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Reference neutron radiations —

Part 1:
Characteristics and methods of production

Rayonnements neutroniques de référence —

Partie 1: Caractéristiques et méthodes de production

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Foreword

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International Standards are drafted in accordance with the rules given in the ISO/IEC Directives, Part 3.

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.

Attention is drawn to the possibility that some of the elements of this part of ISO 8529 may be the subject of patent rights. ISO shall not be held responsible for identifying any or all such patent rights.

International Standard ISO 8529-1 was prepared by Technical Committee ISO/TC 85, *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 of radiation protection devices related to the basic quantities characterizing the radiation field*

— *Part 3: Calibration of area and personal dosimeters and determination of response as a function of energy and angle of incidence*

Annexes A and C form a normative part of this part of ISO 8529. Annex B is for information only.

Introduction

This part of ISO 8529 supersedes ISO 8529:1989. It is the first of a set of three International Standards concerning the calibration of dosimeters and dose-rate 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. ISO 8529-2 describes fundamentals related to the physical quantities characterizing the radiation field and calibration procedures in general terms, with emphasis on active dose-rate meters and the use of radionuclide sources. 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 International Commission on Radiation Units and Measurements (ICRU) operational quantities. Conversion coefficients for converting neutron fluence into these operational quantities are provided in ISO 8529-3.

Reference neutron radiations —

Part 1:
Characteristics and methods of production

1 Scope

This part of ISO 8529 specifies the reference neutron 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 1×10^9 $m^{-2} \cdot s^{-1}$, corresponding, at a neutron energy of 1 MeV, to dose-equivalent rates of up to 100 $mSv \cdot h^{-1}$.

This part of ISO 8529 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 ISO 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 part of ISO 8529, into the following two separate sections.

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

— In clause 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 specialized 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 normative 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 normative documents contain provisions which, through reference in this text, constitute provisions of this part of ISO 8529. For dated references, subsequent amendments to, or revisions of, any of these publications do not apply. However, parties to agreements based on this part of ISO 8529 are encouraged to investigate the possibility of applying the most recent editions of the normative documents indicated below. For undated references, the latest edition of the normative document referred to applies. Members of ISO and IEC maintain registers of currently valid International Standards.

ISO 8529-2:2000 *Reference neutron radiations — Part 2: Calibration fundamentals of radiation protection devices related to the basic quantities characterizing the radiation field*.

ISO 8529-3:1998, *Reference neutron radiations — Part 3: Calibration of area and personal dosimeters and determination of response as a function of energy and angle of incidence*.

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 33:1980, *Radiation Quantities and Units.*

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

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

3 Tests and definitions

For the purposes of this part of ISO 8529, the terms and definitions given in ICRU Reports 33 and 51 and the following apply.

3.1
neutron fluence
$\varPhi$
quotient of dN by da, where dN is the number of neutrons incident on a sphere of cross-sectional area da

$$\varPhi = \frac{\mathrm{d}N}{\mathrm{d}a}$$

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

3.2
neutron fluence rate
neutron flux density
φ
quotient of d$\varPhi$ by dt, where d$\varPhi$ is the increment of **neutron fluence** (3.1) in the time interval dt

$$\varphi = \frac{\mathrm{d}\varPhi}{\mathrm{d}t} = \frac{\mathrm{d}^2 N}{\mathrm{d}a \cdot \mathrm{d}t}$$

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

3.3
spectral neutron fluence
energy distribution of the neutron fluence
$\varPhi_E$
quotient of d$\varPhi$ by dE, where d$\varPhi$ is the increment of neutron fluence in the energy interval between E and E + dE

$$\Phi_E = \frac{\mathrm{d}\Phi}{\mathrm{d}E}$$

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

3.4
spectral neutron fluence rate
spectral neutron flux density
φ_E
quotient of $\mathrm{d}\Phi_E$ by $\mathrm{d}t$, where $\mathrm{d}\Phi_E$ is the increment of spectral neutron fluence in the time interval $\mathrm{d}t$

$$\varphi_E = \frac{\mathrm{d}\Phi_E}{\mathrm{d}t} = \frac{\mathrm{d}^2\Phi}{\mathrm{d}E \cdot \mathrm{d}t}$$

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

3.5
absorbed dose
D
quotient of $\mathrm{d}\bar{\varepsilon}$ by $\mathrm{d}m$, where $\mathrm{d}\bar{\varepsilon}$ is the mean energy imparted by ionizing radiation to matter of mass $\mathrm{d}m$

$$D = \frac{\mathrm{d}\bar{\varepsilon}}{\mathrm{d}m}$$

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

3.6
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

$$H = QD$$

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

3.7
dose-equivalent rate
$\dot{H}$
quotient of $\mathrm{d}H$ by $\mathrm{d}t$, where $\mathrm{d}H$ is the increment of dose equivalent in the time interval $\mathrm{d}t$

$$\dot{H} = \frac{\mathrm{d}H}{\mathrm{d}t}$$

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

3.8
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

$$h_\Phi = \frac{H}{\Phi}$$

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

3.9
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

$$A = \frac{\mathrm{d}N^+}{\mathrm{d}t}$$

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

3.10
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

$$B = \frac{\mathrm{d}N^*}{\mathrm{d}t}$$

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

3.11
angular source strength
B_Ω
quotient of dB by dΩ, where dB is the number of neutrons per unit time propagating in a specified direction within the solid angle dΩ

$$B_\Omega = \frac{\mathrm{d}B}{\mathrm{d}\Omega}$$

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

3.12
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

$$B_E = \frac{\mathrm{d}B}{\mathrm{d}E}$$

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

NOTE 2 The source strength B is derived from B_E as follows:

$$B = \int_0^\infty B_E \mathrm{d}E$$

At a distance l from a point source, the **spectral neutron fluence rate** φ_E (3.4), 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

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

3.13
fluence-average neutron energy
$\bar{E}$
neutron energy averaged over the spectral neutron fluence

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

3.14
dose-equivalent-average neutron energy
$\tilde{E}$
neutron energy averaged over the dose-equivalent spectrum

$$\tilde{E} = \frac{1}{H} \int_0^\infty E \cdot h_\Phi(E) \Phi_E \mathrm{d}E$$

NOTE In the above equation, $H = \int_0^\infty h_\Phi(E) \Phi_E \mathrm{d}E$.

3.15
response
R
quotient of the reading, M, of a measuring instrument and the conventional true value of the measured quantity

NOTE 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

4.1 Introduction

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

4.2 General properties

4.2.1 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 4.2.2), and the method for determining the anisotropy of the neutron emission is specified (see 4.4). 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.

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** a[d]	**Fluence-average energy**[a,b] MeV	**Dose-equivalent-average energy**[a,b] MeV	**Specific source strength**[c] $s^{-1} \cdot kg^{-1}$	**Ratio of photon to neutron dose-equivalent rates**[c]	**Spectrum averaged fluence-to-dose-equivalent conversion coefficient**[b] $pSv \cdot cm^2$
^{252}Cf (D_2O moderated) [e]	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 [f]	385
				$s^{-1} \cdot Bq^{-1}$		
^{241}Am-B(α,n)	432	2,72	2,8	$1{,}6 \times 10^{-5}$	< 0,20 [g]	408
^{241}Am-Be(α,n)	432	4,16	4,4	$6{,}6 \times 10^{-5}$	< 0,05 [g]	391

a Definitions of the fluence, and dose-equivalent-average energies are given in 3.13 and 3.14 respectively.

b Calculated on the basis of the neutron spectra given in annex A and the conversion coefficients given in ICRU Report 57.

c For ^{252}Cf sources, the specific quantities are related to the mass of californium contained in the source (see normative 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], [3] and [5], for ^{252}Cf in [1] and [4], for ^{241}Am-B in [6], and for ^{241}Am-Be in [7].

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

e 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).

f For approximately 2,5 mm thick steel encapsulation.

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

4.2.2 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}Am-Be(α,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. Sources should comply with the encapsulation requirements in ISO 2919.

The ^{241}Am-Be(α,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.

4.3 Characteristics of calibration sources

4.3.1 Types

Preferably ^{241}Am-Be (α,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.

4.3.2 Energy distribution of neutron source strength

The energy distributions of neutron source strength for ^{241}Am-Be (α,n), ^{252}Cf, ^{252}Cf(D_2O-moderated) and ^{241}Am-B (α,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 MeV)[4] (see Figure A.1).

The neutron spectra given in annex A are those recommended for lightly encapsulated sources (see 4.2.2). 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 D_2O-moderated ^{252}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:

$$h_\Phi = \frac{1}{\Phi}\int_0^\infty h_\Phi(E)\,\Phi_E\,\mathrm{d}E$$

where Φ_E is taken to be proportional to B_E.

4.4 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 centre 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/dΩ varies least for θ = 90°, this direction should be used for calibrations.

The neutron source strength, B, and the angular source strength, $dB/d\Omega$, for $\theta = 90°$ shall be determined (see also informative annex B).

For this, $\Delta\theta$ shall not be larger than 4°, corresponding to a solid angle $\Delta\Omega = 3{,}8 \times 10^{-3}$ sr. The neutron fluence rate, at a distance l from the centre of the source in a direction for which $\theta = 90°$, may then be taken as:

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

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.

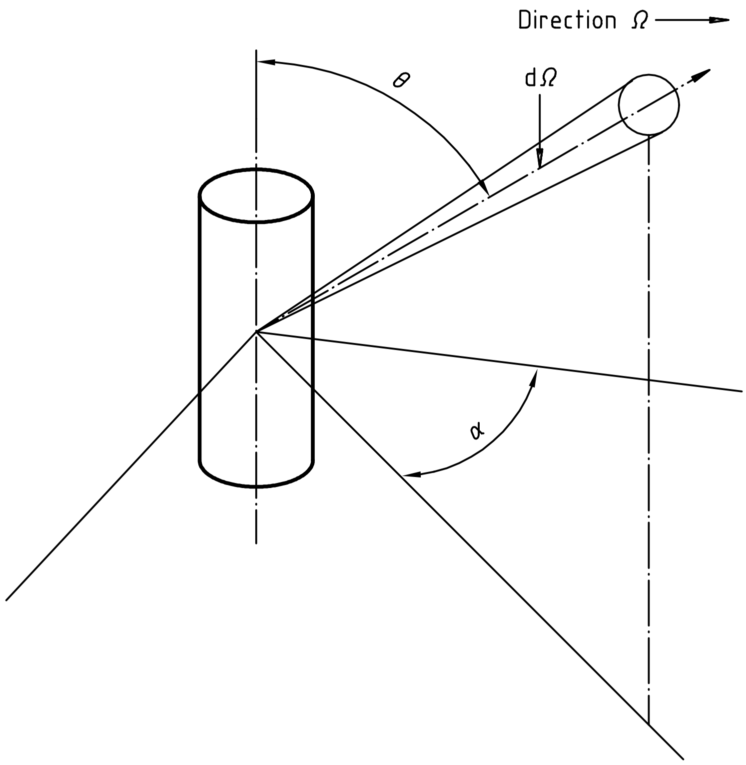

Figure 1 — Coordinate system for the case of an anisotropically emitting source

4.5 Calibration of the neutron source strength

The ^{241}Am-Be (α,n), ^{252}Cf and ^{241}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.

4.6 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

5.1 Introduction

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 clause 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.

5.2 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)
2,5 × 10-8 (thermal) [a]	Moderated-reactor or accelerator-produced neutrons	[10]; [8]
0,002	Scandium-filtered reactor neutron beam or accelerator-produced neutrons from reaction $^{45}Sc(p,n)\ ^{45}Ti$	[9]; [10]
0,024	Iron/aluminium-filtered reactor neutron beam or accelerator-produced neutrons from reaction $^{45}Sc(p,n)\ ^{45}Ti$	[9]; [10]; [11]
0,144 [a]	Silicon-filtered reactor neutron beam or accelerator-produced neutrons from reactions $T(p,n)\ ^{3}He$ and $^{7}Li(p,n)\ ^{7}Be$	[9]; [12]; [13]; [14]
0,25 [a]	Accelerator-produced neutrons from reactions $T(p,n)\ ^{3}He$ and $^{7}Li(p,n)\ ^{7}Be$	[12]; [13]; [14]
0,565 [a]	Accelerator-produced neutrons from reactions $T(p,n)\ ^{3}He$ and $^{7}Li(p,n)\ ^{7}Be$	
1,2	Accelerator-produced neutrons from reaction $T(p,n)\ ^{3}He$	
2,5 [a]	Accelerator-produced neutrons from reaction $T(p,n)\ ^{3}He$	
2,8 [a, b]	Accelerator-produced neutrons from reaction $D(d,n)\ ^{3}He$	
5,0	Accelerator-produced neutrons from reaction $D(d,n)\ ^{3}He$	
14,8 [a, b]	Accelerator-produced neutrons from reaction $T(d,n)\ ^{4}He$	
19,0	Accelerator-produced neutrons from reaction $T(d,n)\ ^{4}He$	

a Energies at which international intercomparisons of neutron fluence measurements were performed [15].

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

5.3 Reference neutron radiations produced by reactors

5.3.1 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].

5.3.2 Thermal-neutron beams

For the purposes of this part of ISO 8529, neutrons in the energy range below the cadmium cut-off energy (corresponding to approximately 0,51 eV for 1 mm of 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 normative 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.

5.3.3 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 ^{3}He proportional counters may be used for the spectrometry of the neutron beam. A boron trifluoride or a ^{3}He proportional counter may be used to measure the absolute fluence rate of the lower energy beams (neutron energies of $E_{\text{n}} \leqslant 24$ keV) and a recoil proton counter for higher energy beams (neutron energies of $E_{\text{n}} > 24$ keV). Boron trifluoride proportional counters or ^{3}He proportional counters may be used as monitors and transfer instruments.

5.4 Accelerator-produced neutron radiations

5.4.1 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 neutrons are used for calibrating instruments, the following parameters shall be assessed:

— charged particle energy;

— neutron fluence;

— neutron spectrum;

— effects of scattered neutrons;

— target age and thickness.

5.4.2 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].

5.4.3 Neutron spectrum

Due to energy losses in the target, and other influences, accelerated charged particles generate, at a given angle, neutrons 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) ^{3}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.

5.4.4 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 4.6), and the target assembly should have as low a mass as possible.

5.4.5 Neutron fluence measurement and monitoring

Neutron reference laboratories shall provide practical guidance on 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

A.1 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}:

$$B_i = \int_{E_i}^{E_{i+1}} B_E \, \mathrm{d}E$$

where B_E is the spectral source strength. The B_i were calculated by numerical integration using the analytical function given in section 4.3.2 for ^{252}Cf spontaneous fission neutrons. For the ^{241}Am-Be(α,n) and ^{241}Am-B(α,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 Tables A.1 to A.4, 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 $B = 1\ \mathrm{s}^{-1}$, i.e.

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

for the ^{241}Am-Be(α,n), ^{241}Am-B(α,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 s^{-1}.

The numbers in Tables A.1 to A.4 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:

$$B_{E_\mathrm{a}}^{E_\mathrm{b}} = \sum_{\mathrm{a}}^{\mathrm{b}-1} B_i$$

A.2 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, $B_E = \mathrm{d}B/\mathrm{d}E$, vs. energy, E, if the E-axis is linearly scaled, or $\mathrm{d}B/\mathrm{d}(\ln E/E_0)$, if the E-axis is logarithmic. (The latter was 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

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

In Figures A.1 to A.4, the spectra are represented as plots of $E \cdot B_E$ (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 in Figures A.1 to A.4 they have been calculated by

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

Table A.1 — Values of group source strength for a D_2O**-moderated** ^{252}Cf **spontaneous fission source** (See Table 1)

E_i(MeV)	B_i(s^{-1})	E_i(MeV)	B_i(s^{-1})
$4{,}14 \times 10^{-7}$	$1{,}90 \times 10^{-2}$	$7{,}00 \times 10^{-1}$	$6{,}78 \times 10^{-3}$
$1{,}00 \times 10^{-6}$	$6{,}31 \times 10^{-2}$	$8{,}00 \times 10^{-1}$	$5{,}75 \times 10^{-3}$
$1{,}00 \times 10^{-5}$	$6{,}04 \times 10^{-2}$	$9{,}00 \times 10^{-1}$	$3{,}57 \times 10^{-3}$
$5{,}00 \times 10^{-5}$	$3{,}17 \times 10^{-2}$	$1{,}00 \times 10^{0}$	$7{,}48 \times 10^{-3}$
$1{,}00 \times 10^{-4}$	$3{,}41 \times 10^{-2}$	$1{,}20 \times 10^{0}$	$8{,}43 \times 10^{-3}$
$2{,}00 \times 10^{-4}$	$3{,}82 \times 10^{-2}$	$1{,}40 \times 10^{0}$	$9{,}13 \times 10^{-3}$
$4{,}00 \times 10^{-4}$	$3{,}28 \times 10^{-2}$	$1{,}60 \times 10^{0}$	$8{,}55 \times 10^{-3}$
$7{,}00 \times 10^{-4}$	$2{,}24 \times 10^{-2}$	$1{,}80 \times 10^{0}$	$8{,}07 \times 10^{-3}$
$1{,}00 \times 10^{-3}$	$7{,}56 \times 10^{-2}$	$2{,}00 \times 10^{0}$	$1{,}34 \times 10^{-2}$
$3{,}00 \times 10^{-3}$	$5{,}09 \times 10^{-2}$	$2{,}30 \times 10^{0}$	$1{,}45 \times 10^{-2}$
$6{,}00 \times 10^{-3}$	$3{,}79 \times 10^{-2}$	$2{,}60 \times 10^{0}$	$1{,}49 \times 10^{-2}$
$1{,}00 \times 10^{-2}$	$5{,}47 \times 10^{-2}$	$3{,}00 \times 10^{0}$	$1{,}23 \times 10^{-2}$
$2{,}00 \times 10^{-2}$	$5{,}12 \times 10^{-2}$	$3{,}50 \times 10^{0}$	$8{,}19 \times 10^{-3}$
$4{,}00 \times 10^{-2}$	$2{,}96 \times 10^{-2}$	$4{,}00 \times 10^{0}$	$8{,}10 \times 10^{-3}$
$6{,}00 \times 10^{-2}$	$2{,}00 \times 10^{-2}$	$4{,}50 \times 10^{0}$	$6{,}54 \times 10^{-3}$
$8{,}00 \times 10^{-2}$	$1{,}45 \times 10^{-2}$	$5{,}00 \times 10^{0}$	$8{,}70 \times 10^{-3}$
$1{,}00 \times 10^{-1}$	$2{,}47 \times 10^{-2}$	$6{,}00 \times 10^{0}$	$4{,}93 \times 10^{-3}$
$1{,}50 \times 10^{-1}$	$1{,}59 \times 10^{-2}$	$7{,}00 \times 10^{0}$	$2{,}42 \times 10^{-3}$
$2{,}00 \times 10^{-1}$	$1{,}14 \times 10^{-2}$	$8{,}00 \times 10^{0}$	$1{,}30 \times 10^{-3}$
$2{,}50 \times 10^{-1}$	$8{,}90 \times 10^{-3}$	$9{,}00 \times 10^{0}$	$7{,}66 \times 10^{-3}$
$3{,}00 \times 10^{-1}$	$6{,}57 \times 10^{-3}$	$1{,}00 \times 10^{1}$	$4{,}43 \times 10^{-4}$
$3{,}50 \times 10^{-1}$	$4{,}89 \times 10^{-3}$	$1{,}10 \times 10^{1}$	$1{,}62 \times 10^{-4}$
$4{,}00 \times 10^{-1}$	$2{,}65 \times 10^{-3}$	$1{,}20 \times 10^{1}$	$1{,}24 \times 10^{-4}$
$4{,}50 \times 10^{-1}$	$3{,}14 \times 10^{-3}$	$1{,}30 \times 10^{1}$	$5{,}93 \times 10^{-5}$
$5{,}00 \times 10^{-1}$	$4{,}20 \times 10^{-3}$	$1{,}40 \times 10^{1}$	$2{,}83 \times 10^{-5}$
$5{,}50 \times 10^{-1}$	$4{,}12 \times 10^{-3}$	$1{,}50 \times 10^{1}$	
$6{,}00 \times 10^{-1}$	$7{,}83 \times 10^{-3}$		

Table A.2 — Values of group source strength for a ^{252}Cf spontaneous fission source

E_i(MeV)	B_i(s^{-1})	E_i(MeV)	B_i(s^{-1})
$4,14 \times 10^{-7}$	$3,10 \times 10^{-10}$	$7,00 \times 10^{-1}$	$3,39 \times 10^{-2}$
$1,00 \times 10^{-6}$	$1,11 \times 10^{-8}$	$8,00 \times 10^{-1}$	$3,37 \times 10^{-2}$
$1,00 \times 10^{-5}$	$1,27 \times 10^{-7}$	$9,00 \times 10^{-1}$	$3,33 \times 10^{-2}$
$5,00 \times 10^{-5}$	$2,76 \times 10^{-7}$	$1,00 \times 10^{0}$	$6,46 \times 10^{-2}$
$1,00 \times 10^{-4}$	$7,82 \times 10^{-7}$	$1,20 \times 10^{0}$	$6,12 \times 10^{-2}$
$2,00 \times 10^{-4}$	$2,21 \times 10^{-6}$	$1,40 \times 10^{0}$	$5,73 \times 10^{-2}$
$4,00 \times 10^{-4}$	$4,53 \times 10^{-6}$	$1,60 \times 10^{0}$	$5,31 \times 10^{-2}$
$7,00 \times 10^{-4}$	$5,68 \times 10^{-6}$	$1,80 \times 10^{0}$	$4,88 \times 10^{-2}$
$1,00 \times 10^{-3}$	$5,51 \times 10^{-5}$	$2,00 \times 10^{0}$	$6,55 \times 10^{-2}$
$3,00 \times 10^{-3}$	$1,28 \times 10^{-4}$	$2,30 \times 10^{0}$	$5,67 \times 10^{-2}$
$6,00 \times 10^{-3}$	$2,30 \times 10^{-4}$	$2,60 \times 10^{0}$	$6,33 \times 10^{-2}$
$1,00 \times 10^{-2}$	$7,74 \times 10^{-4}$	$3,00 \times 10^{0}$	$6,21 \times 10^{-2}$
$2,00 \times 10^{-2}$	$2,17 \times 10^{-3}$	$3,50 \times 10^{0}$	$4,68 \times 10^{-2}$
$4,00 \times 10^{-2}$	$2,80 \times 10^{-3}$	$4,00 \times 10^{0}$	$3,49 \times 10^{-2}$
$6,00 \times 10^{-2}$	$3,29 \times 10^{-3}$	$4,50 \times 10^{0}$	$2,58 \times 10^{-2}$
$8,00 \times 10^{-2}$	$3,68 \times 10^{-3}$	$5,00 \times 10^{0}$	$3,30 \times 10^{-2}$
$1,00 \times 10^{-1}$	$1,05 \times 10^{-2}$	$6,00 \times 10^{0}$	$1,74 \times 10^{-2}$
$1,50 \times 10^{-1}$	$1,21 \times 10^{-2}$	$7,00 \times 10^{0}$	$9,01 \times 10^{-3}$
$2,00 \times 10^{-1}$	$1,33 \times 10^{-2}$	$8,00 \times 10^{0}$	$4,61 \times 10^{-3}$
$2,50 \times 10^{-1}$	$1,42 \times 10^{-2}$	$9,00 \times 10^{0}$	$2,33 \times 10^{-3}$
$3,00 \times 10^{-1}$	$1,49 \times 10^{-2}$	$1,00 \times 10^{1}$	$1,17 \times 10^{-3}$
$3,50 \times 10^{-1}$	$1,55 \times 10^{-2}$	$1,10 \times 10^{1}$	$5,83 \times 10^{-4}$
$4,00 \times 10^{-1}$	$1,60 \times 10^{-2}$	$1,20 \times 10^{1}$	$2,88 \times 10^{-4}$
$4,50 \times 10^{-1}$	$1,63 \times 10^{-2}$	$1,30 \times 10^{1}$	$1,42 \times 10^{-4}$
$5,00 \times 10^{-1}$	$1,66 \times 10^{-2}$	$1,40 \times 10^{1}$	$6,94 \times 10^{-5}$
$5,50 \times 10^{-1}$	$1,68 \times 10^{-2}$	$1,50 \times 10^{1}$	
$6,00 \times 10^{-1}$	$3,38 \times 10^{-2}$		

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

E_i(MeV)	B_i(s^{-1})	E_i(MeV)	B_i(s^{-1})
$4,14 \times 10^{-7}$	$1,75 \times 10^{-2}$	$3,98 \times 10^{0}$	$1,98 \times 10^{-2}$
$8,20 \times 10^{-1}$	$1,13 \times 10^{-2}$	$4,13 \times 10^{0}$	$1,72 \times 10^{-2}$
$1,09 \times 10^{0}$	$8,07E - 03$	$4,27 \times 10^{0}$	$1,45 \times 10^{-2}$
$1,34E \times 10^{0}$	$2,10 \times 10^{-2}$	$4,41 \times 10^{0}$	$9,97 \times 10^{-3}$
$1,56E \times 10^{0}$	$4,54 \times 10^{-2}$	$4,55 \times 10^{0}$	$7,46 \times 10^{-3}$
$1,78 \times 10^{0}$	$6,31 \times 10^{-2}$	$4,69 \times 10^{0}$	$3,41 \times 10^{-3}$
$1,98 \times 10^{0}$	$7,99 \times 10^{-2}$	$4,83 \times 10^{0}$	$2,19 \times 10^{-3}$
$2,17 \times 10^{0}$	$8,90 \times 10^{-2}$	$4,96 \times 10^{0}$	$1,16 \times 10^{-3}$
$2,36 \times 10^{0}$	$9,26 \times 10^{-2}$	$5,09 \times 10^{0}$	$3,03 \times 10^{-4}$
$2,54 \times 10^{0}$	$9,65 \times 10^{-2}$	$5,22 \times 10^{0}$	$2,68 \times 10^{-4}$
$2,72 \times 10^{0}$	$8,3 \times 10^{-2}$	$5,35 \times 10^{0}$	$2,36 \times 10^{-4}$
$2,89 \times 10^{0}$	$7,06 \times 10^{-2}$	$5,48 \times 10^{0}$	$1,15 \times 10^{-4}$
$3,05 \times 10^{0}$	$6,67 \times 10^{-2}$	$5,61 \times 10^{0}$	$1,45 \times 10^{-4}$
$3,22 \times 10^{0}$	$4,99 \times 10^{-2}$	$5,74 \times 10^{0}$	$1,39 \times 10^{-4}$
$3,38 \times 10^{0}$	$4,02 \times 10^{-2}$	$5,86 \times 10^{0}$	$2,78 \times 10^{-4}$
$3,53 \times 10^{0}$	$3,17 \times 10^{-2}$	$5,98 \times 10^{0}$	$1,78 \times 10^{-4}$
$3,68 \times 10^{0}$	$3,03 \times 10^{-2}$	$6,11 \times 10^{0}$	$2,91 \times 10^{-4}$
$3,83 \times 10^{0}$	$2,52 \times 10^{-2}$	$6,19 \times 10^{0}$	

Table A.4 — Values of group source strength for a ^{241}Am-Be(α,n) source

E_i(MeV)	B_i(s^{-1})	E_i(MeV)	B_i(s^{-1})
$4{,}14 \times 10^{-7}$	$1{,}44 \times 10^{-2}$	$5{,}68 \times 10^{0}$	$2{,}06 \times 10^{-2}$
$1{,}10 \times 10^{-1}$	$3{,}34 \times 10^{-2}$	$5{,}89 \times 10^{0}$	$1{,}82 \times 10^{-2}$
$3{,}30 \times 10^{-1}$	$3{,}13 \times 10^{-2}$	$6{,}11 \times 10^{0}$	$1{,}77 \times 10^{-2}$
$5{,}40 \times 10^{-1}$	$2{,}81 \times 10^{-2}$	$6{,}32 \times 10^{0}$	$2{,}04 \times 10^{-2}$
$7{,}50 \times 10^{-1}$	$2{,}50 \times 10^{-2}$	$6{,}54 \times 10^{0}$	$1{,}83 \times 10^{-2}$
$9{,}70 \times 10^{-1}$	$2{,}14 \times 10^{-2}$	$6{,}75 \times 10^{0}$	$1{,}63 \times 10^{-2}$
$1{,}18 \times 10^{0}$	$1{,}98 \times 10^{-2}$	$6{,}96 \times 10^{0}$	$1{,}68 \times 10^{-2}$
$1{,}40 \times 10^{0}$	$1{,}75 \times 10^{-2}$	$7{,}18 \times 10^{0}$	$1{,}68 \times 10^{-2}$
$1{,}61 \times 10^{0}$	$1{,}92 \times 10^{-2}$	$7{,}39 \times 10^{0}$	$1{,}88 \times 10^{-2}$
$1{,}82 \times 10^{0}$	$2{,}23 \times 10^{-2}$	$7{,}61 \times 10^{0}$	$1{,}84 \times 10^{-2}$
$2{,}04 \times 10^{0}$	$2{,}15 \times 10^{-2}$	$7{,}82 \times 10^{0}$	$1{,}69 \times 10^{-2}$
$2{,}25 \times 10^{0}$	$2{,}25 \times 10^{-2}$	$8{,}03 \times 10^{0}$	$1{,}44 \times 10^{-2}$
$2{,}47 \times 10^{0}$	$2{,}28 \times 10^{-2}$	$8{,}25 \times 10^{0}$	$9{,}68 \times 10^{-3}$
$2{,}68 \times 10^{0}$	$2{,}95 \times 10^{-2}$	$8{,}46 \times 10^{0}$	$6{,}52 \times 10^{-3}$
$2{,}90 \times 10^{0}$	$3{,}56 \times 10^{-2}$	$8{,}68 \times 10^{0}$	$4{,}26 \times 10^{-3}$
$3{,}11 \times 10^{0}$	$3{,}69 \times 10^{-2}$	$8{,}89 \times 10^{0}$	$3{,}67 \times 10^{-3}$
$3{,}32 \times 10^{0}$	$3{,}46 \times 10^{-2}$	$9{,}11 \times 10^{0}$	$3{,}81 \times 10^{-3}$
$3{,}54 \times 10^{0}$	$3{,}07 \times 10^{-2}$	$9{,}32 \times 10^{0}$	$5{,}06 \times 10^{-3}$
$3{,}75 \times 10^{0}$	$3{,}00 \times 10^{-2}$	$9{,}53 \times 10^{0}$	$6{,}25 \times 10^{-3}$
$3{,}97 \times 10^{0}$	$2{,}69 \times 10^{-2}$	$9{,}75 \times 10^{0}$	$5{,}52 \times 10^{-3}$
$4{,}18 \times 10^{0}$	$2{,}86 \times 10^{-2}$	$9{,}96 \times 10^{0}$	$4{,}68 \times 10^{-3}$
$4{,}39 \times 10^{0}$	$3{,}18 \times 10^{-2}$	$1{,}02 \times 10^{1}$	$3{,}70 \times 10^{-3}$
$4{,}61 \times 10^{0}$	$3{,}07 \times 10^{-2}$	$1{,}04 \times 10^{1}$	$2{,}78 \times 10^{-3}$
$4{,}82 \times 10^{0}$	$3{,}33 \times 10^{-2}$	$1{,}06 \times 10^{1}$	$1{,}51 \times 10^{-3}$
$5{,}04 \times 10^{0}$	$3{,}04 \times 10^{-2}$	$1{,}08 \times 10^{1}$	$3{,}63 \times 10^{-4}$
$5{,}25 \times 10^{0}$	$2{,}74 \times 10^{-2}$	$1{,}10 \times 10^{1}$	
$5{,}47 \times 10^{0}$	$2{,}33 \times 10^{-2}$		

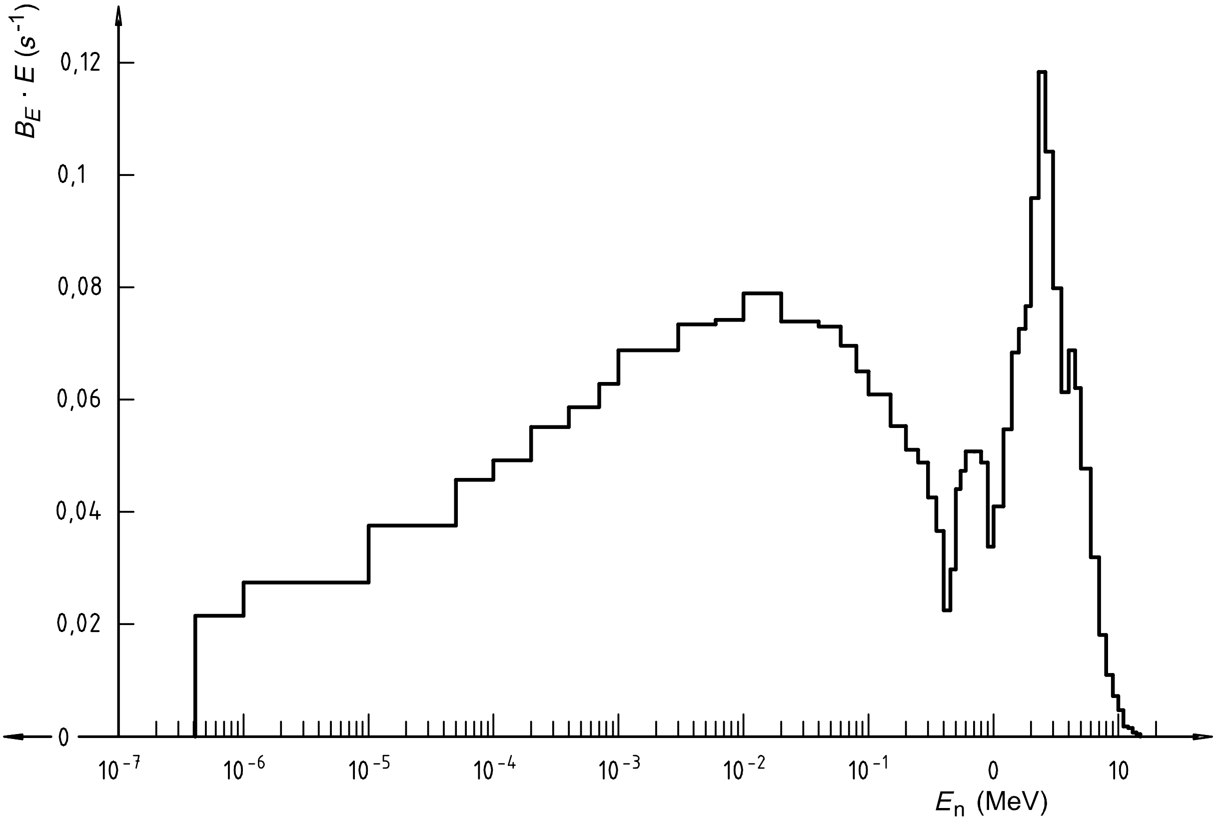

Figure A.1 — Neutron spectrum outside a D_2O sphere of radius 150 mm with a ^{252}Cf spontaneous fission neutron source at its centre

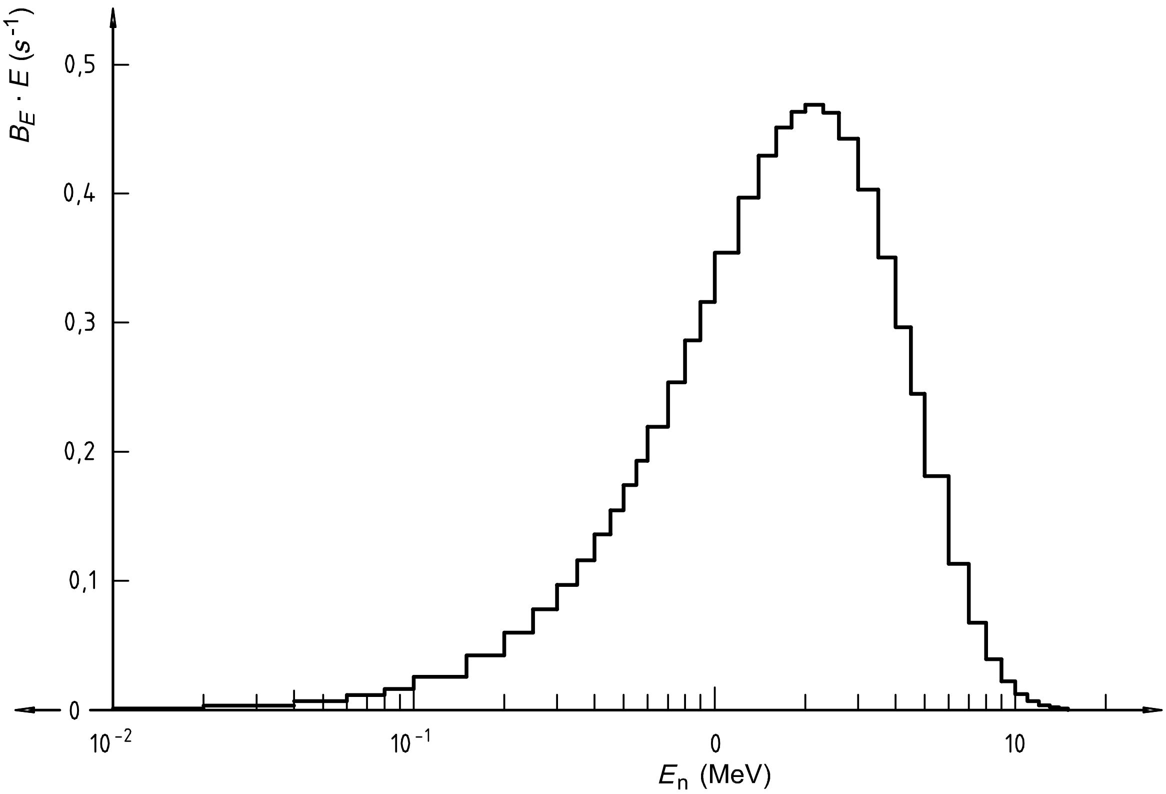

Figure A.2 — Neutron spectrum from a ^{252}Cf spontaneous fission source

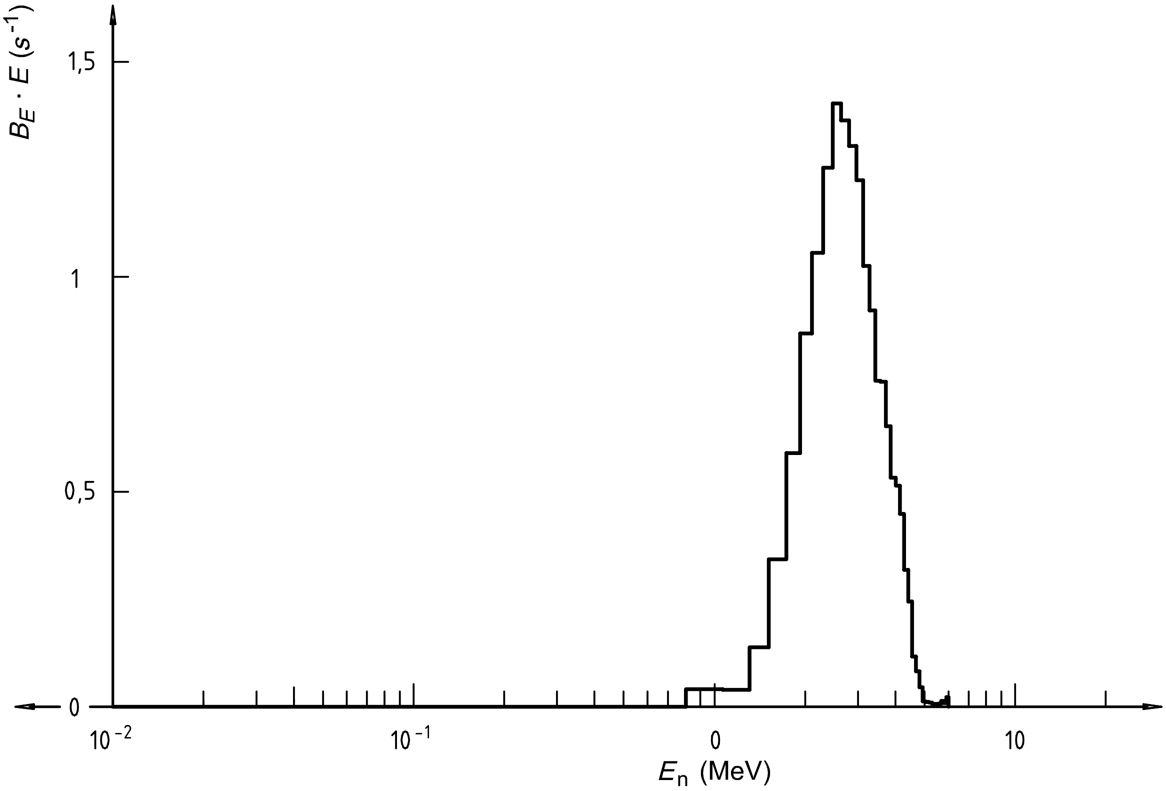

Figure A.3 — Neutron spectrum from a ^{241}AmB(α,n) source

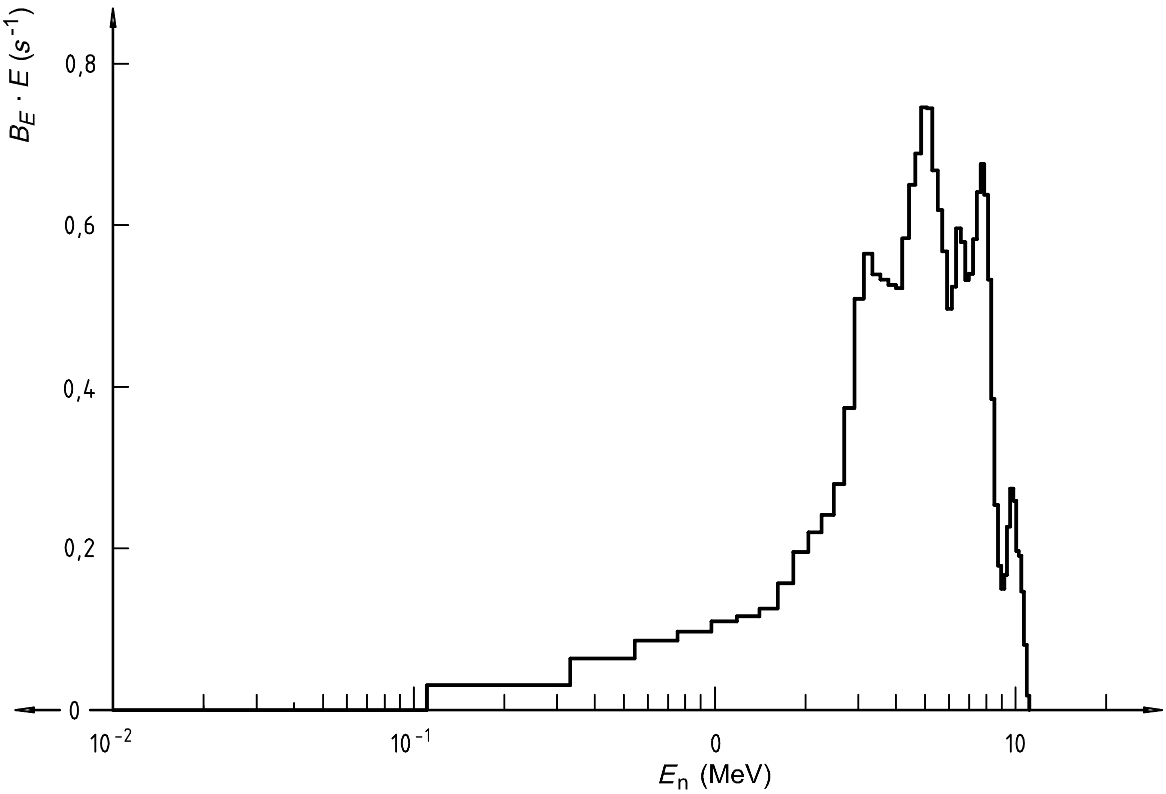

Figure A.4 — Neutron spectrum from a ^{241}Am-Be(α,n) source

Annex B
(informative)

Angular source strength characteristics of two radionuclide neutron sources

Radionuclide neutron sources generally show anisotropic emission in a coordinate system fixed in the geometrical centre of the source. The coordinate system is shown in 4.4, 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.

Dimensions in millimetres

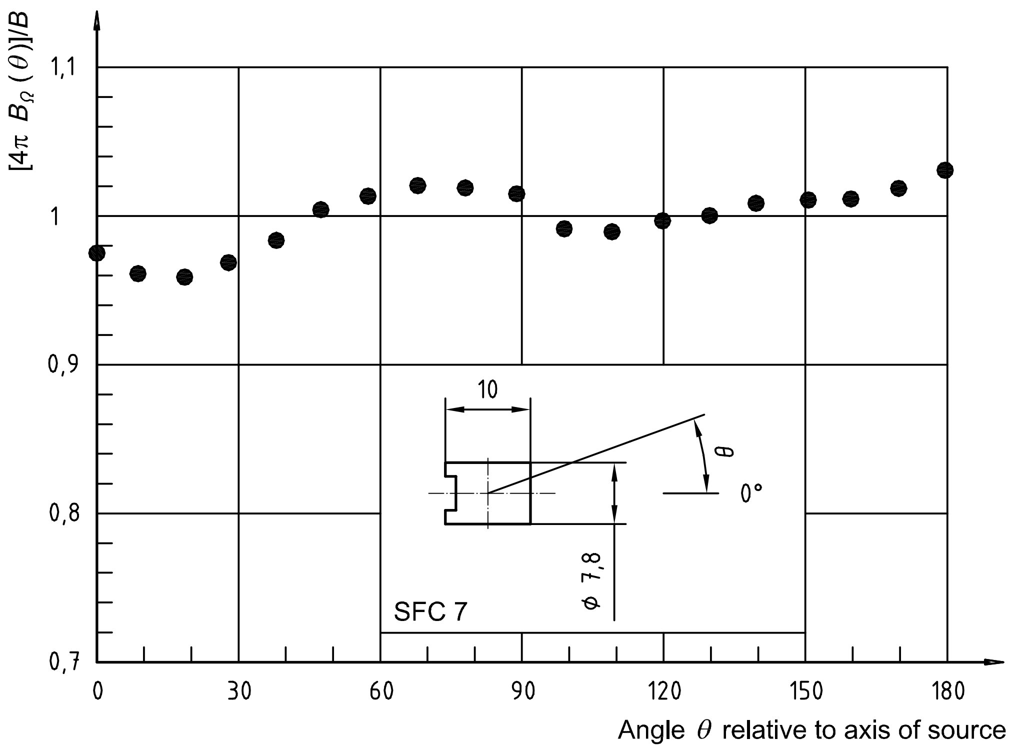

Figure B.1 — Angular source strength of a small size ^{252}Cf spontaneous fission source (approximate dimensions of active volume: 4,6 mm diameter by 6 mm long) normalized to the angular source strength $B/4\pi$ of the equivalent point source

Dimensions in millimetres

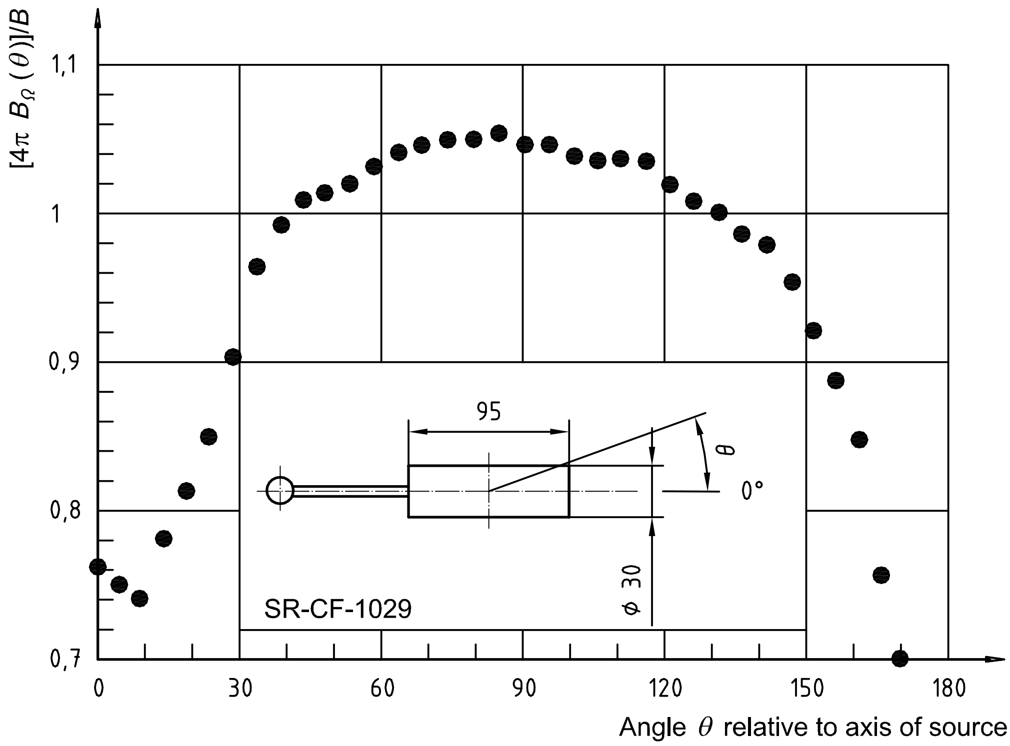

Figure B.2 — Angular source strength of a high intensity ^{252}Cf spontaneous fission source (approximate dimensions of active volume: 3 mm diameter by 30 mm long) normalized to the angular source strength $B/4\pi$ of the equivalent point source

Annex C
(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:

$$\varphi_0 = \int_0^{E_{\mathrm{Cd}}} \left(\frac{E_0}{E}\right)^{1/2} \varphi_E\,(E)\,\mathrm{d}E$$

where

E is the neutron energy;

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

φ_E is the spectral neutron fluence rate;

E_0 = 0,025 3 eV (v = 2 200 m·s^{-1}) is the reference energy for which cross-section a values σ_0 for 1/v detectors are tabulated.

The conventional thermal-neutron fluence rate is also given by

$$\varphi_0 = \frac{\dot{n}_{\mathrm{R}}}{\Sigma_0}$$

where

$\dot{n}_{\mathrm{R}}$ is the reaction rate density;

Σ_0 is the cross-section a density, given by:

$$\Sigma_0 = \rho \frac{p}{M} N_{\mathrm{A}} \sigma_0$$

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;

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 = 0,025 3 eV, the true thermal fluence rate, φ_{th}, is given by:

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

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

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