Designation: E 496 – 96

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 E 496; 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 *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 *ASTM Standards:*

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

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

E 261 Practice for Determining Neutron Fluence Rates, Fluence, and Spectra by Radioactivation Techniques[2]

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

E 720 Guide for Selection and Use of Neutron-Activation Foils for Determining Neutron Spectra Employed in Radiation-Hardness Testing of Electronics[2]

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

ICRU Report 13—Neutron Fluence, Neutron Spectra and Kerma[3]

ICRU Report 26—Neutron Dosimetry for Biology and Medicine[3]

2.3 *ISO Standard:*

Guide to the Expression of Uncertainty in Measurement[4]

2.4 *NIST Document:*

Technical Note 1297—Guidelines for Evaluating and Expressing the Uncertainty of NIST Measurement Results[5]

3. Terminology

3.1 *Definitions*—Refer to Terminology E 170.

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 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 E 261 for a general discussion of the measurement of fast-neutron fluence rates with threshold detectors.

5.2 Refer to Test Method E 265 for a general discussion of the measurement of fast-neutron fluence rates by radioactivation of sulfur-32.

[1] This test method is under the jurisdiction of ASTM Committee E-10 on Nuclear Technology and Applicationsand is the direct responsibility of Subcommittee E10.07on Radiation Dosimetry for Radiation Effects on Materials and Devices.

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

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

[4] Available from American National Standards Institute, 11 W. 42nd St., 13th Floor, New York, NY 10036.

[5] Available from National Institute of Standards and Technology, Gaithersburg, MD 20899.

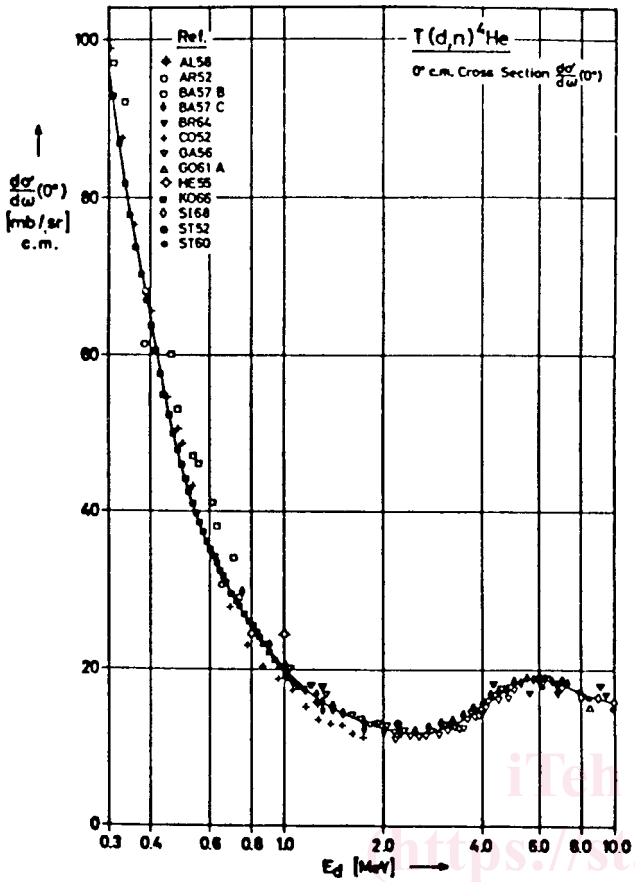

FIG. 1 Variation of 0 Degree $^3H(d,n)^4He$ Differential Cross Section with Incident Deuteron Energy (1)

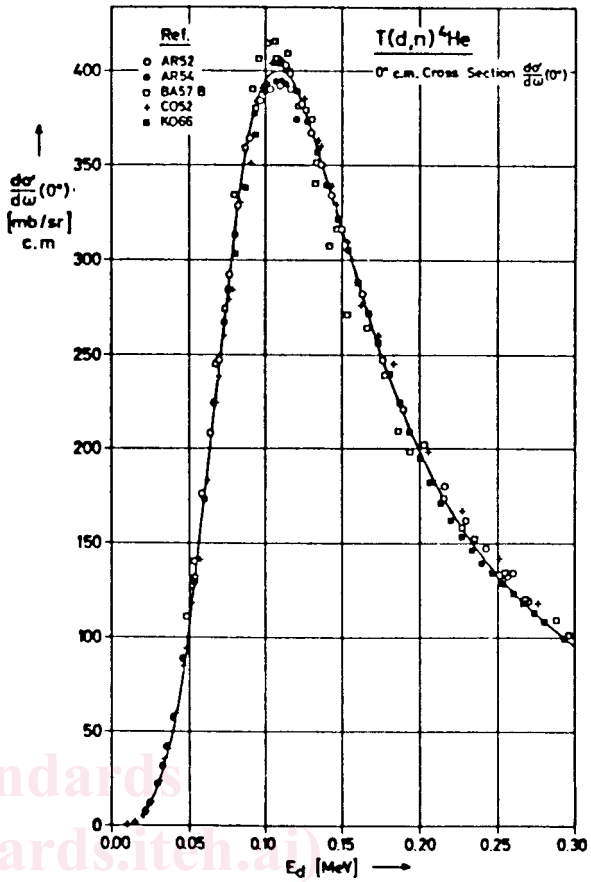

FIG. 2 Variation of 0 Degree $^3H(d,n)^4He$ Differential Cross Section with Incident Deuteron Energy (1)

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 $_3H(d,n)^4He$ 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 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:

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

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.

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

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

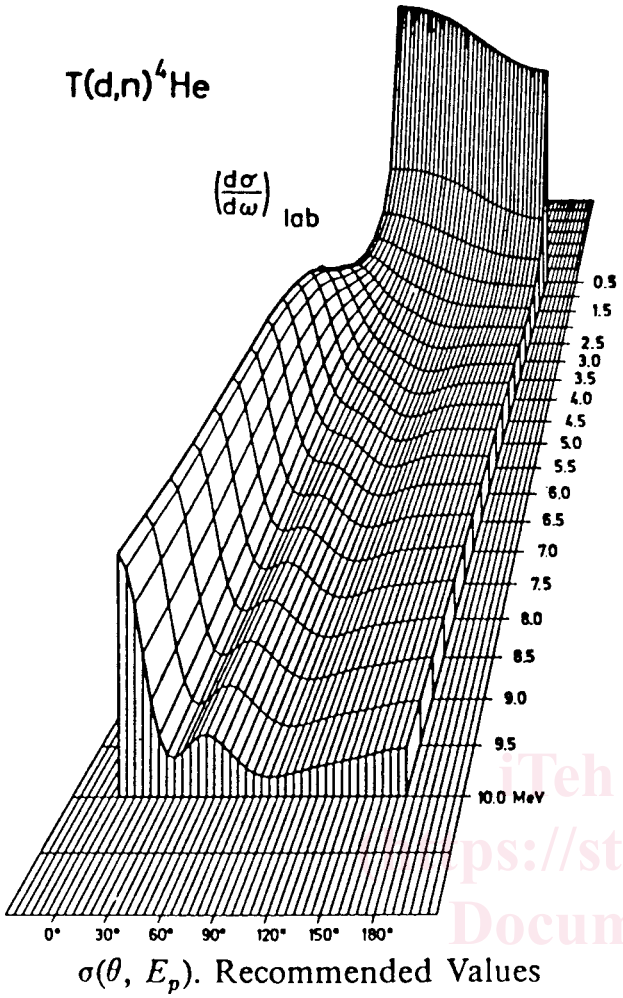

FIG. 3 Energy and Angle Dependence of the $^3H(d,n)^4He$ Differential Cross Section (1)

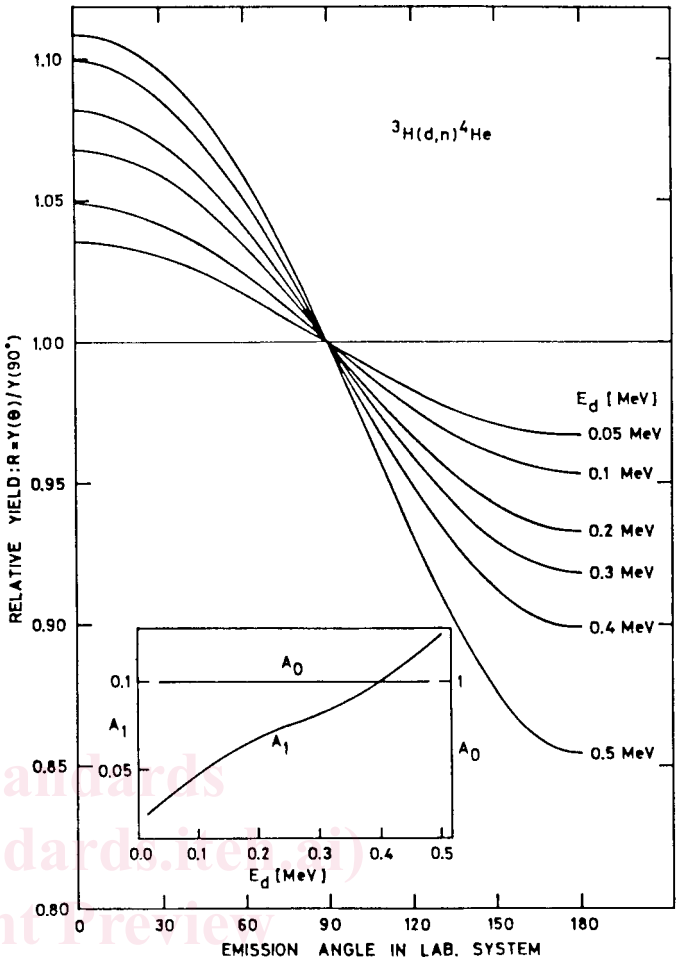

FIG. 4 Change in Neutron Energy From $^3H(d,n)^4He$ Reaction with Laboratory Emission Angle (2)

produces neutrons of lower energy. A thick target is defined as a target thick enough to completely stop the incident 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 gives the difference between the calculated neutron energies for thin and thick targets.

5.6 The Q-value for the primary $^3H(d,n)^4He$ reaction is + 17.59 MeV. When the incident deuteron energy exceeds 3.71 MeV and 4.92 MeV, the break-up reactions $^3H(d,np)^3H$ and $^3H(d,2n)^3He$, 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 **(6)**.

5.7 It is recommended that the dosimetry sensors be fielded in the exact positions that will be used for the customers of the 14-MeV neutron source. There are a number of factors that can affect the monochromaticity or energy spread of the neutron beam **(6,7)**. 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 E 181 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 E 181 and E 265.

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. Prepunched and weighed high purity foils are also available commercially. Guide E 720 provides some details on typical foil masses and purity. Foils of 12.7 and 25.3 mm (0.50 and 1.00 in.) diameter and 0.13 and 0.25 mm (0.005 and 0.010 in.) thickness are typical.

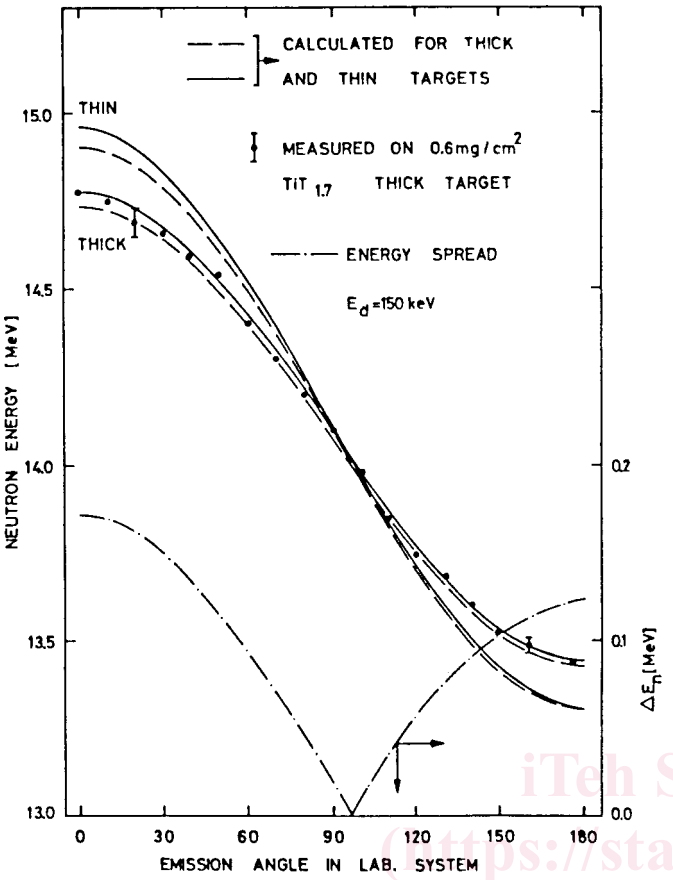

FIG. 5 Dependence of $^{3}H(d,n)^{4}He$ Neutron Energy on Angle (2)

7.2 See Test Method E 265 for details on the availability and preparation of sulfur sensors.

8. Calibration

8.1 See Test Methods E 181 for general detector calibration methods. Test Methods E 181 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 **(8)**. This test method was recently validated for ENDF/B-VI cross sections **(9)** in Ref **(10)**.

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 recommended in Table 1 are from the ENDF/B-VI and IRDF-90 **(15)** cross section compilations. The SNLRML cross section compendium **(16)** is a single-point-of-reference source for the recommended cross sections and uncertainty data for the reactions mentioned in Table 1. The references for the 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 Figs. 7 and 8 show the neutron energy-dependent activity ratios for some commonly used sensor combinations. In general, the steeper the activity ratio, the more sensitive the method is to the neutron energy.

9.1.4 Table 4 shows the energy resolutions of some specific sensor combinations for a 14.5 MeV neutron source. The $^{54}Fe(n,p)^{54}Mn/^{58}Ni(n,2n)^{57}Ni$ and $^{27}Al(n,\alpha)^{24}Na/^{58}Ni(n,2n)^{57}Ni$ are recommended sensor combinations due to their steep slope and their very 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 $^{63}Cu(n,\gamma)^{64}Cu$ 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 $^{55}Mn(n,\gamma)^{56}Mn$ 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 $^{23}Na(n,\gamma)^{24}Na$ contaminant reaction which affects the use of the $^{27}Al(n,\alpha)^{24}Na$ 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 E 720 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 during the irradiation, as explained in Guide E 720. 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 these two sensor activities to determine the neutron average energy.

TABLE 1 Cross Section Parameters for Some Useful Reactions

	Dosimetry Reactions	Target Nucleus					Product Nucleus			Reaction Notes
		Elemental Atomic Weight **(11, 12)**	Isotopic Atomic Number Abundance, % **(11, 12, 13)**	Cross Section Source		Cross Section Uncertainty Near 14-MeV, %	Half-Life **(11, 14)**	E_γ, keV **(11)**	Yield, %, γ per Reaction **(11)**	
				Library	Material Number					
1	$^{24}Mg(n,p)^{24}Na$	24.305	78.99	IRDF-90	1225	0.5	14.959 h	1368.598	100.0	...
								2753.995	99.944	...
2	$^{27}Al(n,p)^{27}Mg$	26.98154	100.0	IRDF-90	1325	15.6	9.462 m	843.757	73.0	A
								1014.429	29.1	...
3	$^{27}Al(n,\alpha)^{24}Na$	26.98154	100.0	IRDF-90	1325	0.4	14.959 h	1368.598	100.0	...
								2753.995	99.944	...
4	$^{32}S(n,p)^{32}P$	32.06	95.02	IRDF-90	1625	8.0	14.262 d	$<E_\beta> = 695$	100.0	B
5	$^{54}Fe(n,p)^{54}Mn$	55.847	5.9	ENDF/B-VI	2625	3.0	312.12 d	834.827	99.975	...
6	$^{56}Fe(n,p)^{56}Mn$	55.847	91.72	ENDF/B-VI	2631	2.1	2.5785 h	846.754	98.87	...
								1810.72	27.189	...
								2113.05	14.336	...
7	$^{58}Ni(n,p)^{58}Co$	58.6934	68.077	ENDF/B-VI	2825	18.6	70.82 d	810.791	99.5	...
							9.15 h (meta)	863.974	0.72	...
								1674.753	0.56	...
								24.889	0.0369	...
8	$^{58}Ni(n,2n)^{57}Ni$	58.6934	68.077	ENDF-B-VI	2825	2.4	35.65 h	1377.62	77.9	...
								1919.46	14.7	...
9	$^{63}Cu(n,2n)^{62}Cu$	63.546	69.17	IRDF-90	2925	1.5	9.74 m	1173.04	0.335	A,C
								875.74	0.147	...
10	$^{63}Cu(n,\alpha)^{60}Co$	63.546	69.17	ENDF/B-VI	2925	6.0	1925.5 d	1173.238	99.857	...
								1332.502	99.9831	...
							10.47 m (meta)	58.603	2.01	...
								826.33	0.0058	...
								1332.501	0.25	...
								2158.86	0.00088	...
11	$^{65}Cu(n,2n)^{64}Cu$	63.546	30.83	ENDF/B-VI	2931	2.8	12.701 h	1345.78	0.48	...
12	$^{64}Zn(n,p)^{64}Cu$	65.39	48.6	IRDF-90	3025	3.4	12.701 h	1345.78	0.48	...
13	$^{90}Zr(n,2n)^{89}Zr$	91.224	51.45	IRDF-90	4025	1.0	74.41 h	909.15	99.01	...
								1712.9	0.76	...
								1744.5	0.129	...
							4.18 m (meta)	587.70	89.5	...
								1507.4	6.04	
14	$^{93}Nb(n,2n)^{92m}Nb$	92.90638	100.0	IRDF-90	4125	1.5	10.15 d	934.53	99.0	D
								912.81	1.73	...
								1847.33	0.90	...

A Use of this reaction requires accurate timing but also provides high specific activity per neutron.
B The β emissions are counted to determine the activity.
C Use of 511 keV line risks high background signals from other positron emitters.
D The cross section is particularly flat near 14-MeV, insensitive to neutron energy, and hence suitable for the measurement of fluence.

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 **(17)** and in international intercomparisons **(18)**. 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.

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 E 720 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 a 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 E 720. 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.

11. Calculation of Neutron Energy

11.1 Form the ratio of the measured activities determined in 9.4 for the two sensor reactions chosen in 9.1. Refer to this ratio as R_{act}:

TABLE 2 Cross Section Near 14-MeV for Dosimetry Reactions

Energy (MeV)		Ratio						
		$^{24}Mg(n,p)^{24}Na$	$^{27}Al(n,p)^{27}Mg$	$^{27}Al(n,\alpha)^{24}Na$	$^{32}S(n,p)^{32}P$	$^{54}Fe(n,p)^{54}Mn$	$^{56}Fe(n,p)^{56}Mn$	$^{58}Ni(n,p)^{58}Co$
1	13.55	0.20894E + 00	0.76781E−01	0.12545E + 00	0.28764E + 00	0.37868E + 00	0.11683E + 00	0.42123E + 00
2	13.65	0.20581E + 00	0.75983E−01	0.12488E + 00	0.28009E + 00	0.37045E + 00	0.11676E + 00	0.41075E + 00
3	13.75	0.20055E + 00	0.75185E−01	0.12384E + 00	0.27253E + 00	0.36222E + 00	0.11670E + 00	0.40027E + 00
4	13.85	0.19314E + 00	0.74387E−01	0.12278E + 00	0.26498E + 00	0.35400E + 00	0.11628E + 00	0.38979E + 00
5	13.95	0.19116E + 00	0.73589E−01	0.12258E + 00	0.25743E + 00	0.34577E + 00	0.11551E + 00	0.37931E + 00
6	14.05	0.19460E + 00	0.72791E−01	0.12220E + 00	0.25069E + 00	0.33806E + 00	0.11474E + 00	0.36882E + 00
7	14.15	0.19642E + 00	0.71993E−01	0.12151E + 00	0.24476E + 00	0.33088E + 00	0.11398E + 00	0.35834E + 00
8	14.25	0.19661E + 00	0.71195E−01	0.12043E + 00	0.23883E + 00	0.32369E + 00	0.11306E + 00	0.34785E + 00
9	14.35	0.19508E + 00	0.70397E−01	0.11808E + 00	0.23290E + 00	0.31650E + 00	0.11200E + 00	0.33736E + 00
10	14.45	0.19184E + 00	0.69599E−01	0.11614E + 00	0.22697E + 00	0.30933E + 00	0.11094E + 00	0.32689E + 00
11	14.55	0.18751E + 00	0.68800E−01	0.11482E + 00	0.22268E + 00	0.30214E + 00	0.10988E + 00	0.31725E + 00
12	14.65	0.18211E + 00	0.68002E−01	0.11330E + 00	0.22001E + 00	0.29496E + 00	0.10871E + 00	0.30848E + 00
13	14.75	0.17714E + 00	0.67204E−01	0.11220E + 00	0.21735E + 00	0.28777E + 00	0.10743E + 00	0.29971E + 00
14	14.85	0.17259E + 00	0.66406E−01	0.11102E + 00	0.21469E + 00	0.28059E + 00	0.10615E + 00	0.29092E + 00
15	14.95	0.17054E + 00	0.65608E−01	0.10969E + 00	0.21202E + 00	0.27339E + 00	0.10487E + 00	0.28217E + 00
16	15.05	0.17097E + 00	0.64755E−01	0.10878E + 00	0.20836E + 00	0.26708E + 00	0.10333E + 00	0.27340E + 00
17	15.15	0.17140E + 00	0.63846E−01	0.10787E + 00	0.20370E + 00	0.26165E + 00	0.10151E + 00	0.26463E + 00
18	15.25	0.17138E + 00	0.62938E−01	0.10674E + 00	0.19905E + 00	0.25622E + 00	0.99689E−01	0.25586E + 00
19	15.35	0.16864E + 00	0.62029E−01	0.10508E + 00	0.19439E + 00	0.25080E + 00	0.97875E−01	0.24709E + 00
20	15.45	0.16546E + 00	0.61120E−01	0.10340E + 00	0.18973E + 00	0.24537E + 00	0.96058E−01	0.23832E + 00
21	15.55	0.16227E + 00	0.60212E−01	0.10171E + 00	0.18508E + 00	0.23994E + 00	0.94262E−01	0.22955E + 00

Energy (MeV)		Reaction						
		$^{58}Ni(n,2n)^{57}Ni$	$^{63}Cu(n,2n)^{62}Cu$	$^{63}Cu(n,\alpha)^{60}Co$	$^{65}Cu(n,2n)^{64}Cu$	$^{64}Zn(n,p)^{64}Cu$	$^{90}Zr(n,2n)^{89}Zr$	$^{93}Nb(n,2n)^{92m}Nb$
1	13.55	0.13385E−01	0.37647E + 00	0.44060E−01	0.83059E + 00	0.20144E + 00	0.44720E + 00	0.45691E + 00
2	13.65	0.15248E−01	0.39166E + 00	0.43980E−01	0.84498E + 00	0.19609E + 00	0.48245E + 00	0.45756E + 00
3	13.75	0.17482E−01	0.40670E + 00	0.43869E−01	0.85935E + 00	0.19074E + 00	0.51698E + 00	0.45807E + 00
4	13.85	0.19669E−01	0.42083E + 00	0.43575E−01	0.87373E + 00	0.18539E + 00	0.55079E + 00	0.45842E + 00
5	13.95	0.21811E−01	0.43480E + 00	0.43250E−01	0.88812E + 00	0.18004E + 00	0.58251E + 00	0.45877E + 00
6	14.05	0.23808E−01	0.44878E + 00	0.42752E−01	0.90066E + 00	0.17469E + 00	0.61213E + 00	0.45913E + 00
7	14.15	0.25660E−01	0.46275E + 00	0.42082E−01	0.91136E + 00	0.16934E + 00	0.64245E + 00	0.45948E + 00
8	14.25	0.27809E−01	0.47681E + 00	0.41433E−01	0.92206E + 00	0.16460E + 00	0.67346E + 00	0.45978E + 00
9	14.35	0.30257E−01	0.49138E + 00	0.40919E−01	0.93276E + 00	0.16348E + 00	0.70327E + 00	0.45980E + 00
10	14.45	0.32818E−01	0.50604E + 00	0.40429E−01	0.94344E + 00	0.16297E + 00	0.73186E + 00	0.45977E + 00
11	14.55	0.35502E−01	0.52069E + 00	0.39909E−01	0.95218E + 00	0.16246E + 00	0.75760E + 00	0.45973E + 00
12	14.65	0.37466E−01	0.53535E + 00	0.39359E−01	0.95891E + 00	0.16157E + 00	0.78048E + 00	0.45970E + 00
13	14.75	0.38696E−01	0.54968E + 00	0.38807E−01	0.96566E + 00	0.15972E + 00	0.80093E + 00	0.45958E + 00
14	14.85	0.39979E−01	0.56207E + 00	0.38243E−01	0.97238E + 00	0.15788E + 00	0.81894E + 00	0.45889E + 00
15	14.95	0.41567E−01	0.57415E + 00	0.37675E−01	0.97912E + 00	0.15716E + 00	0.83770E + 00	0.45811E + 00
16	15.05	0.43211E−01	0.58622E + 00	0.37072E−01	0.98554E + 00	0.15688E + 00	0.85720E + 00	0.45733E + 00
17	15.15	0.44855E−00	0.59830E + 00	0.36432E−01	0.99165E + 00	0.15661E + 00	0.87671E + 00	0.45656E + 00
18	15.25	0.46499E−01	0.61088E + 00	0.35786E−01	0.99777E + 00	0.15633E + 00	0.89596E + 00	0.45578E + 00
19	15.35	0.48144E−01	0.62654E + 00	0.35121E−01	0.10039E + 01	0.15606E + 00	0.91372E + 00	0.45500E + 00
20	15.45	0.49788E−01	0.64272E + 00	0.34450E−01	0.10100E + 01	0.15579E + 00	0.93123E + 00	0.45422E + 00
21	15.55	0.51303E−01	0.65889E + 00	0.33771E−01	0.10148E + 01	0.15385E + 00	0.94874E + 00	0.45293E + 00

$$R_{act} = \frac{A_1}{A_2} \quad (2)$$

where:

A_1 = the activity from the first reaction, and

A_2 = the activity from the second reaction.

In this test method the numerical subscripts 1 and 2 will occur on various quantities. Unless there is an explicit definition, these subscripts refer to the two reactions chosen in 9.1.

11.2 Use the reaction half-lives from Table 1 to convert the activity ratio into a cross section ratio, R_{xsec}.

$$R_{xsec} = R_{act} \times \frac{\tau_1}{\tau_2} = \frac{A_1}{A_2} \times \frac{\tau_1}{\tau_2} \quad (3)$$

11.3 Determine the energy that corresponds to the cross section ratio, R_{xsec}.

11.3.1 Use the data in Table 3 and linear-linear interpolation to determine the neutron energy that corresponds to this cross section ratio. Refer to this energy as E_{eff}. This energy represents an average energy for the neutrons. The neutron energy from a thick target is not truly monoenergetic.

11.3.2 If the dosimetry-quality sensor chosen is not addressed in Table 3, then the experimenter must construct data similar to that presented in Table 3 from a dosimetry-quality cross section evaluation for the reactions used. A necessary component of dosimetry-quality cross sections is the presence of covariance data. The methodology used in this test method makes use of the cross section uncertainty. This test method requires the existence of a complete covariance matrix even though the off-diagonal covariance elements are not utilized. This requirement is made to ensure that a dosimetry-quality cross section evaluation is used. Single point cross section values with uncertainties are not sufficient since the energy-dependent slope of the cross section must be known. All high-quality cross section determinations that address more than a single energy have been observed to provide a state-of-the-art statistical analysis that includes a covariance matrix. The International Reactor Dosimetry File (IRDF-90) **(15)** or a cross section in the SNLRML compendium **(16)** are recommended as sources for cross section data.