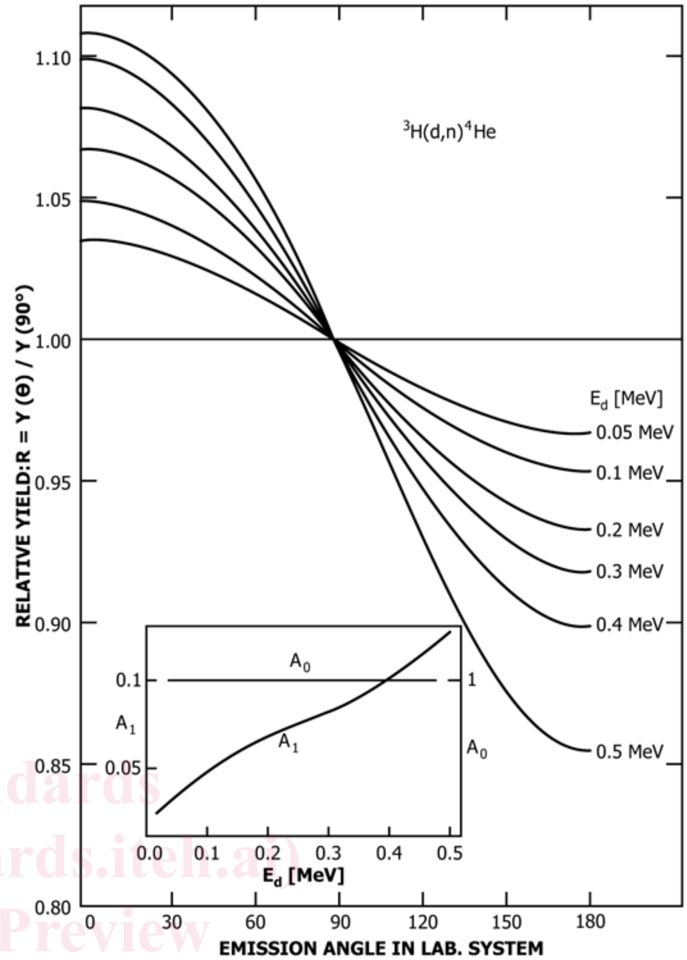

FIG. 4 Change in Neutron Energy from ^{3}H(d,n)^{4}He Reaction with Laboratory Emission Angle (2)

deuteron. The two curves in Fig. 5, for both thick and thin targets, come from different sources. The dashed line calculations come from Ref (**3**); the solid curve calculations come from Ref (**4**); and the measured data come from Ref (**5**). The dash-dot curve and the right-hand axis give the difference between the calculated neutron energies for thin and thick targets. Computer codes are available to assist in calculating the expected thick and thin target yield and neutron spectrum for various incident deuteron energies (**6**).

5.6 The Q-value for the primary ^{3}H(d,n)^{4}He reaction is +17.59 MeV. When the incident deuteron energy exceeds 3.71 MeV and 4.92 MeV, the break-up reactions ^{3}H(d,np)^{3}H and ^{3}H(d,2n)^{3}He, respectively, become energetically possible. Thus, at high deuteron energies (>3.71 MeV) this reaction is no longer monoenergetic. Monoenergetic neutron beams with energies from about 14.8 to 20.4 MeV can be produced by this reaction at forward laboratory angles (**7**).

5.7 It is recommended that the dosimetry sensors be fielded in the exact positions where the dosimetry results are wanted. There are a number of factors that can affect the monochromaticity or energy spread of the neutron beam (**7, 8**). These factors include the energy regulation of the incident deuteron energy, energy loss in retaining windows if a gas target is used or energy loss within the target if a solid tritiated target is used, the irradiation geometry, and background neutrons from scattering with the walls and floors within the irradiation chamber.

6. Apparatus

6.1 Either a NaI(Tl) or a Ge semiconductor gamma-ray spectrometer, incorporating a multichannel pulse-height analyzer is required. See Test Methods E181 for a discussion of spectrometer systems and their use.

6.2 If sulfur is used as a sensor, then a beta particle detector is required. The apparatus required for beta counting of sulfur is described in Test Methods E181 and E265.

6.3 A precision balance for determining foil masses is required.

7. Materials and Manufacture

7.1 High-purity threshold foils are available in a large variety of thicknesses. Foils of suitable diameter can be punched from stock material. Small diameter wire may also be used. Pre-punched and weighed high-purity foils are also available commercially. Guide E720 provides some details on typical foil masses and purity. Foils of 12.7 and 25.4 mm (0.50

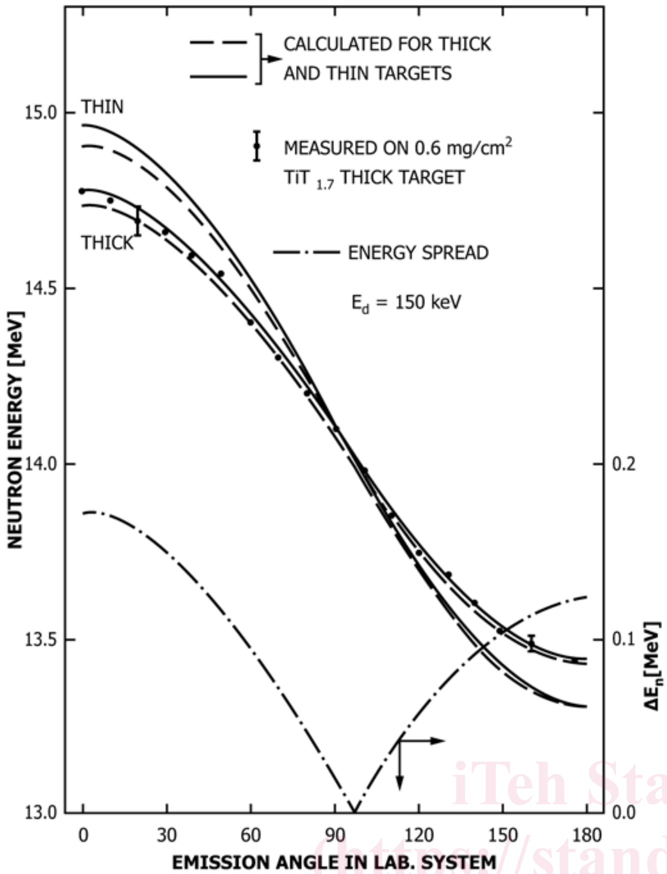

FIG. 5 Dependence of $^3H(d,n)^4He$ Neutron Energy on Angle (2)

and 1.00 in.) diameter and 0.13 and 0.25 mm (0.005 and 0.010 in.) thickness are typical.

7.2 See Test Method E265 for details on the availability and preparation of sulfur sensors.

8. Calibration

8.1 See Test Methods E181 for general detector calibration methods. Test Methods E181 addresses both gamma-ray spectrometers and beta counting methods.

9. Procedure for Determining the Neutron Energy

9.1 *Selection of Sensors:*

9.1.1 Use of an activity ratio method is recommended for the determination of the neutron energy. The activity ratio method has been described in Ref (**9**). This test method has been validated for ENDF/B-VI cross sections (**10**) in Ref (**11**).

9.1.2 Sensor selection depends upon the length of the irradiation, the cross section for the relevant sensor reaction, the reaction half-life, and the expected fluence rate. Table 1 lists some dosimetry-quality reactions that are useful in the 14-MeV energy region. The short half-lives of some of these reaction products, such as ^{27}Mg and ^{62}Cu, generally limit the use of these activation products to irradiation times of less than about 15 min. Table 2 and Fig. 6 show the recommended cross sections, in the vicinity of 14 MeV, for these reactions. The cross sections and uncertainties in Table 1 are from the IRDF-2002 (**12**) cross section compilation. The original source of each cross section is listed in the table. The SNLRML cross section compendium (**13**) is a single-point-of-reference alternative source for the cross sections and uncertainty data for the reactions mentioned in Table 1, but somewhat dated, reflecting larger uncertainties than IRDF 2002. The references for the other nuclear data in Table 1 are given in the table.

9.1.3 Longer high-fluence irradiations are recommended for the determination of the neutron energy. Table 3 and Fig. 7 give the neutron energy-dependent activity ratios for some commonly used sensor combinations. Fig. 8 displays some slopes for these ratios. In general, the larger the slope, the more sensitive the method is to the neutron energy. For the procedures of this standard to work, it is necessary for the ratios of the cross sections to be monotonic in the vicinity of 14 MeV, but the slopes need not be monotonic.

9.1.4 Table 4 shows the energy resolutions of some specific sensor combinations for a 14.5-MeV neutron source. The ^{58}Ni(n,2n)^{57}Ni-based combinations are recommended due to their steep slope and accurate dosimetry cross section evaluations.

9.2 *Determine the Sensor Mass*—Weigh each sensor to a precision of 0.1 %. Nonuniform foil thicknesses can result from the use of dull punches and frequently result in weight variation of 10 % or more.

9.3 *Irradiation of Sensors*—Irradiate the sensors, making certain that both sensors experience exactly the same fluence. The fluence gradients near a 14-MeV source tend to be high and it may be necessary to stack the sensors together or to mount them on a rotating disk during irradiation. Note the length of the irradiation, t_i, and the time the irradiation ended. Some sensors may have an interference reaction that is sensitive to low-energy neutrons. The interference reaction may be associated with the primary sensor element or with a contaminant material in the sensor. Of the reactions listed in Table 1, the use of a Cu sensor is the only case where the primary sensor element may be responsible for an interference reaction. In this case the useful ^{65}Cu(n,2n)^{64}Cu reaction activity must be distinguished from the ^{63}Cu(n,γ)^{64}Cu interference reaction activity (for example, by using an isotopically pure sensor or by experimentally verifying bounds on the maximum possible level of interference). Other examples of interference reactions from contaminant materials include trace impurities of Mn in Fe sensors and Na in Al sensors. Manganese is a frequent contaminant in Fe foils. In this case the ^{55}Mn(n,γ)^{56}Mn reaction interferes with the desired sensor response from the ^{56}Fe(n,p)^{56}Mn reaction. Salt from handling Al sensors can result in the ^{23}Na(n,γ)^{24}Na contaminant reaction which affects the use of the ^{27}Al(n,α)^{24}Na dosimetry sensor. If one is uncertain about the importance of an interference reaction that has a high thermal neutron cross section, it is recommended that the sensor be irradiated with and without a cadmium cover to quantify the importance of this interference term.

9.4 *Determination of Sensor Activity*—Guide E720 provides details on the calculational procedure for determining the activity of an irradiated sensor. The results of this step should be the activities, corrected to a time corresponding to the end of the irradiation. The activity should be corrected for decay

TABLE 1 Cross Section Parameters for Some Useful Reactions

	Dosimetry Reactions	Target Nucleus: Elemental Atomic Weight (14)	Target Nucleus: Isotopic Atomic Number Abundance, % (14)	Target Nucleus: Cross Section Source[A]	Target Nucleus: Cross Section Uncertainty Near 14 MeV, %	Product Nucleus: Half-Life (14)	Product Nucleus: E_γ, keV (15)	Product Nucleus: Yield, %, γ per Reaction (15)	Reaction Notes
1	$^{24}Mg(n,p)\ ^{24}Na$	24.3050	78.99	IRK	0.5	14.997 h	1368.626	99.9936	...
							2754.007	99.855	...
2	$^{27}Al(n,p)\ ^{27}Mg$	26.981539	100.0	RRDF-98	1.5	9.458 m	843.76	71.8	B
							1014.52	28.2	...
3	$^{27}Al(n,\alpha)^{24}Na$	26.981538	100.0	IRK	0.4	14.997 h	1368.626	99.9936	...
							2754.007	99.855	...
4	$^{32}S(n,p)\ ^{32}P$	32.065	94.99	ENDF/B-VI	4.7	14.262 d	$<E_\beta>$ = 695.03	100.0	C
5	$^{54}Fe(n,p)\ ^{54}Mn$	55.845	5.845	EDNF/B-VI	1.1	312.12 d	834.848	99.9760	...
6	$^{56}Fe(n,p)\ ^{56}Mn$	55.845	91.754	RRDF-98	1.1	2.5789 h	846.7638	98.85	...
							1810.726	26.9	...
							2113.092	14.2	...
7	$^{58}Ni(n,p)\ ^{58}Co$	58.6934	68.077	RRDF-98	2.0	70.86 d	810.7593	99.450	...
						9.10 h (meta)	863.951	0.686	D
							1674.725	0.517	...
							24.889	0.0397	...
8	$^{58}Ni(n,2n)\ ^{57}Ni$	58.6934	68.077	JEFF 3.0	0.1	35.60 h	1377.63	81.7	...
							1919.52	12.3	...
9	$^{63}Cu(n,2n)^{62}Cu$	63.546	69.15	ENDF/B-VI	1.5	9.673 m	1172.97	0.342	B,E
							875.66	0.147	...
10	$^{63}Cu(n,\alpha)\ ^{60}Co$	63.546	69.15	RRDF-98	1.7	1925.28 d	1173.228	99.85	...
							1332.492	99.9826	...
						10.467 m (meta)	58.603	2.0359	D
							826.28	0.0077	...
							1332.501	0.24	...
							2158.77	0.00072	...
11	$^{65}Cu(n,2n)^{64}Cu$	63.546	30.85	ENDF/B-VI	1.2	12.701 h	1345.77	0.475	...
12	$^{64}Zn(n,p)\ ^{64}Cu$	65.39	49.17	IRK	3.4	12.701 h	1345.77	0.475	...
13	$^{90}Zr(n,2n)^{89}Zr$	91.224	51.45	IRK	1.0	784.41 h	909.15	99.04	...
							1713.0	0.745	...
							1744.5	0.123	...
						4.161 m (meta)	587.8	89.62	D,F
							1507.4	6.06	...
14	$^{93}Nb(n,2n)^{92}mNb$	92.90638	100.0	RRDF-98	0.7	10.15 d	934.44	99.15	G
							912.6	1.78	...
							1847.5	0.85	...

[A] Original source. Cross sections and uncertainties used in this standard are taken from IRDF-2002.
[B] Use of this reaction requires accurate timing but also provides high specific activity per neutron.
[C] The β emissions are counted to determine the activity.
[D] The use of metastable states is not covered by this standard. Their use involves branching ratios, which may be energy dependent, and complicate the analysis. The metastable states reported here, with the exception of ^{89}Zr, decay to the ground state with almost 100 % probability, so the the ground-state reaction may be used with a branching ratio of 1.0 provided sufficient time is allotted for the metastable state to decay.
[E] Use of 511 keV line risks high background signals from other positron emitters.
[F] ^{89}mZr has a significant probability of production for the metastable state **(16)**, and also a significant probability for decay to other than the ground state **(17)**, so that a correction (~2 %) need be applied even for use of the use of the ground-state reaction. Its use is not covered by this standard.
[G] The cross section is particularly flat near 14 MeV, insensitive to neutron energy, and hence suitable for the measurement of fluence.

during the irradiation, as explained in Guide E720. This decay correction is especially important for short half-life reactions. The activity should have units of Bq per target atom.

9.5 *Calculations*—Section 11 details the calculations that use a ratio of two sensor activities to determine the neutron average energy.

10. Procedure for Determining the Neutron Fluence

10.1 *Selection of Sensor:*

10.1.1 To avoid sensitivity to uncertainty in the exact neutron energy, the 14-MeV neutron fluence sensor is generally chosen to have a flat response in the 13 MeV to 15 MeV energy region. Fig. 6 and Table 2 show the energy dependence near 14 MeV for some frequently used dosimetry sensors. An examination of Fig. 6 and Table 2 clearly indicates a strong preference to use the $^{93}Nb(n,2n)^{92m}Nb$ reaction. This preference is based on the flat energy response and the small cross section uncertainty near 14 MeV. The $^{93}Nb(n,2n)^{92m}Nb$ reaction has been used as a transfer standard for 14-MeV sources by national standards laboratories (**18**) and in international intercomparisons (**19**). The footnotes in Table 1 list some precautions about use of some other reactions. If the $^{93}Nb(n,2n)^{92m}Nb$ reaction cannot be used in a specific case, the uncertainty of the $^{3}H(d,n)^{4}He$ neutron energy, as determined from Section 9, should be used in conjunction with Table 2 and Fig. 6 to determine the best alternative reaction.

TABLE 2 Cross Sections (barn) Near 14 MeV for Dosimetry Reactions

Energy (MeV)		Reaction						
		^{24}Mg(*n*,*p*)^{24}Na	^{27}Al(*n*,*p*)^{27}Mg	^{27}Al(*n*,α)24Na	^{32}S(*n*,*p*)^{32}P	^{54}Fe(*n*,*p*)^{54}Mn	^{56}Fe(*n*,*p*)^{56}Mn	^{58}Ni(*n*,*p*)^{58}Co
1	13.55	2.0899e-01	8.2387e-02	1.2545e-01	2.8777e-01	3.7886e-01	1.1591e-01	4.1302e-01
2	13.65	2.0587e-01	8.1027e-02	1.2489e-01	2.8022e-01	3.7054e-01	1.1573e-01	4.0216e-01
3	13.75	2.0067e-01	7.9668e-02	1.2386e-01	2.7266e-01	3.6239e-01	1.1542e-01	3.9131e-01
4	13.85	1.9327e-01	7.8308e-02	1.2279e-01	2.6511e-01	3.5416e-01	1.1499e-01	3.8046e-01
5	13.95	1.9110e-01	7.6948e-02	1.2257e-01	2.5756e-01	3.4592e-01	1.1442e-01	3.6961e-01
6	14.05	1.9454e-01	7.5589e-02	1.2223e-01	2.5079e-01	3.3819e-01	1.1370e-01	3.5875e-01
7	14.15	1.9641e-01	7.4229e-02	1.2151e-01	2.4486e-01	3.3100e-01	1.1298e-01	3.4819e-01
8	14.25	1.9661e-01	7.2913e-02	1.2046e-01	2.3893e-01	3.2380e-01	1.1212e-01	3.3793e-01
9	14.35	1.9514e-01	7.1643e-02	1.1813e-01	2.3300e-01	3.1661e-01	1.1110e-01	3.2767e-01
10	14.45	1.9189e-01	7.0374e-02	1.1612e-01	2.2708e-01	3.0942e-01	1.1009e-01	3.1774e-01
11	14.55	1.8760e-01	6.9104e-02	1.1480e-01	2.2272e-01	3.0223e-01	1.0896e-01	3.0816e-01
12	14.65	1.8220e-01	6.7835e-02	1.1333e-01	2.2006e-01	2.9504e-01	1.0771e-01	2.9858e-01
13	14.75	1.7722e-01	6.6607e-02	1.1221e-01	2.1740e-01	2.8784e-01	1.0646e-01	2.8937e-01
14	14.85	1.7267e-01	6.5422e-02	1.1105e-01	2.1473e-01	2.8065e-01	1.0512e-01	2.8057e-01
15	14.95	1.7053e-01	6.4238e-02	1.0970e-01	2.1207e-01	2.7346e-01	1.0368e-01	2.7177e-01
16	15.05	1.7096e-01	6.3106e-02	1.0879e-01	2.0844e-01	2.6712e-01	1.0224e-01	2.6330e-01
17	15.15	1.7139e-01	6.2029e-02	1.0789e-01	2.0378e-01	2.6168e-01	1.0080e-01	2.5519e-01
18	15.25	1.7139e-01	6.0945e-02	1.0676e-01	1.9913e-01	2.5625e-01	9.9306e-02	2.4734e-01
19	15.35	1.6870e-01	5.9875e-02	1.0511e-01	1.9447e-01	2.5081e-01	9.7749e-02	2.3977e-01
20	15.45	1.6551e-01	5.8850e-02	1.0343e-01	1.8981e-01	2.4538e-01	9.6193e-02	2.3247e-01
21	15.55	1.6233e-01	5.7880e-02	1.0174e-01	1.8516e-01	2.3995e-01	9.4636e-02	2.2544e-01

Energy (MeV)		Reaction						
		^{58}Ni(*n*,2*n*)^{57}Ni	^{63}Cu(*n*,2*n*)^{62}Cu	^{63}Cu(*n*,α)^{60}Co	^{65}Cu(*n*,2*n*)^{64}Cu	^{64}Zn(*n*,*p*)^{64}Cu	^{90}Zr(*n*,2*n*)^{89}Zr	^{93}Nb(*n*,2*n*)^{92}mNb
1	13.55	1.3549e-02	3.6947e-01	4.7222e-02	8.3027e-01	2.0154e-01	4.4669e-01	4.5371e-01
2	13.65	1.5446e-02	3.8484e-01	4.7005e-02	8.4467e-01	1.9619e-01	4.8187e-01	4.5518e-01
3	13.75	1.7615e-02	4.0033e-01	4.6735e-02	8.5907e-01	1.9083e-01	5.1640e-01	4.5664e-01
4	13.85	1.9795e-02	4.1557e-01	4.6415e-02	8.7347e-01	1.8548e-01	5.5021e-01	4.5772e-01
5	13.95	2.1987e-02	4.3104e-01	4.6031e-02	8.8786e-01	1.8013e-01	5.8200e-01	4.5836e-01
6	14.05	2.4000e-02	4.4630e-01	4.5652e-02	9.0048e-01	1.7478e-01	6.1162e-01	4.5901e-01
7	14.15	2.5821e-02	4.6167e-01	4.5213e-02	9.1119e-01	1.6943e-01	6.4192e-01	4.5965e-01
8	14.25	2.7974e-02	4.7713e-01	4.4707e-02	9.2189e-01	1.6462e-01	6.7293e-01	4.6005e-01
9	14.35	3.0484e-02	4.9207e-01	4.4209e-02	9.3260e-01	1.6349e-01	7.0278e-01	4.5996e-01
10	14.45	3.2689e-02	5.0710e-01	4.3660e-02	9.4330e-01	1.6298e-01	7.3137e-01	4.5964e-01
11	14.55	3.4567e-02	5.2213e-01	4.3060e-02	9.5210e-01	1.6247e-01	7.5721e-01	4.5933e-01
12	14.65	3.6377e-02	5.3715e-01	4.2460e-02	9.5884e-01	1.6158e-01	7.8009e-01	4.5901e-01
13	14.75	3.8112e-02	5.5187e-01	4.1814e-02	9.6558e-01	1.5975e-01	8.0062e-01	4.5837e-01
14	14.85	3.9788e-02	5.6352e-01	4.1138e-02	9.7232e-01	1.5790e-01	8.1863e-01	4.5737e-01
15	14.95	4.1401e-02	5.7480e-01	4.0455e-02	9.7906e-01	1.5716e-01	8.3737e-01	4.5638e-01
16	15.05	4.3056e-02	5.8607e-01	3.9773e-02	9.8550e-01	1.5689e-01	8.5687e-01	4.5539e-01
17	15.15	4.4757e-02	5.9735e-01	3.9060e-02	9.9161e-01	1.5661e-01	8.7637e-01	4.5439e-01
18	15.25	4.6457e-02	6.0860e-01	3.8309e-02	9.9772e-01	1.5634e-01	8.9580e-01	4.5309e-01
19	15.35	4.8158e-02	6.1893e-01	3.7570e-02	1.0038e+00	1.5607e-01	9.1342e-01	4.5146e-01
20	15.45	4.9858e-02	6.2921e-01	3.6825e-02	1.0099e+00	1.5579e-01	9.3093e-01	4.4983e-01
21	15.55	5.1398e-02	6.3950e-01	3.6080e-02	1.0148e+00	1.5391e-01	9.4844e-01	4.4820e-01

10.1.2 Paragraph 9.1.2 indicates some other considerations in the choice of a dosimetry fluence reaction based on the irradiation length and expected strength.

10.2 *Determine the Sensor Mass*—Weigh the sensor to a precision of 0.1 %. Nonuniform foil thicknesses can result from the use of dull punches and frequently result in weight variations of 10 % or more.

10.3 *Irradiation of Sensor*—Paragraph 9.3 provides details and precautions on the irradiation of the sensor.

10.4 *Determination of Sensor Activity*—Guide E720 provides details on the calculational procedure for determining the activity on an irradiated sensor. The result of this step should be the activity, corrected to the time corresponding to the end of the irradiation, for the sensor selected in 10.1. The activity should be corrected for decay during the irradiation, as explained in Guide E720. The activity should have units of Bq per target atom.

10.5 *Calculations*—Section 12 details the calculations that use the sensor activity, in conjunction with the average neutron energy, to determine the neutron fluence.