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INTERNATIONAL **ISO/DIS**
STANDARD **8529-1**

~~**Reference Neutron Radiations**~~

Part 1:

Characteristics and Methods of Production

Rayonnements neutroniques de reference –

Partie 1: Caractéristiques et méthodes de production

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Foreword

ISO (the International Organization for Standardization) is a worldwide federation of national standards bodies (ISO member bodies). The work of preparing International Standards is normally carried out through ISO technical committees. Each member body interested in a subject for which a technical committee has been established has the right to be represented on that committee. International organizations, governmental and non-governmental, in liason with ISO, also take part in the work. ISO collaborates closely with the International Electrotechnical Commission (IEC) on all matters of electrotechnical standardization.

Draft International Standards adopted by the technical committees are circulated to the member bodies for voting. Publication as an International Standard requires approval by at least 75% of the member bodies casting a vote.

International Standard ISO 8529-1 was prepared by Technical Committee ISO/TC85, *Nuclear Energy*, Subcommittee SC 2, *Radiation Protection*.

ISO 8529 consists of the following parts, under the general title *Reference neutron radiations*

- *Part 1: Characteristics and Methods of Production*
- *Part 2: Calibration fundamentals related to the basic quantities characterizing the radiation field*
- *Part 3: Calibration of area and personal dosimeters and determination of their response as a function of neutron energy and angle of incidence*

Introduction

This standard is the first of a set of three standards concerning the calibration of dosimeters and doserate meters for neutron radiation for protection purposes. It describes the characteristics and methods of production of the reference neutron radiations to be used for calibrations. The second standard, ISO 8529-2, describes fundamentals related to the physical quantities characterising the radiation field and calibration procedures in general terms, with emphasis on active doserate meters and the use of radionuclide sources. The third standard, ISO 8529-3, deals with dosimeters for area and individual monitoring, describing the respective procedures for calibrating and determining the response in terms of the ICRU operational quantities. Conversion coefficients for converting neutron fluence into these operational quantities are provided in 8529-3. ISO 8529-1 supersedes ISO 8529:1989

Reference neutron radiations –

Part 1:

Characteristics and Methods of Production

1 Scope

This International Standard specifies the neutron reference radiations, in the energy range from thermal up to 20 MeV, for calibrating neutron-measuring devices used for radiation protection purposes and for determining their response as a function of neutron energy. Reference radiations are given for neutron fluence rates of up to $1 \cdot 10^9 \text{ m}^{-2} \cdot \text{s}^{-1}$, corresponding, at a neutron energy of 1 MeV, to dose equivalent rates of up to $100 \text{ mSv} \cdot \text{h}^{-1}$.

This International Standard is concerned only with the methods of producing and characterizing the neutron reference radiations. The procedures for applying these radiations for calibrations are described in ISO 8529-2 and 8529-3.

The reference radiations specified are the following:

- neutrons from radionuclide sources, including neutrons from sources in a moderator;
- neutrons produced by nuclear reactions with charged particles from accelerators
- neutrons from reactors.

In view of the methods of production and use of them, these reference radiations are divided, for the purposes of this International Standard, into two separate sections:

— In Section 4, radionuclide neutron sources with wide spectra are specified for the calibration of neutron measuring devices. These sources shall be used by laboratories engaged in the routine calibration of neutron measuring devices, the particular design of which has already been type tested.

— In Section 5, accelerator-produced monoenergetic neutrons and reactor-produced neutrons with wide or quasi monoenergetic spectra are specified for determining the response of neutron measuring devices as a function of neutron energy. Since these reference radiations are produced at specialised and well equipped laboratories, only the minimum of experimental detail is given.

For the conversion of neutron fluence into the quantities recommended for radiation protection purposes, conversion coefficients have been calculated based on the spectra presented in Annex A and using the fluence-to-dose equivalent conversion coefficients as a function of neutron energy as given in ICRP Publication 74 and ICRU Report 57.

2 Normative references

The following standards contain provisions which, through reference in this text, constitute provisions of this International Standard. At the time of publication, the editions indicated were valid. All standards are subject to revision, and parties to agreements based on this International Standard are encouraged to investigate the possibility of applying the most recent editions of the

standards listed below. Members of IEC and ISO maintain registers of currently valid International Standards.

ISO 2919: 1999, *Sealed radioactive sources—General requirements and classification*.

ISO 8529-2: To be published, *Reference neutron radiations: calibration fundamentals related to the basic quantities characterizing the radiation field*.

ISO 8529-3: 1998, *Reference neutron radiations: calibration of area and personal dosimeters and the determination of their response as a function of neutron energy*.

ICRP Publication 60, *1990 Recommendations of the International Commission on Radiological Protection*, (Annals of the ICRP Vol. 21 No. 2).

ICRP Publication 74, *Conversion Coefficients for use in Radiological Protection against External Radiation*, (Annals of the ICRP Vol. 26 No.3/4 1996).

ICRU Report 33, *Radiation Quantities and Units*, 1980 edition.

ICRU Report 51, *Quantities and Units in Radiation Protection*. 1993 edition.

ICRU Report 57, *Conversion Coefficients for use in Radiological Protection Against External Radiation*, 1998 edition.

3 Definitions

For the purpose of this International Standard, the definitions follow the recommendations of ICRU Report 51 and the earlier ICRU Report 33.

3.1 quantities and units

3.1.1

neutron fluence

Φ

quotient of dN by da , where dN is the number of neutrons incident on a sphere of cross-sectional area da :

$$\Phi = \frac{dN}{da}$$

NOTE The unit of the neutron fluence is m^{-2} ; a frequently used unit is cm^{-2} .

3.1.2

neutron fluence rate; neutron flux density

φ

quotient of $d\Phi$ by dt , where $d\Phi$ is the increment of neutron fluence (see 3.1.1) in the time interval dt .

$$\varphi = \frac{d\Phi}{dt} = \frac{d^2N}{da \cdot dt}$$

NOTE The unit of the neutron fluence rate is $m^{-2} \cdot s^{-1}$.

3.1.3

spectral neutron fluence; energy distribution of the neutron fluence

Φ_E

quotient of $d\Phi$ by dE , where $d\Phi$ is the increment of neutron fluence in the energy interval between E and $E+dE$:

$$\Phi_E = \frac{d\Phi}{dE}$$

NOTE The unit of the spectral neutron fluence is $\text{m}^{-2} \cdot \text{J}^{-1}$; a frequently used unit is $\text{cm}^{-2} \cdot \text{eV}^{-1}$.

3.1.4

spectral neutron fluence rate; spectral neutron flux density

 φ_E

quotient of $d\Phi_E$ by dt , where $d\Phi_E$ is the increment of spectral neutron fluence in the time interval dt :

$$\varphi_E = \frac{d\Phi_E}{dt} = \frac{d^2\Phi}{dE \cdot dt}$$

NOTE The unit for the spectral neutron fluence rate is $\text{m}^{-2} \cdot \text{s}^{-1} \cdot \text{J}^{-1}$; a frequently used unit is $\text{cm}^{-2} \cdot \text{s}^{-1} \cdot \text{eV}^{-1}$.

3.1.5

absorbed dose

 D

quotient of $d\bar{\varepsilon}$ by dm , where $d\bar{\varepsilon}$ is the mean energy imparted by ionising radiation to matter of mass dm :

$$D = \frac{d\bar{\varepsilon}}{dm}$$

NOTE The unit of the absorbed dose is $\text{J} \cdot \text{kg}^{-1}$ with the special name gray (Gy).

3.1.6

dose equivalent

 H

product of Q and D at a point in tissue, where D is the absorbed dose and Q is the quality factor at that point:

$$H = Q \cdot D$$

NOTE The unit for the dose equivalent is $\text{J} \cdot \text{kg}^{-1}$ with the special name sievert (Sv).

3.1.7

dose equivalent rate

 \dot{H}

quotient of dH by dt , where dH is the increment of dose equivalent in the time interval dt :

$$\dot{H} = \frac{dH}{dt}$$

NOTE The unit for the dose equivalent rate is $\text{J} \cdot \text{kg}^{-1} \cdot \text{s}^{-1}$ with the special name sievert per second ($\text{Sv} \cdot \text{s}^{-1}$).

3.1.8

neutron fluence-to-dose equivalent conversion coefficient

 h_ϕ

quotient of the neutron dose equivalent, H , and the neutron fluence, Φ , at a point in the radiation field, undisturbed by the irradiated object:

$$h_\phi = H / \Phi$$

NOTE Any statement of a fluence-to-dose equivalent conversion coefficient requires the statement of the type of dose equivalent, e.g. ambient dose equivalent, personal dose equivalent. Their specific definitions and respective conversion coefficients are given in ISO 8529-3.

3.1.9

activity of an amount of radioactive nuclide in a particular energy state at a given time

A

quotient of dN^+ by dt , where dN^+ is the expectation value of the number of spontaneous nuclear transitions from that energy state in the time interval dt :

$$A = \frac{dN^+}{dt}$$

NOTE The unit of the activity is s^{-1} with the special name Becquerel (Bq).

3.1.10

neutron source strength of a neutron source at a given time

B

quotient of dN^* by dt , where dN^* is the expectation value of the number of neutrons emitted by the source in the time interval dt :

$$B = \frac{dN^*}{dt}$$

NOTE The unit of the source strength is s^{-1} .

3.1.11

angular source strength

B_Ω

quotient of dB by $d\Omega$, where dB is the number of neutrons per unit time propagating in a specified direction within the solid angle $d\Omega$:

$$B_\Omega = \frac{dB}{d\Omega}$$

NOTE The unit of the angular source strength is $s^{-1} \cdot sr^{-1}$.

3.1.12

spectral source strength, energy distribution of neutron source strength

B_E

quotient of dB by dE , where dB is the increment of neutron source strength in the energy interval between E and $E + dE$:

$$B_E = \frac{dB}{dE}$$

NOTE The unit of the spectral source strength is $s^{-1} \cdot J^{-1}$; a frequently used unit is $s^{-1} \cdot eV^{-1}$.

The source strength B is derived from B_E as follows:

$$B = \int_0^{\infty} B_E dE$$

At a distance l from a point source, the spectral neutron fluence rate φ_E , due to neutrons emitted isotropically from the point source with a spectral neutron source strength B_E (neglecting the influence of surrounding material), is given by (see also 3.1.4):

$$\varphi_E = \frac{B_E}{4\pi l^2}$$

3.1.13

fluence-average neutron energy

$$\bar{E}$$

neutron energy averaged over the spectral neutron fluence. The product of E and Φ_E , is called the fluence spectrum where Φ_E (see 3.1.3 and 3.1.8) is the spectral neutron fluence at neutron energy E :

$$\bar{E} = \frac{1}{\Phi} \int_0^{\infty} E \cdot \Phi_E(E) dE$$

3.1.14

dose equivalent-average neutron energy

$$\tilde{E}$$

neutron energy averaged over the dose equivalent spectrum. The dose equivalent spectrum is given by the product of Φ_E and $h_\phi(E)$, where Φ_E (see 3.3) is the spectral neutron fluence at neutron energy E and $h_\phi(E)$ (see 3.8) is the neutron fluence-to-dose equivalent conversion coefficient at this energy:

$$\tilde{E} = \frac{1}{H} \int_0^{\infty} E \cdot h_\phi(E) \Phi_E dE$$

where:

$$H = \int_0^{\infty} h_\phi(E) \Phi_E dE$$

3.1.15

response

$$R$$

quotient of the reading, M , of a measuring instrument by the conventional true value of the measured quantity. The type of response should be specified, e.g. "fluence response" (response with respect to Φ):

$$R_\phi = \frac{M}{\Phi}$$

or, "dose equivalent response" (response with respect to dose equivalent H).

$$R_H = \frac{M}{H}$$

4 Reference radiations for the calibration of neutron-measuring devices

In this section, reference radiations produced by radionuclide neutron sources are specified which are particularly suited for the calibration of neutron-measuring devices (see ISO 8539-3).

4.1 General Properties

4.1.1 Types

The neutron sources given in Table 1 shall be used to produce reference radiations. The numerical values given in Table 1 are to be taken only as a guide to the prominent features of the sources. The neutron source strengths and the dose equivalent rates may vary with the construction of the source, because of scattering and absorption of neutrons and γ -rays, and with the isotopic impurities of the radioactive material used. Hence details of the source encapsulation are specified (see 4.1.2), and the method for determining the anisotropy of the neutron emission is specified (see 4.3). For ^{252}Cf , the specific photon dose equivalent rate is dependent upon the age of the source because of the build-up of γ -emitting fission products. However, the increase is not more than 5 % during the first 20 years.

NOTE In addition to the sources listed in Table 1, sources such as Pu-Be(α ,n) and Am-Li(α ,n) are also used. However, it is recommended that laboratories should not start using plutonium-beryllium sources if they are not already doing so.

Table 1—Reference radionuclide neutron sources for calibrating neutron-measuring devices

Source	Half-life	Fluence-average energy ¹⁾³⁾	Dose equivalent average energy ¹⁾³⁾	Specific source strength ²⁾	Ratio of photon to neutron dose equivalent rates ¹⁾	Spectrum averaged fluence-to-dose equivalent conversion coefficient ³⁾
	a ⁴⁾	MeV	MeV	s ⁻¹ ·kg ⁻¹	-	pSv · cm ²
^{252}Cf - D ₂ O moderated ⁵⁾	2,65	0,55	2,1	$2,1 \times 10^{15}$	0,18	105
^{252}Cf	2,65	2,13	2,3	$2,4 \times 10^{15}$	0,05 ⁶⁾	385
				s ⁻¹ ·Bq ⁻¹		
$^{241}\text{Am-B}(\alpha,n)$	432	2,72	2,8	$1,6 \times 10^{-5}$	< 0,20 ⁷⁾	408
$^{241}\text{Am-Be}(\alpha,n)$	432	4,16	4,4	$6,6 \times 10^{-5}$	< 0,05 ⁷⁾	391

NOTE 1 Definitions of the fluence, and dose equivalent, average energies are given in Section 3.

NOTE 2 For ^{252}Cf sources, the specific quantities are related to the mass of californium contained in the source [See Annex A]. For the other sources, they are related to the activity of the ^{241}Am contained in the source. Information on the sources is given for moderated ^{252}Cf in the Bibliography [1, 2 and 3], for ^{252}Cf in [4 and 5], for $^{241}\text{Am-B}$ in [6], and for $^{241}\text{Am-Be}$ in [7].

NOTE 3 Calculated on the basis of the neutron spectra given in Annex A and the conversion coefficients given in ICRU Report 57.

NOTE 4 1 a = 1 mean solar year = 31 556 926 s or 365,242 20 days.

NOTE 5 Heavy-water sphere with a diameter of 300 mm, covered with a cadmium shell of thickness approximately 1 mm. Of the source neutrons, 11.5% are moderated below the cadmium cut-off and captured in the cadmium shell (See Annex A).

NOTE 6 For approximately 2,5 mm thick steel encapsulation.

NOTE 7 For a source that has been enclosed within an approximately 1-mm thick lead shield.

4.1.2 Source shape and encapsulation

The shape of the source should be spherical or cylindrical and, in the latter case, it is preferable

that the diameter and length are approximately the same. The thickness of the encapsulation should be uniform and small compared to the external diameter. For a $^{241}\text{Am-Be}(\alpha, n)$ source, the spectral distribution, mainly in the energy range below approximately 2 MeV, depends, to some extent, on the size and the composition of the source [4]. Sources should comply with the encapsulation requirements in ISO 1677 and ISO 2919.

The $^{241}\text{Am-Be}(\alpha, n)$ source may be wrapped in a 1 mm thick lead shield. This reduces the photon dose equivalent rate to less than 5% of the neutron dose equivalent rate. The lead shield produces a negligible change (less than 1%) in the neutron dose equivalent rate. In the absence of the lead shield, the photon dose equivalent rate (mainly from γ -rays having an energy of 59,5 keV) will depend upon the source construction, but may be comparable with the neutron dose equivalent rate.

4.2 Characteristics of calibration sources

4.2.1 Types

Preferably $^{241}\text{Am-Be}(\alpha, n)$ and/or ^{252}Cf spontaneous fission sources should be used for routine calibration (see ISO 8529-3). ^{252}Cf sources generally have a high specific source strength and are therefore comparatively small. Because of their half-life of 2.65 years, they need occasional replacement. The americium-based neutron source shall consist of an americium alloy or a homogeneous, compressed mixture of americium oxide and beryllium or boron as appropriate. Americium alloys may also be used.

4.2.2 Energy distribution of neutron source strength

The energy distributions of neutron source strength for $^{241}\text{Am-Be}(\alpha, n)$, ^{252}Cf , $^{252}\text{Cf}(\text{D}_2\text{O-moderated})$ and $^{241}\text{Am-B}(\alpha, n)$ sources are given in annex A (Tables A.1 to A.4 and Figures A.1 to A.4). The energy distribution of the neutron source strength, B_E , of ^{252}Cf , is given in Annex A. In the energy range from 100 keV to 10 MeV, it can be described by the following formula:

$$B_E = \frac{2}{\sqrt{\pi} T^{3/2}} \times \sqrt{E} \times e^{-E/T} \times B$$

where T is a spectrum parameter given by $T = 1,42 \text{ MeV}$ [4] (see figure A.1).

NOTE The neutron spectra given in Annex A are those recommended for lightly encapsulated sources (see section 4.1.2). The spectrum-averaged fluence-to-dose equivalent conversion coefficients contained in table 1 and in ISO 8529-3 have been calculated for these spectra. For the cases of heavy encapsulation, or special construction of the $\text{D}_2\text{O-moderated } ^{252}\text{Cf}$ source, spectra may change significantly. If such source strength spectra, B_E , or fluence spectra, Φ_E , are known from calculation or measurement, specific spectrum-averaged conversion coefficients should be calculated using:

$$h_\phi = \frac{1}{\Phi_0} \int_0^\infty h_\phi(E) \Phi_E dE$$

(where Φ_E is taken to be proportional to B_E).

4.3 Neutron fluence rate produced by a source

The fluence rate produced by a neutron source is determined primarily from the neutron source strength and the distance between the source center and the point of test. Neutron sources generally show anisotropic neutron emission in a coordinate system fixed in the geometrical centre of the source. For cylindrical sources, the angular source strength, B_Ω , in a direction Ω ,

which is characterized by the angles θ and α (see Figure 1), does not depend noticeably on the azimuth angle α , but only upon angle θ . As the angular source strength $dB/d\Omega$ varies least for $\theta = 90^\circ$, this direction should be used for calibrations.

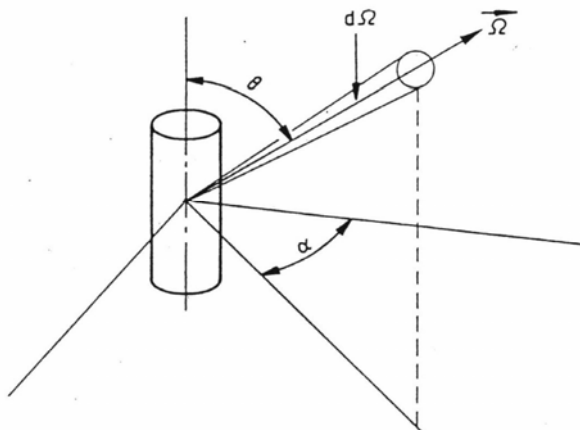


Figure 1—Coordinate system for the case of an anisotropically emitting source.

The neutron source strength, B , and the angular source strength, $dB/d\Omega$, for $\theta = 90^\circ$ shall be determined (see also Annex C).

For this, $\Delta\theta$ shall not be larger than 14° , corresponding to a solid angle $\Delta\Omega = 3,8 \times 10^{-3}$ sr. The neutron fluence rate at a distance l from the center of the source in a direction for which $\theta = 90^\circ$ then may be taken as:

$$\phi_{(l,90^\circ)} = \frac{dB}{d\Omega} \times \frac{1}{l^2}$$

The neutron fluence rate obtained from this expression still has to be corrected for air attenuation, and inscatter from air and the surrounding material. These corrections, which are only negligible in exceptional circumstances, are described in detail in ISO 8529-2.

4.4 Calibration of the neutron source strength

The $^{241}\text{Am-Be}$ (α, n), ^{252}Cf , $^{241}\text{Am-B}$ (α, n) sources should be supplied by the manufacturer with a certificate of their isotopic composition, and the source strength shall be calibrated by a reference laboratory before use. Reference laboratories can generally calibrate these sources to within a relative standard uncertainty of about 1,5%.

There is the possibility, however, that, with time, the constituent components of the americium-beryllium and americium-boron powder sources may shift with respect to each other, with a resultant change in the neutron source strength. It is therefore recommended that these sources be recalibrated every five years.

The source strength of a ^{252}Cf source shall be corrected for radioactive decay on a day-to-day basis. It is important to take into account the decay of all the constituents of the source including the ^{250}Cf in nominal ^{252}Cf . At the present time, the relative standard uncertainty in ^{252}Cf half-life is 0,5% to 0,7%. After about two half-lives (i.e. approximately five years), the uncertainty in the

half-life will thus result in a relative standard uncertainty in the source strength of about 1%, which is comparable to the initial calibration uncertainty. It is therefore recommended that ^{252}Cf sources also be recalibrated every five years.

4.5 Irradiation facility

In general, irradiation rooms have thick walls (for example concrete) for shielding. In this case, the inside dimensions should be as large as practically possible. The magnitude of the correction for room and air-scattered neutrons, and the resulting uncertainty in the irradiation field quantities, depend critically on the size of the room. In all cases, the effects of scattered neutrons shall be determined. Details of the recommended calibration procedures are dealt with in ISO 8529-2.

5 Reference radiations for the determination of the response of neutron-measuring devices as a function of neutron energy

In this clause, reference radiations are specified for the determination of the response of neutron-measuring devices as a function of neutron energy. These reference radiations may also be used to determine dose equivalent rate dependence and directional dependence. Radiations specified in this section may also be used for the calibration of neutron-measuring devices.

Since these reference radiations are available only at specialized laboratories, only the general principles on their method of production and characterization are given.

5.1 General properties

The recommended neutron energies and the methods used for their production are given in Table 2, along with relevant references.

Table 2 – Neutron radiations for determining the response of neutron-measuring devices as a function of neutron energy.

Neutron energy MeV	Method of production	References (see Bibliography)
2,5x10 ⁻⁸ (thermal) ¹⁾	Moderated-reactor or accelerator-produced neutrons	[10] ; [8]
0,002	Scandium-filtered reactor neutron beam or accelerator-produced neutrons from reaction $^{45}\text{Sc}(p,n)^{45}\text{Ti}$	[9] ; [10]
0,024	Iron/aluminium-filtered reactor neutron beam or accelerator-produced neutrons from reaction $^{45}\text{Sc}(p,n)^{45}\text{Ti}$	[9] ; [10] ; [11]
0,144 ¹⁾	Silicon-filtered reactor neutron beam or accelerator-produced neutrons from reactions $\text{T}(p,n)^3\text{He}$ and $^7\text{Li}(p,n)^7\text{Be}$	[9]; [12] ; [13] ; [14]
0,25 ¹⁾	Accelerator-produced neutrons from reactions $\text{T}(p,n)^3\text{He}$ and $^7\text{Li}(p,n)^7\text{Be}$	[12] ; [13] ; [14]
0,565 ¹⁾	Accelerator-produced neutrons from reactions $\text{T}(p,n)^3\text{He}$ and $^7\text{Li}(p,n)^7\text{Be}$	"
1,2	Accelerator-produced neutrons from reaction $\text{T}(p,n)^3\text{He}$	"
2,5 ¹⁾	Accelerator-produced neutrons from reaction $\text{T}(p,n)^3\text{He}$	"
2,8 ^{1) 2)}	Accelerator-produced neutrons from reaction $\text{D}(d,n)^3\text{He}$	"
5,0	Accelerator-produced neutrons from reaction $\text{D}(d,n)^3\text{He}$	"
14,8 ^{1) 2)}	Accelerator-produced neutrons from reaction $\text{T}(d,n)^4\text{He}$	"
19,0	Accelerator-produced neutrons from reaction $\text{T}(d,n)^4\text{He}$	"

NOTE 1 Energies at which international intercomparisons of neutron fluence measurements were performed ^[15],

NOTE 2 Accelerator-produced neutrons, with a deuteron energy of a few hundred keV.

5.2 Reactor reference neutron radiations

5.2.1 General requirements

For calibration purposes, unidirectional beams of neutrons shall be used. If the diameter of the beam is small compared to the dimensions of the measuring device under investigation, broad beam irradiation may be simulated by appropriate sweeping of the measuring device across the beam ^[1,16].

5.2.2 Thermal neutron beams

For the purposes of this International Standard, neutrons in the energy range below the cadmium cut-off energy (corresponding to approximately 0,51 eV for 1 mm cadmium ^[17]), are called "thermal". The "true thermal neutron fluence rate", φ_{th} is the required quantity from which the dose equivalent rate may be derived using the appropriate conversion coefficient, h_{ϕ} .

The true thermal fluence rate shall be determined either directly from a measurement of the spectral fluence rate (for example by time-of-flight spectrometry) or from the "conventional neutron fluence rate" (see annex B), as defined in [17] and measured, for example, by the activation of gold foils [18].

In the special case of a Maxwellian spectrum of thermal neutrons of known temperature, the true neutron fluence rate may be derived directly from the measured activation for a $1/v$ detector (see Annex B).

The neutron beam may be filtered to improve the ratio of dose equivalent produced by thermal neutrons to dose equivalent produced by unwanted radiation (neutrons with energies above the cadmium cut-off energy and photons).

The thermal neutron fluence rate should be carefully monitored, for example by means of a fission chamber, to correct for any variation with time.

5.2.3 Filtered neutron beams from a reactor [9, 10 and 19]

The production of quasi-monoenergetic neutron radiation by means of filtered reactor neutron beams makes use of the existence of deep relative minima in the total cross sections of certain materials at distinct energies (for example 2 keV in scandium, 24 keV in iron and aluminium, and 144 keV in silicon). There also exist further so-called "neutron windows" at other energies. Hence neutron spectrum measurements of the beams shall be made to determine the relative intensity of these neutron groups. In the case of scandium (2 keV), the filters shall be sited in a beam tube tangential to the reactor core [9,10]. The same geometry may also be advantageous for the other filtered reactor beams. Even then the influence of other neutron groups shall be taken into account.

Recoil proton proportional counters and ^3He proportional counters may be used for the spectrometry of the neutron beam. A boron trifluoride or a ^3He proportional counter may be used to measure the absolute fluence rate of the lower energy beams (neutron energies of $E_n \leq 24$ keV) and a recoil proton counter for higher energy beams (neutron energies of $E_n \geq 24$ keV). Boron trifluoride proportional counters or ^3He proportional counters may be used as monitors and transfer instruments.

5.3 Accelerator-produced neutron radiations

5.3.1 General requirements

An accelerator providing protons and deuterons up to an energy of 3,5 MeV is required to generate neutrons of all the energies given in Table 2 [2]. For the production of neutrons with energies of 2,8 MeV and 14,8 MeV, however, a small accelerator with a potential of a few hundred kilovolts is sufficient. When these neutron beams are used for calibrating instruments, the following parameters have to be assessed:

- charged particle energy
- neutron fluence
- neutron spectrum
- effects of scattered neutrons
- target age and thickness

5.3.2 Energy of charged particles

Details on the reaction kinematics determining the neutron energy and the corresponding charged particle energy are given in the Bibliography [12 and 13].

The energy of the incident charged particle beam should be determined. A stabilised analysing magnet calibrated by means of a few known nuclear reaction thresholds may be used in order to select the momentum of the particle beam. The energy loss of the charged particles in the target shall be taken into account in the calculation of the bombarding energy needed to produce the required neutron reference radiation energy. Recent values of the stopping power for protons in different materials are given in the Bibliography [20].

5.3.3 Neutron spectrum

Due to energy losses in the target, and other influences, accelerated charged particles generate, at a given angle, neutron beams with a narrow, but finite, width in energy around the stated reference energy. As a rule, it is not necessary to consider this energy spread when applying the fluence-to-dose equivalent conversion coefficients in order to calculate the dose equivalent quantities. Invariably, the conversion coefficients for the “monoenergetic” neutrons at the stated energy are used.

With endothermic reactions, two neutron groups are produced near the threshold relative to the incident proton beam. This is the case for the $T(p,n) \text{ } ^3\text{He}$ reaction if it is used to provide neutron energies of either 144 keV or 250 keV at 0° . In order to obtain monoenergetic neutrons of these energies, larger angles of neutron emission should be used with charged particles of correspondingly higher energies. For the exothermic $T(d,n)$ reaction, account must be taken of neutrons produced by the lower energy $D(d,n)$ reaction, especially for thin targets.

Excited states of the residual nuclei are formed for scandium and lithium for neutrons produced at 0° with energies above 53 keV and 650 keV, respectively. These higher particle energies should only be used if the response of the instrument to the resulting additional neutron energy group, as well as the relative intensity of the secondary group to that of the primary group are known.

5.3.4 Scattered neutron background

Scattered neutron background shall be considered:

- a) in the measurement of the neutron fluence;
- b) in monitoring the neutron production;
- c) in evaluating the performance of the instrument under investigation.

Measurements with a shadow cone and investigations of deviations from the $1/l^2$ -relationship (where l is the distance between the neutron source and the detector) may be of help.

In order to reduce the influence of the background on a measurement, a reaction angle of 0° should be used wherever possible. In order to reduce the effect of scattered neutrons, the room used for the measurements shall be as large as possible (see section 4.5), and the target assembly should have as low a mass as possible.

5.3.5 Neutron fluence measurement and monitoring

Neutron reference laboratories shall provide practical guidance in the measurement of neutron fluence. Appropriate methods and instruments may include:

- a) counters measuring recoil protons (hydrogen-filled proportional counters, recoil proton telescopes, scintillation detectors);
- b) activation of threshold and resonance detectors;
- c) fission fragment detectors;
- d) detectors of well known, calibrated efficiency (for example a Precision Long Counter).

The neutron fluence shall be determined at the location of the instrument to be calibrated. A fluence monitor at another position shall be used during the calibration. The monitor will then indicate the fluence at the location of calibration.

Annex A

(normative)

Tabular and graphical representation of the neutron spectra for radionuclide sources

Tabular Data Presentation

The spectra for the radionuclide sources listed in Table 1 are represented in tables A.1 to A.4 as group source strengths, B_i , in certain energy intervals, *i.e.*, the source strength of neutrons having energies between E_i and E_{i+1} :

$$B_i = \int_{E_i}^{E_{i+1}} B_E dE$$

where B_E is the spectral source strength. The B_i were calculated by numerical integration using the analytical function given in section 4.2.2 for ^{252}Cf spontaneous fission neutrons. For the $^{241}\text{Am-Be}(\alpha, n)$ and $^{241}\text{Am-B}(\alpha, n)$ sources, experimental data were used; the neutron source strengths below the measurement threshold were estimated by extrapolating the spectral source strength linearly from the value at minimum quoted energy to zero at $E_n = 0$ MeV. For the D_2O -moderated ^{252}Cf neutron spectrum, Monte Carlo calculations were used at a distance > 15 cm from the surface of the source [2].

In the tables, the energy given for each group source strength value, B_i , is the lower limit, E_i , of the energy interval, i ; the last energy given in each table is the upper limit of the last energy interval. The group source strength values are normalized to a total source strength $B = 1 \cdot \text{s}^{-1}$, *i.e.*

$$\sum_{i=1}^n B_i = 1 \cdot \text{s}^{-1}$$

for the $^{241}\text{Am-Be}(\alpha, n)$, $^{241}\text{Am-B}(\alpha, n)$ and ^{252}Cf sources. For the D_2O -moderated ^{252}Cf source, 11,5% of the source neutrons are moderated below the cadmium cut-off and captured in the cadmium shell; hence, for this source group, source strengths sum to $0,885 \cdot \text{s}^{-1}$.

The numbers in the table permit calculation of the portion of the source strength between energies E_a and E_b by simply summing up the respective group source strengths:

$$B_{E_a}^{E_b} = \sum_{i=a}^{i=b-1} B_i$$

Graphical representation

While group source strength values are the basic physical data stemming from measurements or calculations and are to be used for further calculations of integrals, they are inappropriate for graphical representation of the spectra since their values depend on the (arbitrary) width of the energy intervals.

If spectra are given as a continuous (analytical) function, the most common graphical representations are spectral source strength, $B_E = dB/dE$, vs. energy, E , if the E axis is linearly scaled, or $dB/d(\ln E/E_0)$, if the E axis is logarithmic. (The latter had been historically known as "lethargy plots"; the arbitrary parameter E_0 is needed to make the argument of the logarithm of dimension 1.) Since $d(\ln x) = dx/x$, it follows that $dB/d(\ln E/E_0) = E \cdot dB/dE = E \cdot B_E$. With these adoptions spectra can be plotted in such a way that equal areas under curves represent equal

source strength proportions

$$\int_{E_1}^{E_2} B_E(E) \cdot dE = \int_{E_1}^{E_2} E \cdot B_E(E) \cdot (dE/E)$$

In the figures A.1 to A.4, the spectra are represented as plots of $E_n \cdot B_E$ (on a linear scale) versus the neutron energy, E_n (on a logarithmic scale). The curves are histograms reflecting the restricted knowledge of the spectral shape. Whereas in a plot with a linearly scaled abscissa the ordinate values would be derived as $B_E = \Delta B / \Delta E = B_i / (E_{i+1} - E_i)$, for the plots A.1 to A.4 they have been calculated by

$$E_n \cdot B_E = B_i / \ln(E_{i+1}/E_i).$$

Table A.1 - Values of group source strength for a D₂O-moderated ²⁵²Cf spontaneous fission source (See Table 1)

E_i (MeV)	B_i (s ⁻¹)
4,14E-07	1,90E-02
1,00E-06	6,31E-02
1,00E-05	6,04E-02
5,00E-05	3,17E-02
1,00E-04	3,41E-02
2,00E-04	3,82E-02
4,00E-04	3,28E-02
7,00E-04	2,24E-02
1,00E-03	7,56E-02
3,00E-03	5,09E-02
6,00E-03	3,79E-02
1,00E-02	5,47E-02
2,00E-02	5,12E-02
4,00E-02	2,96E-02
6,00E-02	2,00E-02
8,00E-02	1,45E-02
1,00E-01	2,47E-02
1,50E-01	1,59E-02
2,00E-01	1,14E-02
2,50E-01	8,90E-03
3,00E-01	6,57E-03
3,50E-01	4,89E-03
4,00E-01	2,65E-03
4,50E-01	3,14E-03
5,00E-01	4,20E-03
5,50E-01	4,12E-03
6,00E-01	7,83E-03
7,00E-01	6,78E-03
8,00E-01	5,75E-03
9,00E-01	3,57E-03
1,00E+00	7,48E-03
1,20E+00	8,43E-03
1,40E+00	9,13E-03
1,60E+00	8,55E-03
1,80E+00	8,07E-03
2,00E+00	1,34E-02
2,30E+00	1,45E-02
2,60E+00	1,49E-02
3,00E+00	1,23E-02
3,50E+00	8,19E-03
4,00E+00	8,10E-03
4,50E+00	6,54E-03
5,00E+00	8,70E-03
6,00E+00	4,93E-03
7,00E+00	2,42E-03
8,00E+00	1,30E-03
9,00E+00	7,66E-04
1,00E+01	4,43E-04
1,10E+01	1,62E-04
1,20E+01	1,24E-04
1,30E+01	5,93E-05
1,40E+01	2,83E-05
1,50E+01	

Table A.2 - Values of group source strength for a ²⁵²Cf spontaneous fission source.

E_i (MeV)	B_i (s ⁻¹)
4,14E-07	3,10E-10
1,00E-06	1,11E-08
1,00E-05	1,27E-07
5,00E-05	2,76E-07
1,00E-04	7,82E-07
2,00E-04	2,21E-06
4,00E-04	4,53E-06
7,00E-04	5,68E-06
1,00E-03	5,51E-05
3,00E-03	1,28E-04
6,00E-03	2,30E-04
1,00E-02	7,74E-04
2,00E-02	2,17E-03
4,00E-02	2,80E-03
6,00E-02	3,29E-03
8,00E-02	3,68E-03
1,00E-01	1,05E-02
1,50E-01	1,21E-02
2,00E-01	1,33E-02
2,50E-01	1,42E-02
3,00E-01	1,49E-02
3,50E-01	1,55E-02
4,00E-01	1,60E-02
4,50E-01	1,63E-02
5,00E-01	1,66E-02
5,50E-01	1,68E-02
6,00E-01	3,38E-02
7,00E-01	3,39E-02
8,00E-01	3,37E-02
9,00E-01	3,33E-02
1,00E+00	6,46E-02
1,20E+00	6,12E-02
1,40E+00	5,73E-02
1,60E+00	5,31E-02
1,80E+00	4,88E-02
2,00E+00	6,55E-02
2,30E+00	5,67E-02
2,60E+00	6,33E-02
3,00E+00	6,21E-02
3,50E+00	4,68E-02
4,00E+00	3,49E-02
4,50E+00	2,58E-02
5,00E+00	3,30E-02
6,00E+00	1,74E-02
7,00E+00	9,01E-03
8,00E+00	4,61E-03
9,00E+00	2,33E-03
1,00E+01	1,17E-03
1,10E+01	5,83E-04
1,20E+01	2,88E-04
1,30E+01	1,42E-04
1,40E+01	6,94E-05
1,50E+01	

Table A.3 - Values of group source strength for a ²⁴¹Am-B(α,n) source.

E_i (MeV)	B_i (s ⁻¹)
-------------	--------------------------

4,14E-07	1,75E-02
8,20E-01	1,13E-02
1,09E+00	8,07E-03
1,34E+00	2,10E-02
1,56E+00	4,54E-02
1,78E+00	6,31E-02
1,98E+00	7,99E-02
2,17E+00	8,90E-02
2,36E+00	9,26E-02
2,54E+00	9,65E-02
2,72E+00	8,31E-02
2,89E+00	7,06E-02
3,05E+00	6,67E-02
3,22E+00	4,99E-02
3,38E+00	4,02E-02
3,53E+00	3,17E-02
3,68E+00	3,03E-02
3,83E+00	2,52E-02
3,98E+00	1,98E-02
4,13E+00	1,72E-02
4,27E+00	1,45E-02
4,41E+00	9,97E-03
4,55E+00	7,46E-03
4,69E+00	3,41E-03
4,83E+00	2,19E-03
4,96E+00	1,16E-03
5,09E+00	3,03E-04
5,22E+00	2,68E-04
5,35E+00	2,36E-04
5,48E+00	1,15E-04
5,61E+00	1,45E-04
5,74E+00	1,39E-04
5,86E+00	2,78E-04
5,98E+00	1,78E-04
6,11E+00	2,91E-04
6,19E+00	

1,10E-01	3,34E-02
3,30E-01	3,13E-02
5,40E-01	2,81E-02
7,50E-01	2,50E-02
9,70E-01	2,14E-02
1,18E+00	1,98E-02
1,40E+00	1,75E-02
1,61E+00	1,92E-02
1,82E+00	2,23E-02
2,04E+00	2,15E-02
2,25E+00	2,25E-02
2,47E+00	2,28E-02
2,68E+00	2,95E-02
2,90E+00	3,56E-02
3,11E+00	3,69E-02
3,32E+00	3,46E-02
3,54E+00	3,07E-02
3,75E+00	3,00E-02
3,97E+00	2,69E-02
4,18E+00	2,86E-02
4,39E+00	3,18E-02
4,61E+00	3,07E-02
4,82E+00	3,33E-02
5,04E+00	3,04E-02
5,25E+00	2,74E-02
5,47E+00	2,33E-02
5,68E+00	2,06E-02
5,89E+00	1,82E-02
6,11E+00	1,77E-02
6,32E+00	2,04E-02
6,54E+00	1,83E-02
6,75E+00	1,63E-02
6,96E+00	1,68E-02
7,18E+00	1,68E-02
7,39E+00	1,88E-02
7,61E+00	1,84E-02
7,82E+00	1,69E-02
8,03E+00	1,44E-02
8,25E+00	9,68E-03
8,46E+00	6,52E-03
8,68E+00	4,26E-03
8,89E+00	3,67E-03
9,11E+00	3,81E-03
9,32E+00	5,06E-03
9,53E+00	6,25E-03
9,75E+00	5,52E-03
9,96E+00	4,68E-03
1,02E+01	3,70E-03
1,04E+01	2,78E-03
1,06E+01	1,51E-03
1,08E+01	3,63E-04
1,10E+01	

Table A.4 - Values of group source strength for a $^{241}\text{Am-Be}(\alpha, n)$ source.

E_i (MeV)	B_i (s $^{-1}$)
4,14E-07	1,44E-02

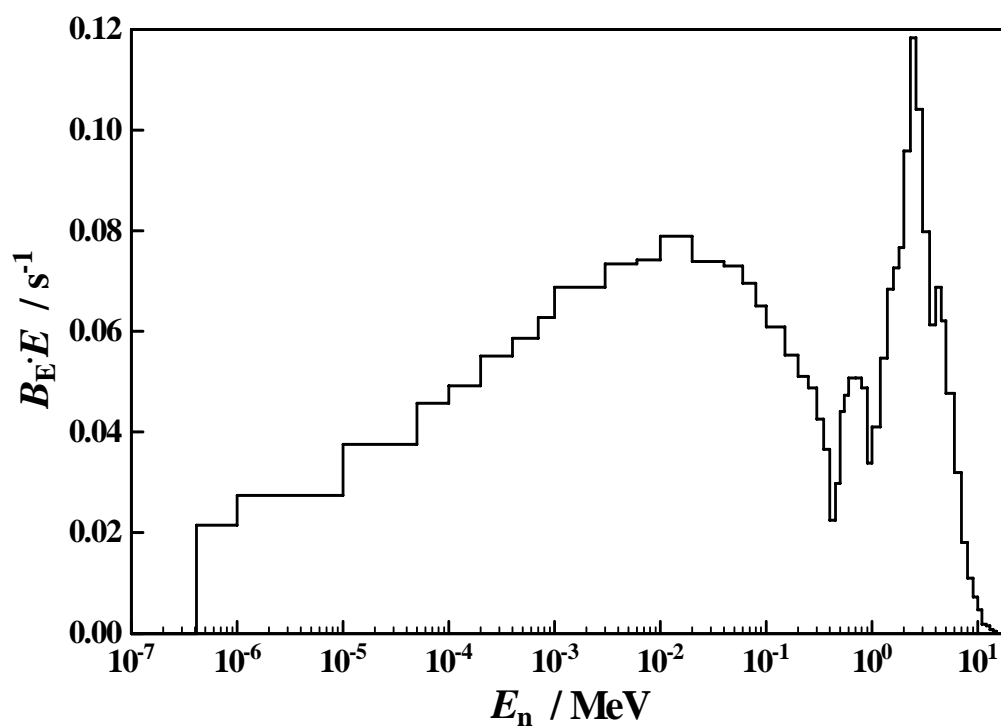


Figure A.1 Neutron spectrum outside a 150-mm radius D_2O sphere with a ^{252}Cf spontaneous fission neutron source at its centre.

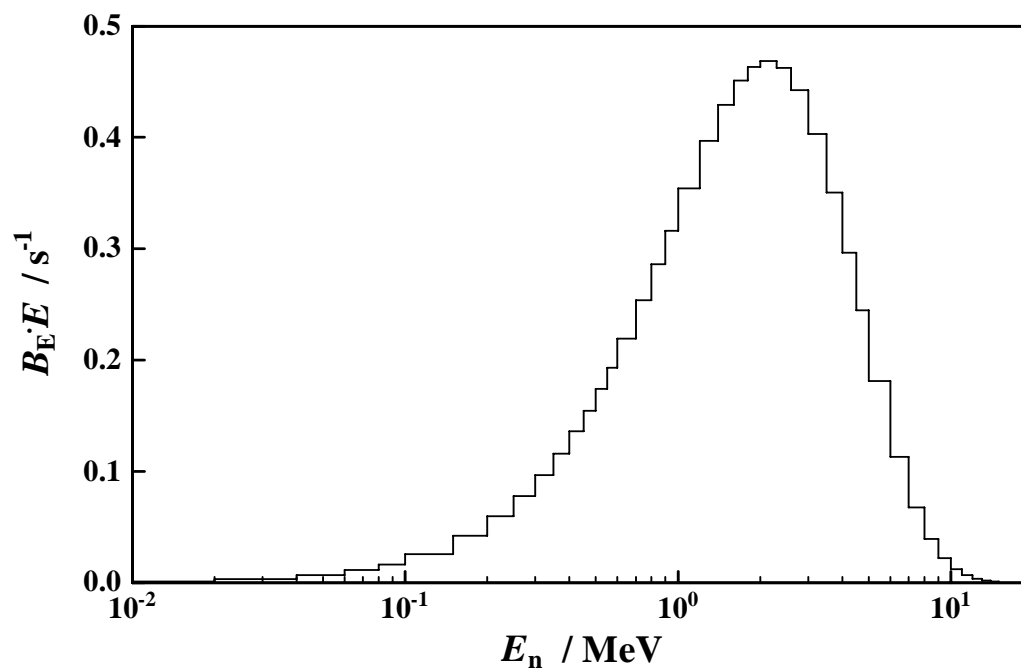


Figure A.2 Neutron spectrum from a ^{252}Cf spontaneous fission source.

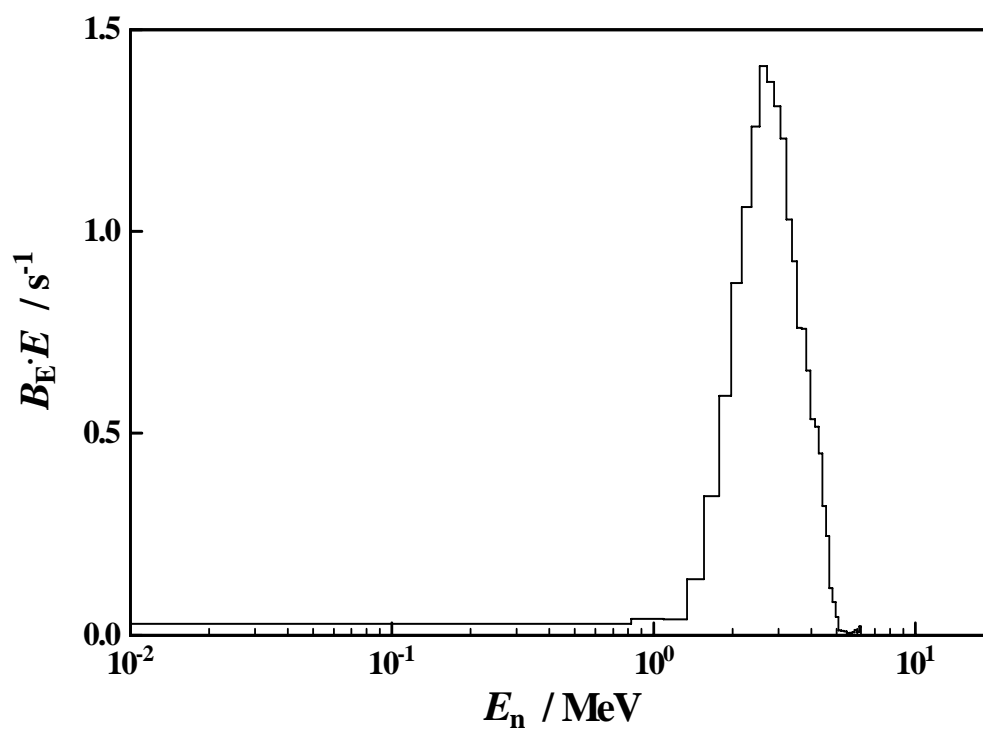


Figure A.3 Neutron spectrum from a $^{241}\text{AmB}(\alpha, n)$ source.

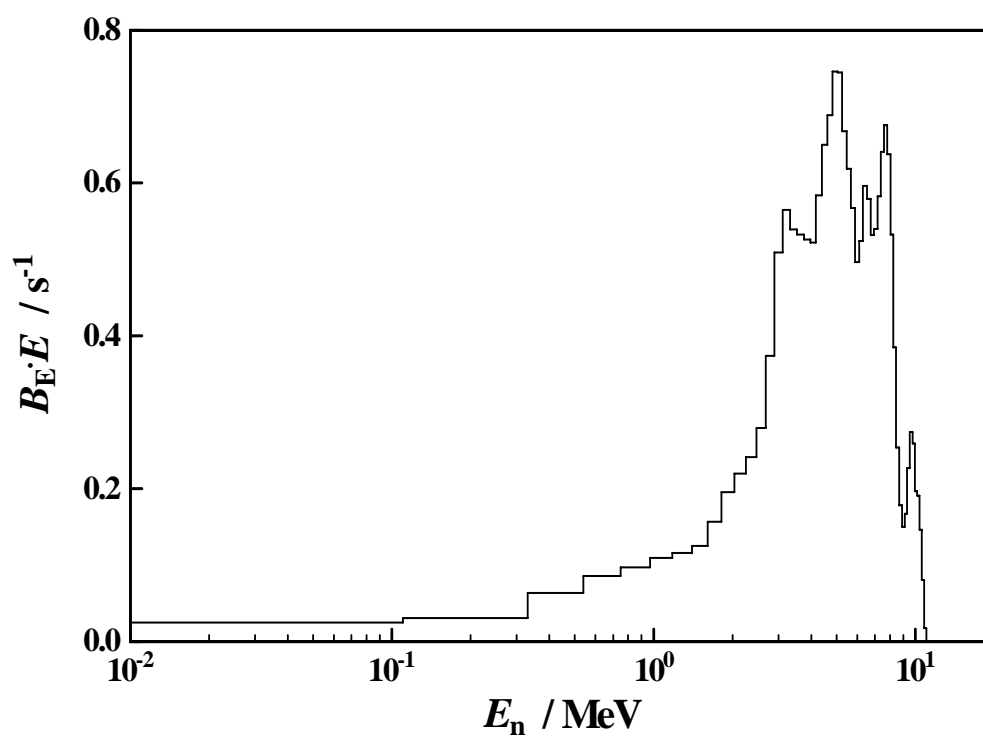


Figure A.4 Neutron spectrum from a $^{241}\text{Am-Be}(\alpha, n)$ source.

Annex B

(normative)

Conventional thermal neutron fluence rate

The “conventional thermal neutron fluence rate” or “conventional thermal neutron flux density”, φ_0 , is given by the following formula:

$$\varphi_0 = \int_0^{E_{cd}} \left(\frac{E_0}{E} \right)^{1/2} \varphi_E(E) dE$$

where,

E is the neutron energy;

E_{cd} is the cadmium cut-off energy;

φ_E is the spectral neutron fluence rate;

$E_0 = 0,0253$ eV ($v = 2200$ m · s⁻¹) is the reference energy for which cross-section values σ_0 for 1/v detectors are tabulated.

The conventional thermal neutron fluence rate is also given by,

$$\varphi_0 = \dot{n}_R / \Sigma_0$$

where,

\dot{n}_R is the reaction rate density, and

Σ_0 is the cross-section density, given by:

$$\Sigma_0 = \rho \frac{p}{M} N_A \sigma_0$$

in which,

ρ is the density of the detector;

p is the isotopic abundance;

M is the molar mass of the detector

N_A is the Avogadro constant and,

σ_0 is the cross-section at E_0 .

For a Maxwellian velocity distribution at a thermodynamic temperature of 20° C, with the energy parameter, $E_0 = 0,0253$ eV, the true thermal fluence rate, φ_{th} , is given by:

$$\varphi_{th} = \frac{2}{\sqrt{\pi}} \varphi_0 = 1,128 \varphi_0$$

where the conventional neutron fluence rate used here does not include neutron energies above E_{cd} .

Annex C

(informative)

Angular Source Strength Characteristics of Two Radionuclide Neutron Sources

Radionuclide neutron source generally show anisotropic emission in a co-ordinate system fixed in the geometrical centre of the source. The co-ordinate system is shown in Section 4.3, Figure 1. In general, the variation of the angular source strength with angle is specific to each individual neutron source, with measurable differences between radionuclide neutron sources of the same type, and supposedly, identical encapsulation. Figures C.1 and C.2 indicate the variations that have been observed for two differently encapsulated californium spontaneous fission sources.

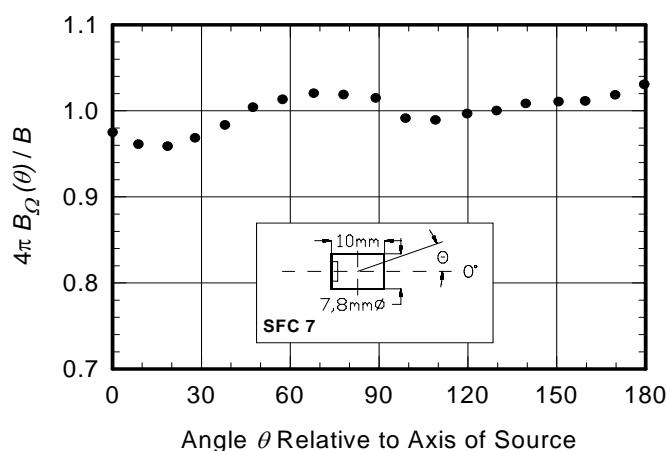


Figure C.1 Angular source strength of a small size ^{252}Cf spontaneous fission source (approximate dimensions of active volume: 4.6 mm diameter by 6 mm long) normalized to the angular source strength $B/4\pi$ of the equivalent point source.

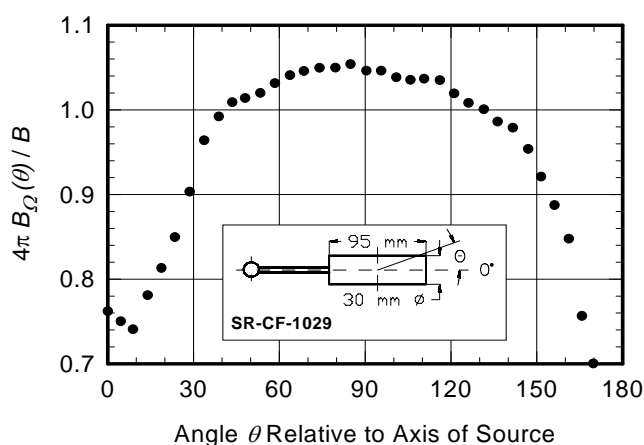


Figure C.2 Angular source strength of a high intensity ^{252}Cf spontaneous fission source (approximate dimensions of active volume: 3 mm diameter by 30 mm long) normalized to the angular source strength $B/4\pi$ of the equivalent point source.

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