



Standard Test Method for Measuring Reaction Rates and Fast-Neutron Fluences by Radioactivation of Sulfur-32¹

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

1. Scope

1.1 This test method describes procedures for measuring reaction rates and fast-neutron fluences by the activation reaction $^{32}\text{S}(n,p)^{32}\text{P}$.

1.2 This activation reaction is useful for measuring neutrons with energies above approximately 3 MeV.

1.3 With suitable techniques, fission-neutron fluences from about 5×10^8 to 10^{16} n/cm² can be measured.

1.4 Detailed procedures for other fast-neutron detectors are described in Practice E 261.

1.5 *This standard does not purport to address all of the safety problems, 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 Dosimetry²

E 181 Test Methods for Detector Calibration and Analysis of Radionuclides²

E 261 Practice for Determining Neutron Fluence Rate, Fluence, and Spectra by Radioactivation Techniques²

E 844 Guide for Sensor Set Design and Irradiation for Reactor Surveillance, E706(IIC)²

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

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

3. Terminology

3.1 Definitions:

3.1.1 Refer to Terminology E 170.

4. Summary of Test Method

4.1 Elemental sulfur or a sulfur-bearing compound is irradiated in a neutron field, producing radioactive ^{32}P by means of the $^{32}\text{S}(n,p)^{32}\text{P}$ activation reaction.

4.2 The beta particles emitted by the radioactive decay of ^{32}P are counted by techniques described in Methods E 181 and the reaction rate, as defined in Practice E 261, is calculated from the decay rate and irradiation conditions.

4.3 The neutron fluence above 3 MeV can then be calculated from the spectral-averaged neutron activation cross section, $\bar{\sigma}$, as defined in Practice E 261.

5. Significance and Use

5.1 Refer to Guide E 844 for the selection, irradiation, and quality control of neutron dosimeters.

5.2 Refer to Practice E 261 for a general discussion of the determination of fast-neutron fluence and fluence rate with threshold detectors.

5.3 The activation reaction produces ^{32}P , which decays by the emission of a single beta particle in 100 % of the decays, and which emits no gamma rays. The half life of ^{32}P is 14.262 (14)³ days (**1**)⁴ and the maximum beta energy is 1710 keV (**2**).

5.4 Elemental sulfur is readily available in pure form and any trace contaminants present do not produce significant amounts of radioactivity. Natural sulfur, however, is composed of ^{32}S (95.02 % (9)), ^{34}S (4.21 % (8)) 1, and trace amounts of other sulfur isotopes. The presence of these other isotopes leads to several competing reactions that can interfere with the counting of the 1710-keV beta particle. This interference can usually be eliminated by the use of appropriate techniques, as discussed in Section 8.

6. Apparatus

6.1 Since only beta particles of ^{32}P are counted, proportional counters or scintillation detectors can be used. Because of the high resolving time associated with Geiger-Mueller counters,

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² Annual Book of ASTM Standards, Vol 12.02.

³ The non-bolface number in parentheses after the nuclear data indicates the uncertainty in the last significant digit of the preceding number. For example, 8.1 s (5) means 8.1 ± 0.5 seconds.

⁴ The **bolface** numbers in parentheses refer to the list of references at the end of this test method.

their use is not recommended. They can be used only with relatively low counting rates, and then only if reliable corrections for coincidence losses are applied.

6.2 Refer to Methods E 181 for preparation of apparatus and counting procedures.

7. Materials and Manufacture

7.1 Commercially available sublimed flowers of sulfur are inexpensive and sufficiently pure for normal usage. Sulfur can be used directly as a powder or pressed into pellets. Sulfur pellets are normally made at least 3 mm thick in order to obtain maximum counting sensitivity independent of small variations in pellet mass. A 0.8 g/cm² pellet can be considered infinitely thick for the most energetic beta particle from ³²P (see Table 1). Due to the relatively long half-life of ³²P, it may not be practical to use a pellet more than once. A period of at least one year is recommended between uses. However, see 8.2 regarding long-lived interfering reaction products.

7.2 Where temperatures approaching the melting point of sulfur are encountered (113°C), sulfur-bearing compounds such as ammonium sulfate (NH₄)₂SO₄, lithium sulfate Li₂SO₄, or magnesium sulfate MgSO₄ can be used. These are suitable for temperatures up to 250, 850, and 1000°C, respectively. The reduced sensitivity of these compounds offers no disadvantage since high temperatures are usually associated with a high-neutron fluence rate. The sulfur content by weight of (NH₄)₂SO₄ is 24 %, of Li₂SO₄ is 29.2 %, and of MgSO₄ is 26.6 %.

7.3 The isotopic abundance of ³²S in natural sulfur is 95.02 ± 0.09 atom % (1).

8. Sample Preparation and Irradiation

8.1 Place sulfur in pellet or powdered form in a uniform fast-neutron flux for a predetermined period of time. Record the beginning and end of the irradiation period.

8.2 Table 2 lists competing reaction products that must be eliminated from the counting. Those resulting from thermal-neutron capture, that is, ³³P, ³⁵S, and ³⁷S, can be reduced by the irradiation of the sulfur inside 1 mm-thick cadmium shields.

This should be done whenever possible in thermal-neutron environments. Those reaction products having relatively short half-lives, that is, ³¹S, ³⁴P, ³¹Si, and ³⁷S, can be eliminated by a waiting period before the counting is started. A delay of 24 h is sufficient for the longest lived of these, although shorter delays are possible depending on the degree of thermalization of the neutron field. Finally, those with relatively low beta particle energies, that is, ³³P and ³⁵S, can be eliminated by the inclusion of a 70-mg/cm² aluminum absorber in front of the detector. For particularly long decay times, an absorber must be used because the ³⁵S becomes dominant. Note that the use of an internal (windowless) detector maximizes the interference in counting from ³⁵S.

8.3 Irradiated sulfur can be counted directly, or may be burned to increase the efficiency of the counting system. Dilution may be used to reduce counting system efficiency for measurements of high neutron fluences.

8.4 Burning the sulfur leaves a residue of ³²P that can be counted without absorption of the beta particles in the sulfur pellet. Place the sulfur in an aluminum planchet on a hot plate until the sulfur melts and turns to a dark amber color. At this point the liquid gives off sulfur fumes. Ignite the fumes by bringing a flame close to the dish, and allow the sulfur to burn out completely. In order to reduce the sputtering that can lead to variations in the amount of ³²P remaining on the planchet, the hot plate must be only as hot as necessary to melt the sulfur. In addition, air flow to the burning sulfur must be controlled, such as by the placement of a chimney around the sulfur. Count the residue remaining on the dish for beta activity.

NOTE 1—The fumes given off by the burning sulfur are toxic. Burning should be done under a ventilating hood.

8.5 An alternative to burning is sublimation of the sulfur under a heat lamp. Removal of the sulfur is very gradual, and there is no loss of ³²P from sputtering.

8.6 Counting of dilute samples is useful for measuring high neutron fluences, although it is applicable to virtually all irradiation conditions. Use lithium sulfate, reagent grade or better, as the target material because of its high melting point (860°C), good solubility in water, and minimum production of undesirable activation products. Prepare a dry powder by spreading about 10 g of Li₂SO₄ in a weighing bottle and place in a drying oven for 24 h at 150°C. Place the dried Li₂SO₄ in a desiccator for cooling and storage. Prepare a phosphorus carrier solution by dissolving 21.3 g of (NH₄)₂HPO₄ in water to make 1 L of solution. Prepare a Li₂SO₄ sample for irradiation by placing about 150 mg of material in an air-tight aluminum capsule or other suitable container. Following the irradiation, accurately weigh a sample of about 100 mg and dissolve in 5 mL of phosphorus carrier solution to minimize adsorption of ³²P on the glass container. A drop of concentrated HCl may be used to speed solution of the sample. Place the solution in a volumetric flask and add additional phosphorus carrier solution to bring the total volume to 100 mL. Prepare a sample for counting by pipetting 0.050 mL of the ³²P solution onto a standard planchet and evaporating in air to dryness. Counting procedures and calculations are the same as in other methods with the exception that an aliquot factor of 2000 must be

TABLE 1 Sulfur Counting Rate Versus Mass for a Pellet of 25.4-mm Diameter

Sample Mass, g	Relative Counting Rate
0.4	0.46
0.6	0.58
0.8	0.66
1.0	0.73
1.2	0.78
1.4	0.82
1.6	0.86
1.8	0.89
2.0	0.91
2.2	0.93
2.4	0.94
2.6	0.95
2.8	0.96
3.0	0.97
3.2	0.98
3.4	0.99
3.6	0.99
3.8	1.0
4.0	1.0

TABLE 2 Neutron-induced Reactions in Sulfur Giving Radioactive Products

Reaction	Cross Section		Cross Section (mb)			Product Half-life (1)	Maximum Energy of Product Beta (MeV) (2)	Average Energy of Product Beta (MeV) (2)	Isotopic Abundance of Target (%) (1)
	Library	Material ID	Thermal ^A	²³⁵ U Thermal Fission	Fast ^B ²⁵² Cf Fission				
1. ³² S(n,p) ³² P	GLUCS-93 ^C	1625	...	64.69	70.44	14.262 d (14)	1.7104	0.6949	95.02 (9)
2. ³² S(n,2n) ³¹ S	JENDL-3 ^D	3161	...	7.742 × 10 ⁻⁶	2.5 × 10 ⁻⁵	2.572 s (13)	5.3956 (β +)	1.9975 (β +)	95.02 (9)
3. ³³ S(n,p) ³³ P	JENDL-3 ^D	3162	1.6	57.46	58.77	25.34 d (12)	0.2485	0.0764	0.75 (1)
4. ³⁴ S(n,p) ³⁴ P	JENDL-3 ^D	3163	...	0.8001	1.079	12.43 s (8)	5.3743	2.3108	4.21 (8)
5. ³⁴ S(n,α) ³¹ Si	JENDL-3 ^D	3163	...	3.281	4.064	157.3 m (3)	1.4908	0.59523	4.21 (8)
6. ³⁴ S(n,γ) ³⁵ S	JENDL-3 ^D	3163	226	0.2749	0.2705	87.51 d (12)	0.16684	0.04863	4.21 (8)
7. ³⁶ S(n,γ) ³⁷ S	JENDL-3 ^D	3164	151	0.2511	0.2509	5.05 m (2)	4.86516	0.800418	0.02 (1)

^AThe thermal cross section corresponds to neutrons with a velocity of 2200 m/s or an energy of 0.0253 eV.

^BThe fast cross section corresponds to the spectrum-averaged cross section from the ENDF/B-VI (MAT=9228, MF=5, MT=18) ²³⁵U thermal fission spectrum (5,6) and the ENDF/B-VI (MAT=9861, MF=5, MT=18) ²⁵²Cf spontaneous fission spectrum (4-6).

^CCross section produced for the 1993 GLUCS library (7) and is similar to that in the IRDF-90 library (8).

^DThe JENDL-3 (9) sulfur isotopes were adopted in the latest JEF 2.2 cross section (10) compilations. The ENDF/B-VI library (5) does not include the individual sulfur isotopic cross sections.

introduced for the 0.050-mL sample removed from the 100-mL flask.

9. Calibration

9.1 Calibration is achieved by irradiation of sulfur in a fast-neutron field of known spectrum and intensity, and measuring the resulting ³²P activity to determine a counting system efficiency. This calibration is specific for a given detector system, counting geometry, and sulfur pellet size and mass or sample preparation. It is, however, valid for subsequent use in measuring activities in any arbitrary spectrum, and therefore, may be used with activation data from other foils in determining neutron energy spectra as described in Practice E 944.

9.2 ²³⁵U fission and ²⁵²Cf spontaneous fission neutron sources of known source strength are available for direct free-field calibrations (3).

9.3 Once a sulfur counting system is calibrated, it must be monitored to ensure that the calibration remains valid. There are several isotopes that can be used as reference standards for this monitoring. One is ²³⁴Pa, having a maximum beta energy of about 2000 keV, comparable to the 1710-keV beta from ³²P. It is obtained as a daughter of ²³⁸U, that can be dispersed as a powder in plastic granules and formed to the shape of a standard pellet. The concentration of ²³⁸U can be varied to obtain the desired counting rate. Uranium alpha particles can be prevented from reaching the detector by use of a 7-mg/cm² absorber. Another useful isotope is ²¹⁰Bi, that produces beta particles having a maximum energy of 1161 keV. It is obtained as a daughter of ²¹⁰Pb, and sources are commercially available.

10. Activity and Fluence by Detector Efficiency Method

10.1 Using a sulfur sample irradiated in a calibration neutron field, determine the efficiency, ϵ , for the detector system:

$$\epsilon = \frac{C f_{\tau} \exp[\lambda t_d] \lambda t_i}{N \bar{\sigma}_s \Phi (1 - \exp[-\lambda t_c]) (1 - \exp[-\lambda t_i])} \quad (1)$$

where:

C = counts recorded in detector, less background,

f_{τ} = correction for coincidence losses, if needed,

λ = ³²P decay constant, = 5.625 × 10⁻⁷ s⁻¹,

t_d = decay time, s,

t_c = count time, s,

t_i = duration of irradiation, s,

N = number of ³²S atoms in pellet,

$\bar{\sigma}_s$ = spectrum-averaged cross section for ³²S in the calibration neutron field, cm² = 10²⁴ b, and

Φ = neutron fluence, n/cm².

10.1.1 Fig. 1 shows a plot of sulfur cross section as a function of energy. Fig. 2 shows a plot of the uncertainty in the sulfur cross section as a function of energy. (See also Guide E 1018.) The spectrum-averaged cross section for ²⁵²Cf fission neutrons is about 70 mb, and for ²³⁵U fission is about 63 mb. (See Table 2.)

10.1.2 The correction for coincidence losses, f_{τ} , is a function of the particular counting system, and may be already accounted for by the system electronics if “live time” is used (see Methods E 181). Coincidence loss corrections can be large, especially when Geiger-Mueller counters are used.

NOTE 2—Because of β self-absorption in counting thick pellets intact, detection efficiency is not sensitive to small variations in pellet mass but is rather a function of the pellet dimensions. The detection efficiency should be determined for each different pellet size that is to be used. The value of N in Eq 1 can be taken to be the arithmetic mean over a number of pellets. Whether or not this is adequate depends on the uniformity of the pellets and the desired measurement uncertainties.

NOTE 3—When the calibration is performed using a point source such as ²⁵²Cf, a correction should be made for fluence gradients and neutron scattering in the sulfur pellet. An estimate of the gradient effect is complicated by the fact that the detector is thick with respect to the range of the beta particles. If the calibration fluence is averaged over the front face of the pellet, the gradient correction is typically about [16/ r (cm)] % for a 2.54-cm diameter pellet, where r is the distance from the calibration source to the front face of the pellet. Effects of neutron scattering in the detector are typically a few tenths of a percent and can generally be

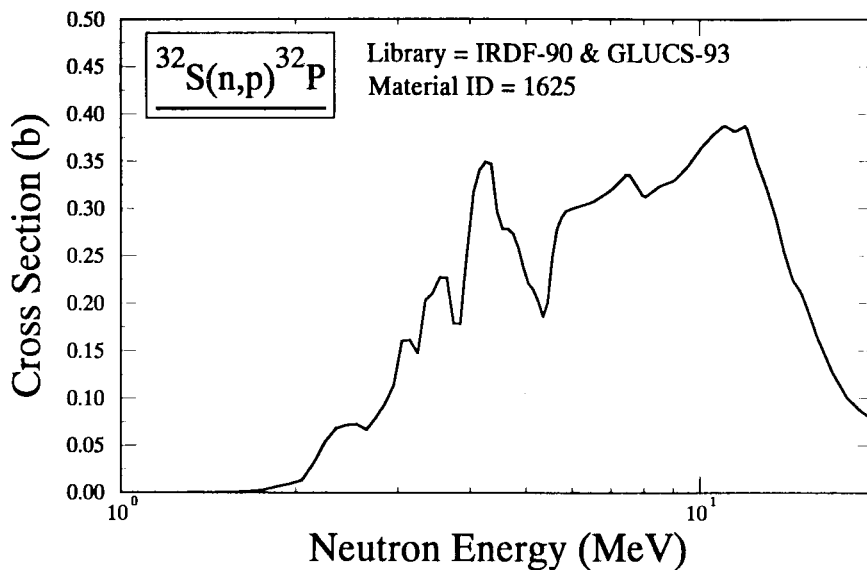


FIG. 1 Cross Section Versus Energy for the $^{32}\text{S}(n,p)^{32}\text{P}$ Reaction

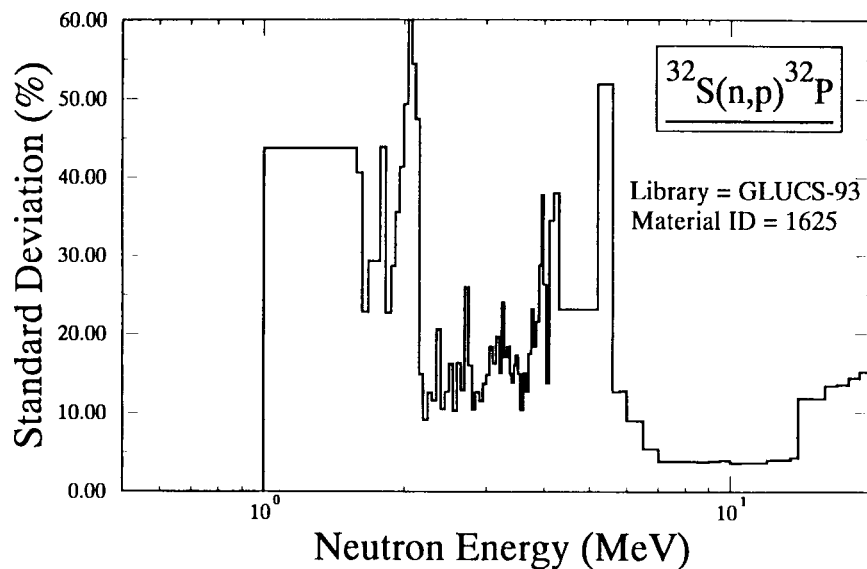


FIG. 2 Energy-dependence of Uncertainty in $^{32}\text{S}(n,p)^{32}\text{P}$ Cross Section

assumed to be the same in both calibration and measurement fields (11).

10.2 Count the beta particles from the sulfur pellet or prepared sample in a calibrated detector system, and determine the ^{32}P activity:

$$A_o = \frac{C_f \tau \exp[\lambda t_d] \lambda}{\epsilon(1 - \exp[-\lambda t_c])} \quad (2)$$

where:

A_o = pellet activity adjusted to the end of the irradiation, and

ϵ = detector system efficiency previously determined as described in 10.1.

10.3 The saturation activity, A_s , as defined in Practice E 261, is related to A_o by the following equation:

$$A_s = \frac{A_o}{(1 - \exp[-\lambda t_i])} \quad (3)$$

10.4 The reaction rate, R_s , is determined from the ^{32}P activity as follows:

$$R_s = A_s/N \quad (4)$$

10.5 Determination of the neutron fluence requires knowledge of the spectrum average cross section for the specific neutron field. The neutron fluence is determined from the ^{32}P activity:

$$\Phi = \frac{A_o t_i}{N \bar{\sigma}_x (1 - \exp[-\lambda t_i])} = \frac{A_s t_i}{N \bar{\sigma}_x} \quad (5)$$

where:

$\bar{\sigma}_x$ = spectrum averaged cross section for $^{32}\text{S}(n,p)$ in the neutron field being measured, $\text{cm}^2 = 10^{24}$ b.

or for a pulse rather than an extended irradiation:

$$\Phi = \frac{A_o}{N\bar{\sigma}_x\lambda} \quad (6)$$

The value of $\bar{\sigma}_x$ used in Eq 5, Eq 6 must be calculated using the same set of cross sections as is used to calculate the value of $\bar{\sigma}_s$ in Eq 1.

NOTE 4—The GLUCS-93 cross-section library (7) is recommended for sulfur (material ID 1625). The IRDF-90 cross-section library (8) contains essentially the same cross sections for sulfur, and may also be used. However, the covariance data in the IRDF-90 library is incorrect.⁵

NOTE 5—If the neutron fluence rate varies during the irradiation, adjustments to the calculations of saturated activities and fluences should be made in accordance with procedures recommended in Practice E 261.

11. Fluence by Direct Comparison Method

11.1 If a value for sulfur activity is not required, the neutron fluence in a test field can be determined by direct comparison between the dosimeter response to the neutron field being measured and that for a calibration neutron field, usually a $n^{252}\text{Cf}$ or ^{235}U fission spectrum.

11.1.1 Determine the corrected detector count rate, C_s , for a sulfur sample irradiated in a calibration neutron field:

$$C_s = \frac{C_f \exp[\lambda t_d] \lambda}{(1 - \exp[-\lambda t_c])} \quad (7)$$

11.1.2 Similarly, determine the corrected detector count rate, C_x , for a sulfur sample irradiated in the test neutron field. Then the neutron fluence, Φ_x , in the test field is given by:

$$\Phi_x = \frac{C_x}{C_s} \cdot \frac{\bar{\sigma}_s}{\bar{\sigma}_x} \cdot \Phi_s \quad (8)$$

11.1.3 The values for $\bar{\sigma}_x$ and $\bar{\sigma}_s$ must both be calculated using the same set of cross sections. The effect on the uncertainty in the measured fluence from uncertainties in the spectrum-averaged cross sections is reduced to the extent that the calibration and measured spectra are similar and known.

11.2 Fluences are often expressed as fluences greater than some reference energy, E_0 :

$$\Phi_x(>E_0) = \frac{C_x}{C_s} \cdot \frac{\bar{\sigma}_s(>E_0)}{\bar{\sigma}_x(>E_0)} \cdot \Phi_s(>E_0) \quad (9)$$

where:

$\Phi_x(>E_0)$ = neutron fluence above energy E_0 for the test irradiation,

$\Phi_s(>E_0)$ = neutron fluence above energy E_0 for the calibration irradiation,

$\bar{\sigma}_x(>E_0)$ = spectrum-averaged cross section above energy E_0 for the test field,

$\bar{\sigma}_s(>E_0)$ = spectrum-averaged cross section above energy E_0 for the test field,

and where $\bar{\sigma}(>E_0)$ is defined by:

$$\bar{\sigma}(>E_0) = \bar{\sigma} \cdot \frac{\Phi}{\Phi(>E_0)} = \bar{\sigma} \cdot \frac{\int_0^\infty \Phi(E)dE}{\int_{E_0}^\infty \Phi(E)dE} \quad (10)$$

11.2.1 For sulfur, E_0 is traditionally chosen to be 3 MeV. This value is close to the true threshold in the sulfur cross section, and results in values of $\bar{\sigma}(>E_0)$ that differ only slightly for various fission spectra.

12. Report

12.1 Practice E 261 describes reporting requirements.

13. Precision and Bias

NOTE 6—Measurement uncertainty is described by a precision and bias statement in this standard. Another acceptable approach is to use Type A and B uncertainty components (15),(16). This type A/B uncertainty specification is now used in International Organization for Standardization (ISO) standards and this approach can be expected to play a more prominent role in future uncertainty analyses.

13.1 Precision (Random Uncertainties):

13.1.1 Factors affecting the precision of a measurement include variations in sulfur mass or counting geometry, variations in ^{32}P losses during burning or dilutions, fluctuations in detector system efficiency or background, and counting statistics. These uncertainties can be determined for a given measurement system by repetitive measurements in a fixed neutron field, and by use of a routine check as discussed in 9.3. With care, precision better than $\pm 2\%$ (1S) can be obtained.

13.2 Bias (Systematic Uncertainties):

13.2.1 Factors affecting the bias in an activity or fluence measurement include calibration errors, fluence gradients and neutron scattering in the sulfur, coincidence losses during counting, uncertainties in nuclear data constants, and contributions from radioisotopes other than ^{32}P . If handled properly, these factors should not contribute more than $\pm 3\%$ (1S) to the uncertainty of a measurement.

13.2.2 Uncertainty in an activity measurement also depends directly on the uncertainty in $\bar{\sigma}_s$. Energy-dependent uncertainties in the sulfur cross section can be very large, as shown in Fig. 2, and can result in uncertainties in $\bar{\sigma}_s$ of $\pm 8\text{--}10\%$ (1S).

13.2.3 The $^{32}\text{S}(n,p)^{32}\text{P}$ cross section has an activity measurement uncertainty of 3.5% (12) in the ^{252}Cf standard spontaneous fission field with a calculated-to-experiment (C/E) cross section of $0.9699 \pm 5.13\%$ (14). In the ^{235}U standard thermal fission field the activity measurement uncertainty is 5.5% (13) with a C/E of $0.9684 \pm 8.23\%$ (14).

13.2.4 Uncertainties in $\bar{\sigma}_s$ and $\bar{\sigma}_x$ affect only slightly the uncertainty in a fluence measurement, because only their ratio is involved. The magnitude of this uncertainty depends on the similarity between the calibration and test neutron spectra, and on how well the spectra are known. However, this should not exceed $\pm 2\%$ (1S) for fission neutron fields. The two methods described for deciding neutron fluence are equivalent, and therefore result in the same uncertainty.

14. Keywords

14.1 activation analysis; beta counting; fast-neutron fluences; neutron measurement; reaction rates; sulfur

⁵ D. M. Hetrick, Private communication to P. J. Griffin, Feb 5, 1993.



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