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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

Reference number
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Annexes

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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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INTERNATIONAL STANDARD **ISO 8529 : 1989 (E)**

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}·s^{-1}, corresponding, at a neutron energy of 1 MeV, to dose equivalent rates of up to 100 mSv·h^{-1} (10 rem·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.

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[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:

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

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:

$$\phi = \frac{\mathrm{d}\Phi}{\mathrm{d}t} = \frac{\mathrm{d}^2 N}{\mathrm{d}a\mathrm{d}t}$$

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:

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

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:

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

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:

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

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 ^{1}H(n,γ) ^{2}D: 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

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[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 $\mathrm{d}E_{\mathrm{tr}}$ by $\mathrm{d}m$, expressed in grays, where $\mathrm{d}E_{\mathrm{tr}}$ 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 $\mathrm{d}m$:

$$K = \frac{\mathrm{d}E_{\mathrm{tr}}}{\mathrm{d}m}$$

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·m^2), 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 $\mathrm{d}H$ by $\mathrm{d}t$, expressed in sieverts reciprocal seconds (Sv·s^{-1}), 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 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·m^2), 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 $\mathrm{d}Q$ by $\mathrm{d}m$, expressed in coulombs reciprocal kilograms (C·kg^{-1}), where the value of $\mathrm{d}Q$ 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 $\mathrm{d}m$ are completely stopped in air:

$$X = \frac{\mathrm{d}Q}{\mathrm{d}m}$$

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 $\mathrm{d}X$ by $\mathrm{d}t$, expressed in coulombs reciprocal kilograms reciprocal seconds (C·kg^{-1}·s^{-1}), where $\mathrm{d}X$ is the increment of exposure in the time interval $\mathrm{d}t$:

$$\dot{X} = \frac{\mathrm{d}X}{\mathrm{d}t} = \frac{\mathrm{d}^2Q}{\mathrm{d}m\mathrm{d}t}$$

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 $\mathrm{d}N^+$ by $\mathrm{d}t$, expressed in becquerels (Bq)[2], where $\mathrm{d}N^+$ is the expectation value of the number of spontaneous nuclear transitions from that energy state in the time interval $\mathrm{d}t$:

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

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 $\mathrm{d}N^*$ by $\mathrm{d}t$, expressed in reciprocal

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[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:

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

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

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

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

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

The source strength B is derived from B_E as follows:

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

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, $\overline{h}_\Phi$: 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:

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

For the purposes of this International Standard, the symbol $\overline{h}_\Phi$ 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:

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

where

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

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

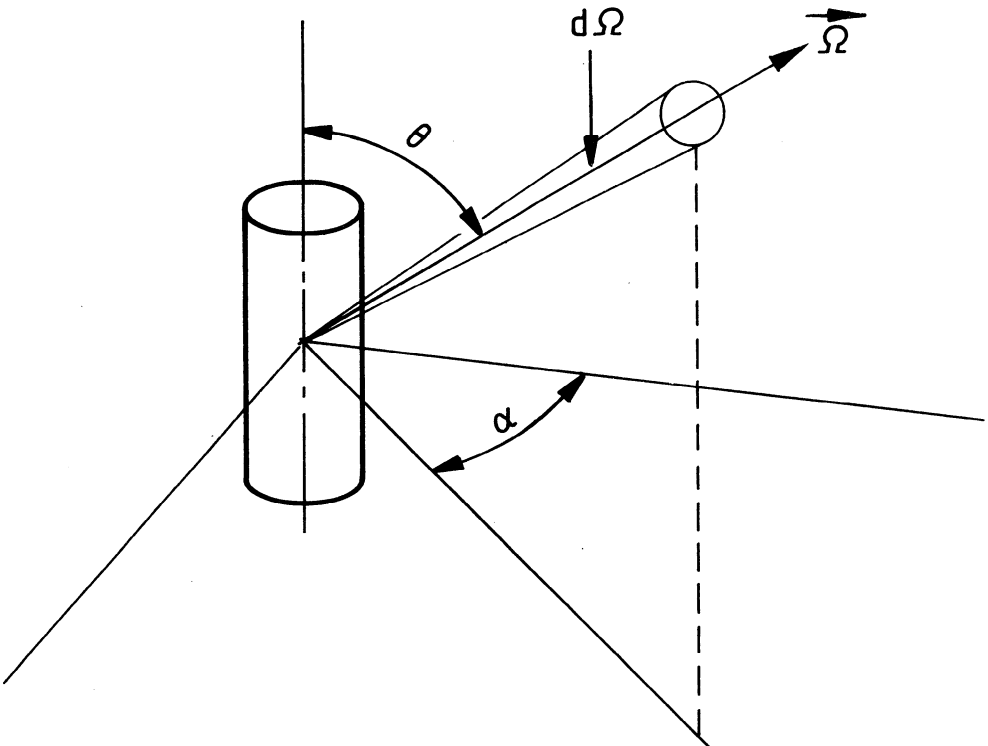

Figure 1 — Coordinate system for the case of an anisotropically emitting source (see 4.3)

α (see figure 1), does not depend noticeably on the azimuth angle α, but only upon angle θ. As the angular source strength $\mathrm{d}B/\mathrm{d}\Omega$ varies least for $\theta = 90°$, this direction should be used.

The neutron source strength, B, and the angular source strength, $\mathrm{d}B/\mathrm{d}\Omega$, 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

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

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.

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[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 energy 1)

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	
1,2	Accelerator-produced neutrons from reaction T(p,n) ^{3}He	
2,5 1)	Accelerator-produced neutrons from reaction T(p,n) ^{3}He	
2,8 2)	Accelerator-produced neutrons from reaction D(d,n) ^{3}He	
5,0	Accelerator-produced neutrons from reaction D(d,n) ^{3}He	
14,8 1) 2)	Accelerator-produced neutrons from reaction T(d,n) ^{4}He	
19,0	Accelerator-produced neutrons from reaction T(d,n) ^{4}He	

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.