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Neutron reference radiations for calibrating neutron-measuring devices used for radiation protection purposes and for determining their response as a function of neutron energy

Rayonnements neutroniques de référence destinés à l'étalonnage des instruments de mesure des neutrons utilisés en radioprotection et à la détermination de leur réponse en fonction de l'énergie des neutrons

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Contents

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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 liaison 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 approval before their acceptance as International Standards by the ISO Council. They are approved in accordance with ISO procedures requiring at least 75 % approval by the member bodies voting.

International Standard ISO 8529 was prepared by Technical Committee ISO/TC 85, *Nuclear energy*.

Annexes A, B, C and D form an integral part of this International Standard. Annex E is for information only.

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Neutron reference radiations for calibrating neutron-measuring devices used for radiation protection purposes and for determining their response as a function of neutron energy

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 variation in response as a function of neutron energy. Reference radiations are given for neutron fluence rates of up to 10^5 $cm^{-2} \cdot s^{-1}$, corresponding, at a neutron energy of 1 MeV, to dose equivalent rates of up to 100 $mSv \cdot h^{-1}$ (10 $rem \cdot h^{-1}$). This International Standard applies to radiation protection calibrations in units of the quantity "dose equivalent", but values are also given in units of the quantities "absorbed dose" and "kerma" in "standard man tissue".

It should be noted that at the present time, the definitions of "dose equivalent" quantities to be used for radiation protection purposes are under review by both the ICRU and the ICRP[1]. The definitions of "dose equivalent" and the conversion factors from neutron fluence to dose equivalent given in this International Standard are therefore subject to possible revision.

This International Standard is concerned only with the methods of producing the neutron reference radiations. The procedures for applying these radiations will be described in a future International Standard.

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 clauses:

— In clause 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 clause 5, accelerator-produced monoenergetic neutrons, reactor-produced neutrons with wide and quasi-monoenergetic spectra, and special radionuclide sources 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 and related purposes, the following conversion factors are specified:

— "neutron fluence" to "dose equivalent";

— "neutron fluence" to "charged particle absorbed dose";

— "neutron fluence" to "photon absorbed dose";

— "neutron fluence" to "kerma".

The conversion factors given in annexes B and C are based on the spectra presented in annex A, and on the "fluence" to "dose" conversion factors referred to in 3.6, 3.8 and 3.11. This International Standard does not preclude the use of the reference radiations specified in this International Standard with other "fluence" to "dose" conversion factors, if applicable, in particular those obtained for a different phantom and/or dose equivalents defined at different positions in the phantom. At the present time, the "fluence" to "dose" conversion factors presented in this International Standard are the only internationally accepted values.

1) ICRU: International Commission on Radiation Units and Measurements
ICRP: International Commission on Radiological Protection

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 1677 : 1977, *Sealed radioactive sources — General.*

ISO 2919 : 1980, *Sealed radioactive sources — Classification.*

ICRP Publication 21, *Protection against Ionizing Radiation from External Sources,* 1973 edition. (Supplement to ICRP Publication 15.)

ICRU Report 26, *Neutron Dosimetry for Biology and Medicine,* 1977 edition.

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

3 Definitions of quantities and units

NOTES

1 The definitions follow the recommendations of ICRU Report 33.

2 Multiples and submultiples of SI units are also used throughout this International Standard.

3.1 neutron fluence, Φ: Quotient of dN by da, expressed in reciprocal square metres (m^{-2}), where dN is the number of neutrons incident on a sphere of cross-sectional area da:

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

3.2 neutron fluence rate; neutron flux density, ϕ: Quotient of dΦ by dt, expressed in reciprocal seconds reciprocal square metres ($s^{-1}.m^{-2}$), where dΦ is the increment of neutron fluence (see 3.1) in the time interval dt:

$$\phi = \frac{d\Phi}{dt} = \frac{d^2N}{da\,dt}$$

3.3 spectral distribution of the neutron fluence, Φ_E: Quotient of dΦ by dE, expressed in reciprocal joules reciprocal square metres ($J^{-1}.m^{-2}$), where dΦ is the increment of neutron fluence in the energy interval between E and E + dE:

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

NOTE — Another unit frequently used is reciprocal electronvolt reciprocal square centimetre ($eV^{-1}.cm^{-2}$).

3.4 spectral neutron fluence rate; spectral neutron flux density, ϕ_E: Quotient of dΦ_E by dt, expressed in reciprocal joules reciprocal square metres reciprocal seconds ($J^{-1}.m^{-2}.s^{-1}$), where dΦ_E is the increment of spectral distributions of the neutron fluence in the time interval dt:

$$\phi_E = \frac{d\Phi_E}{dt} = \frac{d^2\Phi}{dE\,dt}$$

NOTE — Another unit frequently used is reciprocal electronvolt reciprocal square centimetre reciprocal second ($eV^{-1}.cm^{-2}.s^{-1}$).

3.5 absorbed dose, D: Quotient of d$\bar{\varepsilon}$ by dm, expressed in grays (Gy)[1], where d$\bar{\varepsilon}$ is the mean energy imparted by ionizing radiation to matter of mass dm:

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

NOTE — The special unit of absorbed dose, rad, may be used temporarily; 1 rad = 10^{-2} Gy.

3.6 "neutron fluence" to "absorbed dose" conversion factor, d_Φ: Quotient of absorbed dose, D, and neutron fluence, Φ, expressed in grays square metres (Gy·m^2), at the point of reference undisturbed by the irradiated object:

$$d_\Phi = \frac{D}{\Phi}$$

Conversion factors are given in annexes B and C for the following two components:

— the heavy charged particle component of absorbed dose: d_Φ^c

— the neutron capture photon component of absorbed dose for $^1H(n,\gamma)\,^2D$: d_Φ^γ

It should be noted that, for neutron sources emitting gamma radiations, the total absorbed dose from photons will be given by the sum of the doses from incident gamma radiations and from neutron capture photons.

The values for the "fluence" to "absorbed dose" conversion factors given in this International Standard were derived using the analytical functions for d_Φ^c and d_Φ^γ [1].

These functions, based on the original calculations computed in [26], give the mean values for the components of absorbed dose in tissue for the volume element 57 of a cylindrical phantom (diameter 300 mm, height 600 mm) irradiated by a unidirectional broad beam of neutrons incident normally to the axis of the phantom. The phantom was considered to be composed of

1) 1 Gy = 1 J·kg^{-1}

hydrogen, carbon, nitrogen and oxygen in proportions of standard man. Volume element 57, of thickness 30 mm, is situated in the centre of the front surface of the phantom facing the neutron beam.

3.7 kerma, K: Quotient of dE_{tr} by dm, expressed in grays, where dE_{tr} is the sum of the initial kinetic energies of all the charged ionizing particles liberated by uncharged indirectly ionizing particles in a material of mass dm:

$$K = \frac{dE_{tr}}{dm}$$

NOTE — The special unit of kerma, rad, may be used temporarily; 1 rad = 10^{-2} Gy.

3.8 "neutron fluence" to "kerma" conversion factor, k_Φ: Quotient of kerma, K, and neutron fluence, Φ, expressed in grays square metres (Gy·m²), at the point of reference, undisturbed by the irradiated object:

$$k_\Phi = \frac{K}{\Phi}$$

NOTE — Conversion factors are given in annexes B and C. The values are for standard man tissue, in accordance with appendix A of ICRU Report 26.

3.9 dose equivalent, H: Product, expressed in sieverts (Sv)[1], at the point of interest in tissue, of the absorbed dose, D, the quality factor, Q, and the product of any other modifying factors, N:

$$H = DQN$$

NOTE — The special unit of dose equivalent, rem, may be used temporarily; 1 rem = 10^{-2} Sv.

3.10 dose equivalent rate, $\dot{H}$: Quotient of dH by dt, expressed in sieverts reciprocal seconds (Sv·s^{-1}), where dH is the increment of dose equivalent in the time interval dt:

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

NOTE — The special unit of dose equivalent rate, rem reciprocal second, may be used temporarily; 1 rem·s^{-1} = 10^{-2} Sv·s^{-1}.

3.11 "neutron fluence" to "dose equivalent" conversion factor, h_Φ: Quotient of the neutron dose equivalent, H, and the neutron fluence, Φ, expressed in sieverts square metres (Sv·m²), at the point of reference, undisturbed by the irradiated object:

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

Values of the "neutron fluence" to "dose equivalent" conversion factor in this International Standard (see also annexes B and C) are taken from ICRP Publication 21. These values refer to irradiation by a unidirectional broad beam of monoenergetic neutrons and are evaluated at the maxima of the depth-dose equivalent curves. The calculations have been mainly made in a 300 mm diameter, 600 mm high cylinder, equivalent to soft tissue, with the broad beam incident perpendicular to the cylinder axis.

3.12 exposure, X: Quotient of dQ by dm, expressed in coulombs reciprocal kilograms (C·kg^{-1}), where the value of dQ is the absolute value of the total charge of the ions of one sign produced in air when all the electrons (negatrons and positrons) liberated by photons in air of mass dm are completely stopped in air:

$$X = \frac{dQ}{dm}$$

NOTE — The special unit of exposure, röntgen (R), may be used temporarily; 1 R = 2,58 × 10^{-4} C·kg^{-1}.

3.13 exposure rate, $\dot{X}$: Quotient of dX by dt, expressed in coulombs reciprocal kilograms reciprocal seconds (C·kg^{-1}·s^{-1}), where dX is the increment of exposure in the time interval dt:

$$\dot{X} = \frac{dX}{dt} = \frac{d^2Q}{dm\,dt}$$

NOTE — The special unit of exposure rate, röntgen reciprocal second, may be used temporarily; 1 R·s^{-1} = 2,58 × 10^{-4} C·kg^{-1}·s^{-1}.

3.14 activity (of an amount of radioactive nuclide in a particular energy state at a given time), A: Quotient of dN^+ by dt, expressed in becquerels (Bq)[2], 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 special unit of activity, curie (Ci), may be used temporarily; 1 Ci = 3,7 × 10^{10} Bq.

3.15 neutron source strength (of a neutron source at a given time), B: Quotient of dN^* by dt, expressed in reciprocal

1) 1 Sv = 1 J·kg^{-1}

2) 1 Bq = 1 s^{-1}

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

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

3.16 angular source strength, B_Ω: In the case of a neutron source, the quotient of dB by $d\Omega$, expressed in reciprocal seconds reciprocal steradians ($s^{-1} \cdot sr^{-1}$), where dB is the number of neutrons propagating in a specified direction within the solid angle $d\Omega$:

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

3.17 spectral distribution of neutron source strength, B_E: Quotient of dB by dE, expressed in reciprocal joules reciprocal seconds ($J^{-1} \cdot s^{-1}$) [reciprocal electronvolts reciprocal seconds ($eV^{-1} \cdot s^{-1}$)], where dB is the increment of neutron source strength in the energy interval between E and $E + dE$:

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

The source strength B is derived from B_E as follows:

$$B = \int_0^\infty B_E \, dE$$

The spectral neutron fluence rate ϕ_E, due to neutrons emitted isotropically from a point source with a spectral neutron source strength B_E at a distance l (neglecting the influence of surrounding material), is given by (see also 3.4)

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

3.18 mean "neutron fluence" to "dose equivalent" conversion factor, $\bar{h}_\Phi$: In the case of a neutron source, the "neutron fluence" to "dose equivalent" conversion factor, h_Φ (see 3.11), averaged over the neutron source spectrum at the point of reference, undisturbed by the irradiated object:

$$\bar{h}_\Phi = \frac{1}{B} \int_0^\infty B_E \, h_\Phi(E) \, dE$$

For the purposes of this International Standard, the symbol $\bar{h}_\Phi$ is used for the mean conversion factor derived from the h_Φ values given in ICRP Publication 21.

3.19 dose equivalent average neutron energy, $\tilde{E}$: In the case of neutrons emitted from a neutron source, the neutron energy averaged over the dose equivalent spectrum at the point of reference. 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, at the point of reference and undisturbed by the irradiated object, and $h_\Phi(E)$ (see 3.11) is the "neutron fluence" to "dose equivalent" conversion factor at this energy:

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

where

$$H = \int_0^\infty h_\Phi(E) \Phi_E \, dE$$

The dose equivalent average neutron energy can be regarded as the neutron energy value of the centre of gravity of the dose equivalent spectrum.

3.20 response, R: In the case of a neutron-detecting instrument, the quotient

$$R = \frac{M}{G}$$

where

M is the value of the quantity indicated by the instrument or evaluated from its indication;

G is the quantity causing the instrument response. Generally, G is the quantity to be measured.

For the sake of clarity, the response may be specified as the response to this quantity, for example dose equivalent response R_H.

4 Reference radiations for the calibration of neutron-measuring devices

In this clause, reference radiations produced by radionuclide neutron sources are specified which are particularly suited for the calibration of neutron-measuring devices. It is generally not necessary to calibrate an instrument with all the listed reference radiations.

4.1 General properties

4.1.1 Type

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 specific dose equivalent rates vary with the construction of the source, because of scattering and absorption of neutrons and gamma radiations 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 fluence rate 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.

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}Am-Be(α,n) source, the spectral distribution,

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

Source[1)]	Half-life	Dose equivalent average energy[2)]	Specific source strength[3)]	Specific neutron dose equivalent rate at 1 m distance[4)]	Specific photon dose equivalent rate[5)] at 1 m distance[4)]
	a[6)]	MeV	$s^{-1} \cdot kg^{-1}$	$Sv \cdot s^{-1} \cdot kg^{-1}$	$Sv \cdot s^{-1} \cdot kg^{-1}$
$^{252}Cf(D_2O$ moderated)[7)] (sphere 300 mm in diameter)	2,65	2,2	$2,1 \times 10^{15}$	1,5	0,25
^{252}Cf	2,65	2,4	$2,4 \times 10^{15}$	6,5	0,31[8)]
	a	MeV	$s^{-1} \cdot Bq^{-1}$	$Sv \cdot s^{-1} \cdot Bq^{-1}$	$Sv \cdot s^{-1} \cdot Bq^{-1}$
^{241}Am-B(α,n)	432	2,8	$1,6 \times 10^{-5}$	5×10^{-20}	$1,9 \times 10^{-19}$
^{241}Am-Be(α,n)	432	4,4	$6,6 \times 10^{-5}$	2×10^{-19}	$1,9 \times 10^{-19}$

1) In addition to the sources listed, 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.

2) Neutron spectra of sources are given in figures A.1 to A.4. Definition of the dose equivalent average energy is given in 3.19.

3) The specific source strength, the specific neutron dose equivalent rate and the specific photon dose equivalent rate are the respective quantities related to the mass of 1 kg or the source activity of 1 Bq. Information on the sources is given for moderated ^{252}Cf in references [1, 2 and 3], for ^{252}Cf in [4], for ^{241}Am-B in [5], and for ^{241}Am-Be in [6].

4) For ^{252}Cf sources, this is related to the mass of californium contained in the source; for the other sources, this is related to the activity of the ^{241}Am contained in the source.

5) Conversion of exposure to dose equivalent was performed using the factor 0,01 $Sv \cdot R^{-1}$.

6) 1 a = 1 mean solar year = 31 556 926 s or 365,242 20 days.

7) Heavy-water sphere with a diameter of 300 mm covered with a cadmium shell of thickness approximately 1 mm.

8) For approximately 2,5 mm thick steel encapsulation.

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 laid down in ISO 1677 and 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 gamma radiations 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 sources for routine calibrations

4.2.1 Types

Preferably ^{252}Cf spontaneous fission and/or ^{241}Am-Be(α,n) sources should be used for routine calibrations. ^{252}Cf sources generally have a high specific source strength and are therefore comparatively small. The americium-based neutron sources shall consist of a homogeneous, compressed mixture of americium oxide and beryllium or boron as appropriate. Americium alloys may also be used.

4.2.2 Spectral distribution of neutron source strength

The spectral distributions of neutron source strength for ^{252}Cf, ^{241}Am-Be(α,n), $^{252}Cf(D_2O$ 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 spectral distribution of the neutron source strength, B_E, of ^{252}Cf can be described in the energy range from 100 keV to 10 MeV 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.)

4.2.3 "Neutron fluence" to "dose equivalent" conversion factors

The dose equivalent for the ^{252}Cf, ^{241}Am-Be(α,n), $^{252}Cf(D_2O$ moderated) and ^{241}Am-B(α,n) sources shall be calculated from the fluence using the values of the mean "neutron fluence" to "dose equivalent" conversion factor, $\overline{h}_\Phi$, given in annex B.

4.3 Neutron fluence rate produced by a source

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

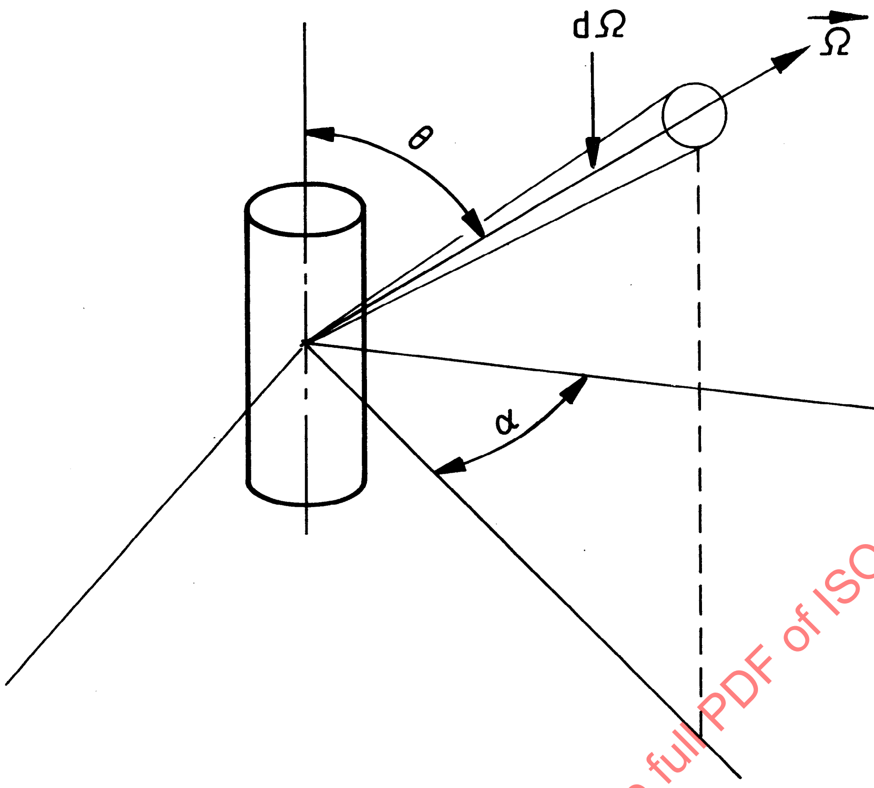

Figure 1 — Coordinate system for the case of an anisotropically emitting source (see 4.3)

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 $\theta = 90°$, this direction should be used.

The neutron source strength, B, and the angular source strength, $dB/d\Omega$, for $\theta = 90°$ shall be determined by a reference laboratory.

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 centre of the source in a direction for which $\theta = 90°$ then may be taken as

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

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, will be described in detail in a future International Standard on calibration procedures.

4.4 Calibration of the neutron source strength

The ^{241}Am-Be(α,n), ^{241}Am-B(α,n) and ^{252}Cf 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 an uncertainty[1)] of about ± 1,5 %.

There is the possibility, however, that, with time, the constituent components of the americium-beryllium and americium-boron 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.

1) This uncertainty, and all others given in this International Standard, are of one standard deviation.

The source strength of a ^{252}Cf source shall be corrected for radioactive decay on a day-to-day basis. At the present time the 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 an 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 will be dealt with in a future International Standard.

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 clause may also be used for the routine calibration of neutron-measuring devices.

Since these reference radiations are available only at specialized laboratories, only the general principles on their method of production 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. A radionuclide source with a narrow energy distribution of neutrons is included.

Table 2 — Neutron radiations for determining the response of neutron-measuring devices as a function of neutron energy1)

Neutron energy MeV	Method of production	Reference (see annex E)
$2,5 \times 10^{-8}$ (thermal)1)	Moderated-reactor or accelerator-produced neutrons	[10]; [7]
0,000 5	Sb-Be(γ,n), radionuclide source, water-moderated	[8]
0,002	Scandium-filtered reactor neutron beam or accelerator-produced neutrons from reaction $^{45}Sc(p,n)\,^{45}Ti$	[9]; [10]
0,021	Sb-Be(γ,n) radionuclide source	[11]; [12]
0,024	Iron/aluminium-filtered reactor neutron beam or accelerator-produced neutrons from reaction $^{45}Sc(p,n)\,^{45}Ti$	[9]; [10]; [13]
0,144 1)	Silicon-filtered reactor neutron beam or accelerator-produced neutrons from reactions $T(p,n)\,^{3}He$ and $^{7}Li(p,n)\,^{7}Be$	[9]; [14]; [15]; [16]
0,25 1)	Accelerator-produced neutrons from reactions $T(p,n)\,^{3}He$ and $^{7}Li(p,n)\,^{7}Be$	[14]; [15]; [16]
0,565 1)	Accelerator-produced neutrons from reactions $T(p,n)\,^{3}He$ and $^{7}Li(p,n)\,^{7}Be$	[14]; [15]; [16]
1,2	Accelerator-produced neutrons from reaction $T(p,n)\,^{3}He$	[14]; [15]; [16]
2,5 1)	Accelerator-produced neutrons from reaction $T(p,n)\,^{3}He$	[14]; [15]; [16]
2,8 2)	Accelerator-produced neutrons from reaction $D(d,n)\,^{3}He$	[14]; [15]; [16]
5,0	Accelerator-produced neutrons from reaction $D(d,n)\,^{3}He$	[14]; [15]; [16]
14,8 1) 2)	Accelerator-produced neutrons from reaction $T(d,n)\,^{4}He$	[14]; [15]; [16]
19,0	Accelerator-produced neutrons from reaction $T(d,n)\,^{4}He$	[14]; [15]; [16]

1) Energies at which international intercomparisons of neutron fluence measurements were performed [17].

2) Accelerator-produced neutrons, with a deuteron energy of a few hundred kiloelectronvolts.

5.2 Reactor reference neutrons

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 [18] 1).

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 [19]), are called "thermal" 2). The "true thermal neutron fluence rate", ϕ_{th}, is the required quantity from which the dose equivalent rate may be derived using the conversion factor h_Φ given in annex C.

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 D), as defined in [19] and measured, for example, by the activation of gold foils [20].

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 [20].

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.

Suitable methods for the measurement of neutron fluence rate include activation of gold foils, the use of boron trifluoride or ^{3}He proportional counters, and ^{235}U fission chambers [20].

5.2.3 Filtered neutron beams from a reactor [9, 10 and 21]

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 and 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 $E_n \leqslant 24$ keV) and a recoil proton counter for higher energy beams (neutron energies of $E_n \geqslant 24$ keV). Boron trifluoride proportional counters or ^{3}He proportional counters may be used as monitors and transfer instruments.

5.3 Photoneutron sources

5.3.1 Antimony-beryllium source

The ^{124}Sb-Be(γ,n) source is used for producing low-energy neutrons [11 and 12]. It utilizes the 1,691 0 MeV gamma radiation which is emitted after ^{124}Sb β-decay to ^{124}Te, with a half-life of 60,2 days ± 0,03 days. The resulting energy of the neutrons arising from the beryllium target material is calculated to be 22,8 keV ± 0,3 keV. However, the neutron spectrum is not strictly monoenergetic due to differences in the angle between the incident photon and the emitted neutron, θ_n, which leads to an energy distribution according to (22,8 + 1,3 cos θ_n) keV. It should be noted that problems may arise due to its characteristics being very dependent of the design of the source.

Furthermore, in practice, there is some energy degradation due to multiple scattering of the gamma radiations and neutrons in the source materials and its encapsulation. Monte-Carlo calculations for spherical sources, with all the scattering processes taken into account, indicate that the average neutron energy is between 21 keV and 21,5 keV. No similar calculations have been published for commercially available cylindrical sources.

A second neutron group has to be taken into account due the 2,091 MeV gamma radiation emitted in the decay of ^{124}Sb. The energy of the associated neutron group is approximately 378 keV. From the knowledge of the ^{124}Sb decay scheme and the ^{9}Be(γ,n)^{8}Be, photoneutron cross-section, it follows that the fluence rate of this high energy group, relative to the 22,8 keV group, is 0,03 ± 0,008.

Recommended values of the mean "neutron fluence" to "dose equivalent" conversion factors $\bar{h}_\Phi$ are as follows:

$\bar{h}_\Phi(1) = 1,7 \times 10^{-11}$ Sv·cm^2 (for the 21,5 keV group)

$\bar{h}_\Phi(T) = 2,1 \times 10^{-11}$ Sv·cm^2 (for the total neutron spectrum of the source)

Although the high-energy group (378 keV) contributes only 3 % to the total fluence, its contribution to the total neutron dose equivalent is nearly 23 % [12]. The photon dose equivalent rate is approximately 10^4 times the neutron dose equivalent rate.

Some properties of an Sb-Be(γ,n) source have been described in [11].

1) This subject will be dealt with in a future International Standard.

2) This definition also includes a small contribution of epithermal neutrons below the cadmium cut-off energy [19].

5.3.2 Moderated antimony-beryllium source

A source with a broad spectrum of low-energy neutrons around 0,5 keV may be produced by moderating the neutrons of an Sb-Be(γ,n) source with light water (shell of thickness 40 mm surrounded by the equivalent of 1 mm of ^{10}B [8]). Three independent Monte-Carlo calculations of the neutron spectrum of a practically realized source all indicate the same spectral shape. The emission rate of the moderator assembly was calculated and measured with a manganese bath and was found to be 18 % of the bare Sb-Be(γ,n) source.

There is also a high-energy neutron group which originates from the 378 keV secondary neutron group emitted by the unmoderated source. This is approximately centred around 200 keV. Recommended values of the mean "neutron fluence" to "dose equivalent" conversion factors, $\bar{h}_\Phi$, are as follows:

$\bar{h}_\Phi(1) = 1,1 \times 10^{-11}$ Sv.cm^2 (for the 0,5 keV primary group)

$\bar{h}_\Phi(T) = 1,4 \times 10^{-11}$ Sv.cm^2 (for the total neutron spectrum of the source)

The influence of the high-energy neutron group and the fact that the photon dose equivalent rate is approximately 10^5 times higher than the neutron dose equivalent rate have to be considered [22 and 23].

5.4 Accelerator-produced neutrons

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 [4]. 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.

5.4.2 Energy of charged particles

Details on the reaction kinematics determining the neutron energy and the corresponding charged particle energy are given in [14 and 15].

The energy of the incident charged particle beam should be determined. A stabilized 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 hydrogen in different materials are given in [24].

5.4.3 Neutron spectrum

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.

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 4.5), 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 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)

Graphical and tabular representation of the neutron spectra for radionuclide sources

The spectra are represented as plots of $\Delta B_0/\Delta \ln(E/E_o)$ (linear scale) versus neutron energy E (logarithmic scale) with the ΔB_0 group source strength normalized to source strength $B_0 = 1\ s^{-1}$ and E_o taken as 1 MeV (see figures A.1 to A.4).

For each neutron energy group, E_{lo} and E_{hi} being the lower and higher boundaries, a value of $\Delta B_0/\Delta \ln(E/E_o) = \Delta B_0/\ln(E_{hi}/E_{lo})$ was calculated. ΔB_0 was obtained by numerical integration as follows:

— by using the analytical function given in 4.2.2, in the case of the unmoderated ^{252}Cf spontaneous fission source;

— by using the experimental data of the spectra in the case of the ^{241}Am-B(α,n) and ^{241}Am-Be(α,n) sources. 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 $E = 0$ MeV. The graphs only give the experimentally determined spectra;

— the ^{252}Cf(D_2O moderated) spectrum was obtained by Monte-Carlo calculation at a distance > 15 cm from the surface of the source [2].

In tables A.1 to A.4, the energy associated with a value of $\Delta B_0/\ln(E_{hi}/E_o)$ corresponds to the upper limit of the particular interval. This energy also serves as the lower boundary of the next higher energy interval and so on.

The spectral data given in tables A.1 to A.4 has been calculated for unit source strength. Hence, for the Am-Be(α,n), Am-B(α,n) and ^{252}Cf sources, the integrals of the source spectra are equal to unity. However, for the ^{252}Cf(D_2O moderated) source, 11,5 % of the moderated source neutrons are absorbed by the cadmium shell; hence, for the ^{252}Cf(D_2O moderated) source, the spectrum sums to 0,885.

For a ^{241}Am-Be(α,n) source, the fraction of low-energy neutrons generally increases with the source strength and the correspondingly larger dimensions of the source, but the influence on the mean "fluence" to "dose equivalent" conversion factor is small because the conversion factor decreases for lower energy neutrons. The mean conversion factor, $\overline{h}_\Phi$, is $3{,}81 \times 10^{-10}$ Sv·cm^2 for the theoretical spectrum [25] and $3{,}75 \times 10^{-10}$ Sv·cm^2 for the spectrum given in figure A.1, which was measured for a cylindrical source (25,2 mm × 25,2 mm) having a source strength of $3{,}14 \times 10^{-6}\ s^{-1}$ [6]. For the ^{241}Am-Be(α,n) neutron spectrum given in figure A.2, the contribution to the total source strength of the neutrons with energies below 1,5 MeV is 18 % whereas the relative contribution of these neutrons to the maximum dose equivalent in a 300 mm diameter, 600 mm high cylinder of tissue equivalent material is 13 %.

Table A.1 — Values of group source strength per logarithmic energy interval for a ^{252}Cf spontaneous fission source

Neutron energy, E	$\frac{\Delta B_0}{\Delta \ln(E/E_0)}$
MeV	s^{-1}
$4,14 \times 10^{-7}$	
0,01	$4,40 \times 10^{-5}$
0,05	$2,74 \times 10^{-3}$
0,10	$1,24 \times 10^{-2}$
0,20	$3,33 \times 10^{-2}$
0,25	$6,04 \times 10^{-2}$
0,30	$7,90 \times 10^{-2}$
0,40	$1,07 \times 10^{-1}$
0,50	$1,46 \times 10^{-1}$
0,60	$1,84 \times 10^{-1}$
0,70	$2,21 \times 10^{-1}$
0,80	$2,55 \times 10^{-1}$
1,00	$3,01 \times 10^{-1}$
1,20	$3,53 \times 10^{-1}$
1,40	$3,95 \times 10^{-1}$
1,50	$4,19 \times 10^{-1}$
1,60	$4,32 \times 10^{-1}$
1,80	$4,46 \times 10^{-1}$
2,00	$4,58 \times 10^{-1}$
2,20	$4,62 \times 10^{-1}$
2,30	$4,61 \times 10^{-1}$
2,40	$4,59 \times 10^{-1}$
2,60	$4,53 \times 10^{-1}$
2,80	$4,42 \times 10^{-1}$
3,00	$4,27 \times 10^{-1}$
3,40	$4,01 \times 10^{-1}$
3,70	$3,66 \times 10^{-1}$
4,20	$3,25 \times 10^{-1}$
4,60	$2,78 \times 10^{-1}$
5,00	$2,39 \times 10^{-1}$
5,50	$1,99 \times 10^{-1}$
6,00	$1,61 \times 10^{-1}$
6,50	$1,28 \times 10^{-1}$
7,00	$1,01 \times 10^{-1}$
7,50	$7,92 \times 10^{-2}$
8,00	$6,16 \times 10^{-2}$
8,50	$4,76 \times 10^{-2}$
9,00	$3,65 \times 10^{-2}$
9,50	$2,79 \times 10^{-2}$
10,00	$2,13 \times 10^{-2}$
11,00	$1,42 \times 10^{-2}$
12,00	$8,04 \times 10^{-3}$
13,00	$4,51 \times 10^{-3}$
14,00	$2,50 \times 10^{-3}$
16,00	$1,08 \times 10^{-3}$
18,00	$3,20 \times 10^{-4}$

Table A.2 — Values of group source strength per logarithmic energy interval for a ^{241}Am-Be(α,n) source

Neutron energy, E	$\frac{\Delta B_0}{\Delta \ln(E/E_0)}$
MeV	s^{-1}
$4,14 \times 10^{-7}$	
0,11	$1,15 \times 10^{-3}$
0,33	$3,04 \times 10^{-2}$
0,54	$6,35 \times 10^{-2}$
0,75	$8,56 \times 10^{-2}$
0,97	$9,72 \times 10^{-2}$
1,18	$1,09 \times 10^{-1}$
1,40	$1,16 \times 10^{-1}$
1,61	$1,25 \times 10^{-1}$
1,82	$1,57 \times 10^{-1}$
2,04	$1,95 \times 10^{-1}$
2,25	$2,19 \times 10^{-1}$
2,47	$2,41 \times 10^{-1}$
2,68	$2,79 \times 10^{-1}$
2,90	$3,74 \times 10^{-1}$
3,11	$5,09 \times 10^{-1}$
3,32	$5,64 \times 10^{-1}$
3,54	$5,39 \times 10^{-1}$
3,75	$5,32 \times 10^{-1}$
3,97	$5,26 \times 10^{-1}$
4,18	$5,22 \times 10^{-1}$
4,39	$5,84 \times 10^{-1}$
4,61	$6,50 \times 10^{-1}$
4,82	$6,90 \times 10^{-1}$
5,04	$7,47 \times 10^{-1}$
5,25	$7,45 \times 10^{-1}$
5,47	$6,67 \times 10^{-1}$
5,68	$6,19 \times 10^{-1}$
5,89	$5,67 \times 10^{-1}$
6,11	$4,95 \times 10^{-1}$
6,32	$5,23 \times 10^{-1}$
6,54	$5,96 \times 10^{-1}$
6,75	$5,79 \times 10^{-1}$
6,96	$5,32 \times 10^{-1}$
7,18	$5,39 \times 10^{-1}$
7,39	$5,83 \times 10^{-1}$
7,61	$6,42 \times 10^{-1}$
7,82	$6,75 \times 10^{-1}$
8,03	$6,37 \times 10^{-1}$
8,25	$5,31 \times 10^{-1}$
8,46	$3,85 \times 10^{-1}$
8,68	$2,54 \times 10^{-1}$
8,89	$1,78 \times 10^{-1}$
9,11	$1,50 \times 10^{-1}$
9,32	$1,67 \times 10^{-1}$
9,53	$2,27 \times 10^{-1}$
9,75	$2,74 \times 10^{-1}$
9,96	$2,59 \times 10^{-1}$
10,18	$2,14 \times 10^{-1}$
10,39	$1,81 \times 10^{-1}$
10,60	$1,39 \times 10^{-1}$
10,82	$7,37 \times 10^{-2}$
11,03	$1,89 \times 10^{-2}$
11,09	0

Table A.3 — Values of a group source strength per logarithmic energy interval for a ^{252}Cf spontaneous fission source in the centre of a D_2O sphere with a radius of 150 mm

Neutron energy, E	$\frac{\Delta B_0}{\Delta \ln(E/E_0)}$
MeV	s^{-1}
$4{,}14 \times 10^{-7}$	
$1{,}0 \times 10^{-6}$	$2{,}15 \times 10^{-2}$
$1{,}0 \times 10^{-5}$	$2{,}74 \times 10^{-2}$
$5{,}0 \times 10^{-5}$	$3{,}75 \times 10^{-2}$
$1{,}0 \times 10^{-4}$	$4{,}57 \times 10^{-2}$
$2{,}0 \times 10^{-4}$	$4{,}92 \times 10^{-2}$
$4{,}0 \times 10^{-4}$	$5{,}51 \times 10^{-2}$
$7{,}0 \times 10^{-4}$	$5{,}86 \times 10^{-2}$
$1{,}0 \times 10^{-3}$	$6{,}29 \times 10^{-2}$
$3{,}0 \times 10^{-3}$	$6{,}88 \times 10^{-2}$
$6{,}0 \times 10^{-3}$	$7{,}34 \times 10^{-2}$
$1{,}0 \times 10^{-2}$	$7{,}42 \times 10^{-2}$
$2{,}0 \times 10^{-2}$	$7{,}89 \times 10^{-2}$
$4{,}0 \times 10^{-2}$	$7{,}38 \times 10^{-2}$
$6{,}0 \times 10^{-2}$	$7{,}30 \times 10^{-2}$
$8{,}0 \times 10^{-2}$	$6{,}95 \times 10^{-2}$
$1{,}0 \times 10^{-1}$	$6{,}52 \times 10^{-2}$
$1{,}5 \times 10^{-1}$	$6{,}10 \times 10^{-2}$
$2{,}0 \times 10^{-1}$	$5{,}54 \times 10^{-2}$
$2{,}5 \times 10^{-1}$	$5{,}12 \times 10^{-2}$
$3{,}0 \times 10^{-1}$	$4{,}88 \times 10^{-2}$
$3{,}5 \times 10^{-1}$	$4{,}26 \times 10^{-2}$
$4{,}0 \times 10^{-1}$	$3{,}66 \times 10^{-2}$
$4{,}5 \times 10^{-1}$	$2{,}25 \times 10^{-2}$
$5{,}0 \times 10^{-1}$	$2{,}98 \times 10^{-2}$
$5{,}5 \times 10^{-1}$	$4{,}41 \times 10^{-2}$
$6{,}0 \times 10^{-1}$	$4{,}73 \times 10^{-2}$
$7{,}0 \times 10^{-1}$	$5{,}08 \times 10^{-2}$
$8{,}0 \times 10^{-1}$	$5{,}08 \times 10^{-2}$
$9{,}0 \times 10^{-1}$	$4{,}88 \times 10^{-2}$
1,0	$3{,}39 \times 10^{-2}$
1,2	$4{,}10 \times 10^{-2}$
1,4	$5{,}47 \times 10^{-2}$
1,6	$6{,}84 \times 10^{-2}$
1,8	$7{,}26 \times 10^{-2}$
2,0	$7{,}66 \times 10^{-2}$
2,3	$9{,}57 \times 10^{-2}$
2,6	$1{,}18 \times 10^{-1}$
3,0	$1{,}04 \times 10^{-1}$
3,5	$8{,}01 \times 10^{-2}$
4,0	$6{,}13 \times 10^{-2}$
4,5	$6{,}88 \times 10^{-2}$
5,0	$6{,}21 \times 10^{-2}$
6,0	$4{,}77 \times 10^{-2}$
7,0	$3{,}20 \times 10^{-2}$
8,0	$1{,}81 \times 10^{-2}$
9,0	$1{,}10 \times 10^{-2}$
10,0	$7{,}27 \times 10^{-3}$
11,0	$4{,}65 \times 10^{-3}$
12,0	$1{,}86 \times 10^{-3}$
13,0	$1{,}55 \times 10^{-3}$
14,0	$8{,}00 \times 10^{-4}$
15,0	$4{,}10 \times 10^{-4}$

Table A.4 — Values of group source strength per logarithmic energy interval for a ^{241}Am-B(α,n) source

Neutron energy, E	$\frac{\Delta B_0}{\Delta \ln(E/E_0)}$
MeV	s^{-1}
$4{,}14 \times 10^{-7}$	
0,82	$1{,}21 \times 10^{-3}$
1,09	$3{,}97 \times 10^{-2}$
1,34	$3{,}91 \times 10^{-2}$
1,56	$1{,}38 \times 10^{-1}$
1,78	$3{,}44 \times 10^{-1}$
1,98	$5{,}93 \times 10^{-1}$
2,17	$8{,}72 \times 10^{-1}$
2,36	1,06
2,54	1,26
2,72	1,41
2,89	1,37
3,05	1,31
3,22	1,23
3,38	1,03
3,53	$9{,}26 \times 10^{-1}$
3,68	$7{,}62 \times 10^{-1}$
3,83	$7{,}59 \times 10^{-1}$
3,98	$6{,}57 \times 10^{-1}$
4,13	$5{,}35 \times 10^{-1}$
4,27	$5{,}17 \times 10^{-1}$
4,41	$4{,}49 \times 10^{-1}$
4,55	$3{,}19 \times 10^{-1}$
4,69	$2{,}46 \times 10^{-1}$
4,83	$1{,}16 \times 10^{-1}$
4,96	$8{,}26 \times 10^{-2}$
5,09	$4{,}49 \times 10^{-2}$
5,22	$1{,}20 \times 10^{-2}$
5,35	$1{,}09 \times 10^{-2}$
5,48	$9{,}83 \times 10^{-3}$
5,61	$4{,}92 \times 10^{-3}$
5,74	$6{,}34 \times 10^{-3}$
5,86	$6{,}74 \times 10^{-3}$
5,98	$1{,}37 \times 10^{-2}$
6,11	$8{,}28 \times 10^{-3}$
6,19	$2{,}24 \times 10^{-2}$
6,25	0

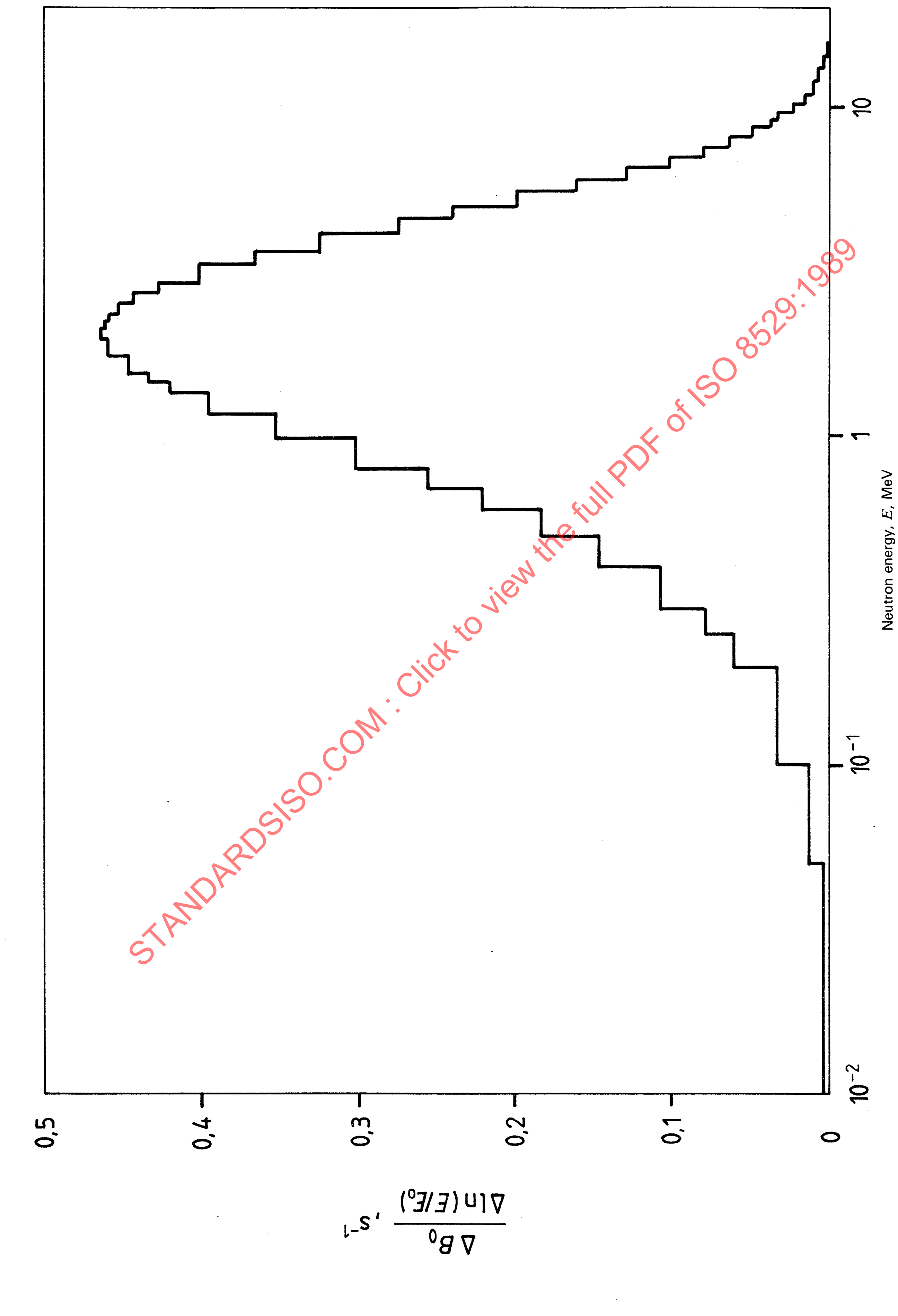

Figure A.1 — Neutron spectrum from a ^{252}Cf spontaneous fission source (see 4.2.2 and [4])

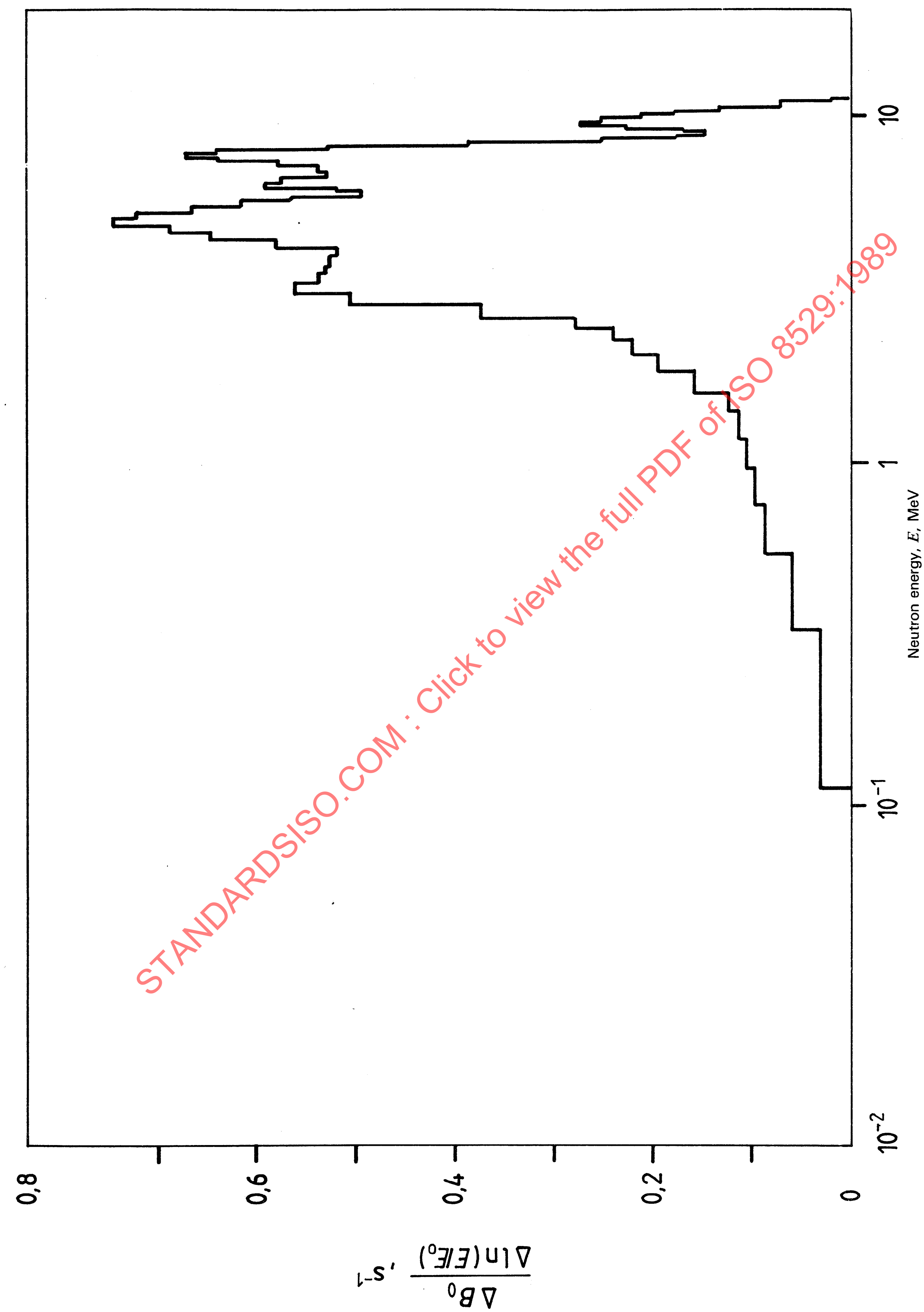

Figure A.2 — Neutron spectrum from a ^{241}Am-Be(α,n) source [6]

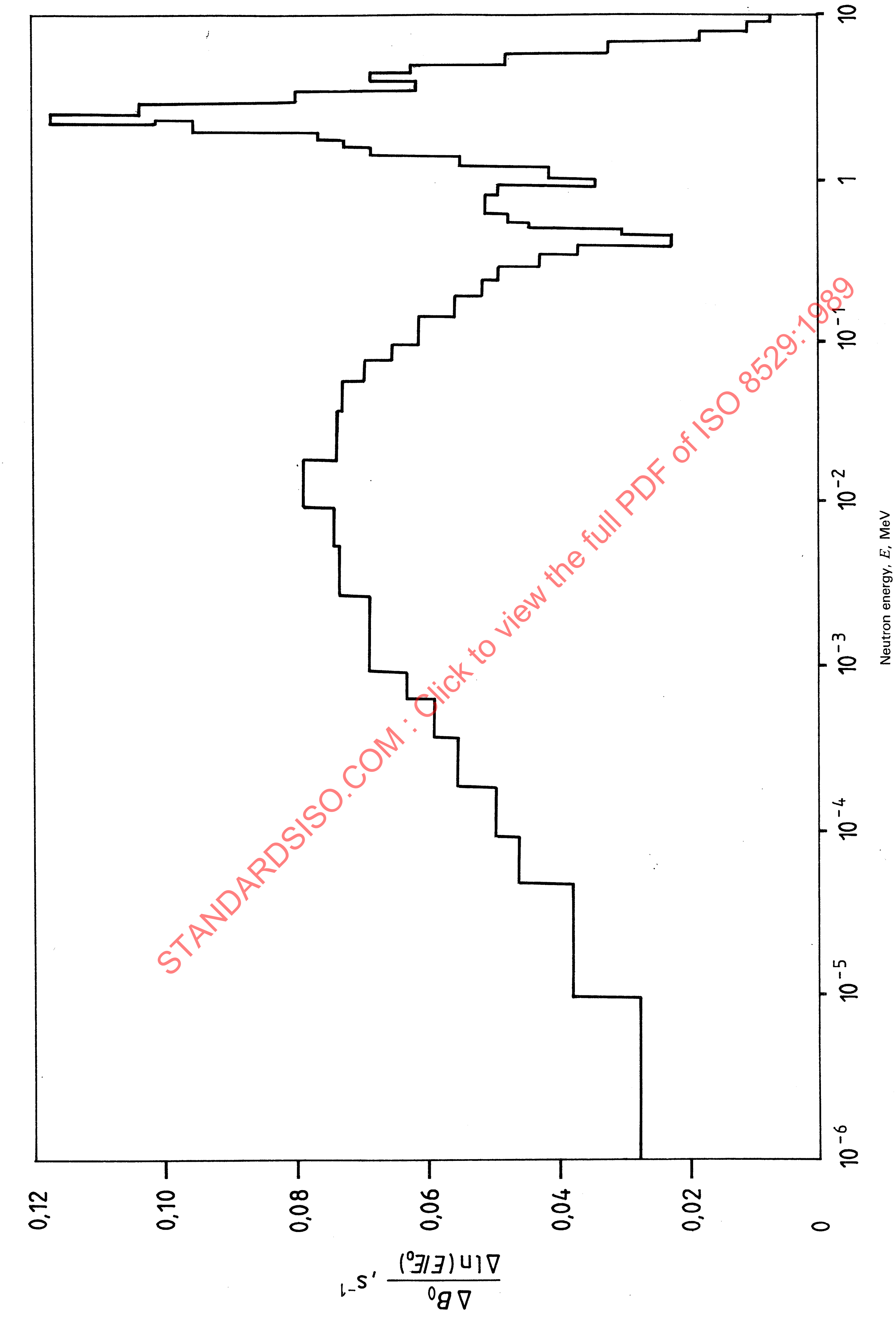

Figure A.3 — Neutron spectrum from a ^{252}Cf spontaneous fission source in the centre of a D_2O sphere with a radius of 150 mm [2]

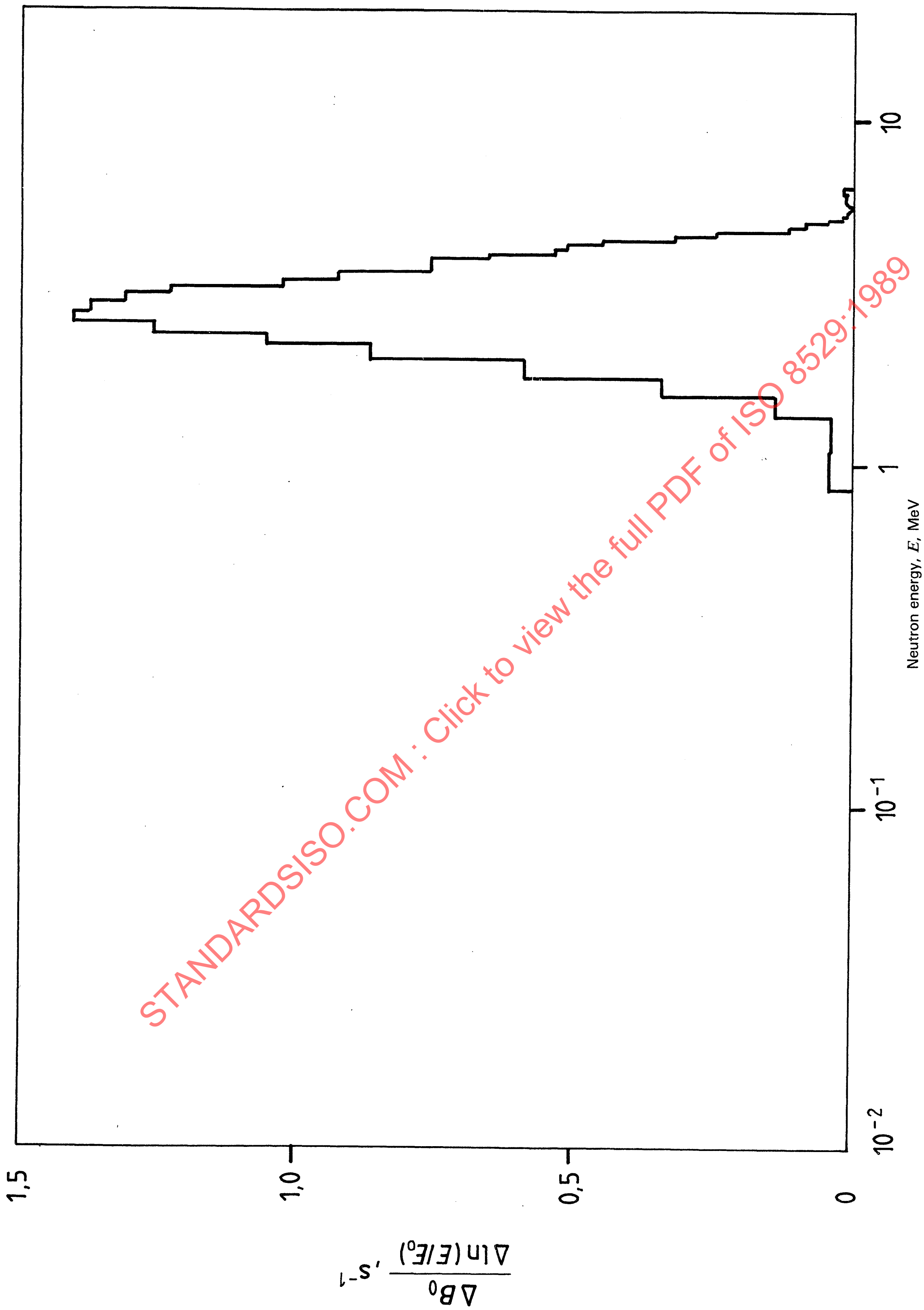

Figure A.4 — Neutron spectrum from a ^{241}Am-B(α,n) source [5]

Annex B
(normative)

"Fluence" to "dose" conversion factors for radionuclide sources

Radionuclide sources[5]	Mean "neutron fluence" to "dose equivalent" conversion factor[1)2)] $\overline{h}_{\Phi}$	Mean "neutron fluence" to "charged particle dose" conversion factor[1)3)] $\overline{d}^{c}_{\Phi}$	Mean "neutron fluence" to "photon dose" conversion factor for $^{1}H(n,\gamma)\,^{2}D$[1)3)] $\overline{d}^{\gamma}_{\Phi}$	Mean "neutron fluence" to "kerma" conversion factor[1)4)] $\overline{k}_{\Phi}$
	Sv·cm²	Gy·cm²	Gy·cm²	Gy·cm²
$^{252}Cf(D_2O$ moderated) (sphere 300 mm in diameter)[5]	$9,1 \times 10^{-11}$	$8,8 \times 10^{-12}$	$3,4 \times 10^{-12}$	$7,6 \times 10^{-12}$
^{252}Cf	$3,4 \times 10^{-10}$	$3,1 \times 10^{-11}$	$2,1 \times 10^{-12}$	$2,8 \times 10^{-11}$
^{241}Am-B(α,n)	$3,9 \times 10^{-10}$	$4,1 \times 10^{-11}$	$1,8 \times 10^{-12}$	$3,3 \times 10^{-11}$
^{241}Am-Be(α,n)	$3,8 \times 10^{-10}$	$4,8 \times 10^{-11}$	$1,9 \times 10^{-12}$	$3,7 \times 10^{-11}$

1) Mean "neutron fluence" to "dose equivalent" conversion factors were calculated according to $\overline{h}_{\Phi} = \frac{1}{B}\int_0^{\infty} B_E\, h_{\Phi}(E)\, \mathrm{d}E$ (see 3.18).
Corresponding relationships apply to the calculation of $\overline{k}_{\Phi}$, $\overline{d}^{c}_{\Phi}$ and $\overline{d}^{\gamma}_{\Phi}$ [6].

2) Values calculated for the spectra represented in annex A with $h_{\Phi}(E)$ for monoenergetic neutrons in accordance with ICRP Publication 21.

3) Values calculated by means of the analytical functions for d^{c}_{Φ} and d^{γ}_{Φ} given in [1]. Absorbed dose in volume element 57 of a tissue equivalent cylinder (300 mm in diameter and 600 mm high).

4) Values calculated for the spectra represented in annex A with $k_{\Phi}(E)$ for monoenergetic neutrons for material with the composition of standard man (see ICRU Report 26).

5) Heavy-water sphere with a diameter of 300 mm covered with a cadmium shell of thickness approximately 1 mm.

Annex C
(normative)

"Fluence" to "dose" conversion factors for monoenergetic neutrons

Neutron energy	"Neutron fluence" to "dose equivalent" conversion factors in accordance with ICRP Publication 21[1)] h_Φ	"Neutron fluence" to "charged particle" dose conversion factor[2) 3)] d^c_Φ	"Neutron fluence" to "photon dose" conversion factor for $^1H(n,\gamma)$ ^2D dose[2) 3)] d^γ_Φ	"Neutron fluence" to "kerma" conversion factor for standard man in accordance to ICRU Report 26[4)] k_Φ
MeV	Sv·cm^2	Gy·cm^2	Gy·cm^2	Gy·cm^2
Thermal	$1{,}07 \times 10^{-11}$			
0,002	$9{,}43 \times 10^{-12}$	$4{,}79 \times 10^{-13}$	$3{,}63 \times 10^{-12}$	$2{,}00 \times 10^{-13}$
0,025	$1{,}93 \times 10^{-11}$	$1{,}76 \times 10^{-12}$	$3{,}42 \times 10^{-12}$	$2{,}15 \times 10^{-12}$
0,144	$7{,}73 \times 10^{-11}$	$6{,}55 \times 10^{-12}$	$3{,}28 \times 10^{-12}$	$8{,}07 \times 10^{-12}$
0,250	$1{,}18 \times 10^{-10}$	$9{,}90 \times 10^{-12}$	$0{,}34 \times 10^{-11}$	$1{,}11 \times 10^{-11}$
0,565	$2{,}20 \times 10^{-10}$	$1{,}83 \times 10^{-11}$	$0{,}27 \times 10^{-11}$	$1{,}66 \times 10^{-11}$
1,20	$3{,}52 \times 10^{-10}$	$3{,}01 \times 10^{-11}$	$0{,}22 \times 10^{-11}$	$2{,}46 \times 10^{-11}$
2,50	$4{,}06 \times 10^{-10}$	$4{,}03 \times 10^{-11}$	$0{,}18 \times 10^{-11}$	$3{,}27 \times 10^{-11}$
2,80	$4{,}09 \times 10^{-10}$	$4{,}23 \times 10^{-11}$	$0{,}17 \times 10^{-11}$	$3{,}49 \times 10^{-11}$
3,20	$4{,}10 \times 10^{-10}$	$4{,}46 \times 10^{-11}$	$0{,}17 \times 10^{-11}$	$3{,}83 \times 10^{-11}$
5,0	$4{,}08 \times 10^{-10}$	$5{,}33 \times 10^{-11}$	$0{,}15 \times 10^{-11}$	$4{,}46 \times 10^{-11}$
14,8	$4{,}18 \times 10^{-10}$	$8{,}23 \times 10^{-11}$	$0{,}76 \times 10^{-11}$	$6{,}56 \times 10^{-11}$
19,0	$4{,}26 \times 10^{-10}$	$9{,}10 \times 10^{-11}$	$0{,}82 \times 10^{-11}$	$7{,}07 \times 10^{-11}$

1) The dose equivalent is the maximum value for a unidirectional broad beam of monoenergetic neutrons incident normally onto a slab or cylinder of tissue-equivalent material with a thickness or diameter of 300 mm. It should be realized that these values are an accepted convention and may not correspond to actual values for real phantoms or the human body.

Values of h_Φ for monoenergetic neutrons were calculated from the "dose equivalent rate" to "fluence rate" conversion factors given in table 4 of appendix 6 of ICRP Publication 21 with additional values of 305, 170 and 85 cm^{-2}·s^{-1}·mrem^{-1}·h at energy values of 0,005 MeV, 0,02 MeV and 0,05 MeV taken from figure 14 of ICRP Publication 21 to improve the interpolation procedure. Intermediate values of h_Φ at energies not given directly in ICRP Publication 21 were interpolated using the following Lagrange four-point interpolation formula:

$$\lg\left[h_\Phi(E)/h^0_\Phi\right] = \sum_{i=0}^{3}\left[\lg\left(h_{\Phi_i}/h^0_\Phi\right) \times \prod_{\substack{k=0\\(k\neq i)}}^{3} \frac{\lg(E/E_0) - \lg(E_k/E_0)}{\lg(E_i/E_0) - \lg(E_k/E_0)}\right]$$

where E_0 = 1 MeV and h^0_Φ = 1 Sv·cm^2 (arbitrary) [6].

2) Values calculated using the analytical function given in [1]. Absorbed dose in the volume element 57 of a tissue equivalent cylinder (300 mm in diameter).

3) Absorbed dose in element 57 of tissue equivalent cylinder (300 mm in diameter and 600 mm high).

4) Kerma is approximated by absorbed dose for a small piece of material with the composition of standard man irradiated free in air under the conditions of charged particle equilibrium.