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Reference neutron radiations — Characteristics and methods of production of simulated workplace neutron fields

Rayonnements neutroniques de référence — Caractéristiques et méthodes de production de champs de neutrons simulant ceux de postes de travail

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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.

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 International Standard may be the subject of patent rights. ISO shall not be held responsible for identifying any or all such patent rights.

International Standard ISO 12789 was prepared by Technical Committee ISO/TC 85, *Nuclear energy*, Subcommittee SC 2, *Radiation protection*. It is based on data published by the International Commission on Radiological Protection and by the International Commission on Radiation Units and Measurements, as well as on several scientific and technological investigations carried out by Subcommittee members.

Annex A of this International Standard is for information only.

Introduction

ISO 8529-1, ISO 8529-2 and ISO 8529-3 deal with the production, characterization and use of neutron fields for the calibration of personal dosimeters and area survey meters. These standards describe reference radiations with neutron energy spectra that are well defined and well suited for use in the calibration laboratory. However, the neutron spectra commonly encountered in routine radiation protection situations are, in many cases, quite different from those produced by the sources specified in the ISO standards. Since personal neutron dosimeters, and to a lesser extent survey meters, are generally quite energy-dependent in their dose equivalent response, it may not be possible to achieve an appropriate calibration for a device that is to be used in a workplace where the neutron energy spectrum and angular distribution differ significantly from those of the reference radiation used for calibration. ISO 8529-1 describes four radionuclide-based neutron reference radiations in detail. This International Standard includes the specification of neutron reference radiations that were developed to closely resemble radiations that are encountered in practice. Specific examples of simulated workplace neutron source facilities are included in annex A, for illustration.

Reference neutron radiations — Characteristics and methods of production of simulated workplace neutron fields

1 Scope

This International Standard gives guidance for producing and characterizing simulated workplace neutron fields that are to be used for calibrating neutron-measuring devices for radiation protection purposes. Both calculational and spectrometric measurement methods are discussed. Neutron energies in these reference fields range from approximately thermal neutron energies to several hundred GeV. The methods of production and the monitoring techniques for the various types of neutron fields are discussed, and the methods of evaluating and reporting uncertainties for these fields are also given.

2 Normative references

The following normative documents contain provisions which, through reference in this text, constitute provisions of this International Standard. For dated references, subsequent amendments to, or revisions of, any of these publications do not apply. However, parties to agreements based on this International Standard 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 8529-1:—[1)], *Reference neutron radiations — Part 1: Characteristics and methods of production.*

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.*

Guide to the expression of uncertainty in measurement, 1993, BIPM, IEC, IFCC, ISO, IUPAC, IUPAP, OIML.

3 Terms and definitions

For the purposes of this International Standard, the following terms and definitions apply.

NOTE 1 The definitions follow the recommendations of ICRU Report 51[8] and ICRU Report 33[4].

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

1) To be published.

3.1
neutron fluence
Φ
dN by da, where dN is the number of neutrons incident on a sphere of cross-sectional area da:

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

The unit of the neutron fluence is m^{-2}.

3.2
neutron fluence rate
φ
dΦ by dt, where dΦ is the increment of neutron fluence in the time interval dt:

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

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

3.3
spectral distribution of the neutron fluence
Φ_E
dΦ by dE, 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 The unit of the spectral distribution of the neutron fluence is $\mathrm{J}^{-1}\cdot\mathrm{m}^{-2}$.

3.4
ambient dose equivalent
$H^*(d)$
⟨at a point in a radiation field⟩ the dose equivalent at a point in a radiation field that would be produced by the corresponding expanded and aligned field in the ICRU sphere at a depth d on the radius opposing the direction of the aligned field

NOTE 1 For strongly penetrating radiation, a depth of 10 mm is currently recommended.

NOTE 2 The unit of ambient dose equivalent is $\mathrm{J}\cdot\mathrm{kg}^{-1}$ with the special name of sievert (Sv).

3.5
personal dose equivalent
$H_\mathrm{p}(d)$
the dose equivalent in soft tissue at an appropriate depth d below a specified point on the body

NOTE 1 For strongly penetrating radiation, a depth of 10 mm is currently recommended.

NOTE 2 The unit of personal dose equivalent is $\mathrm{J}\cdot\mathrm{kg}^{-1}$ with the special name of sievert (Sv).

NOTE 3 ICRU Report 39[5] defines the mass composition of soft tissue as: 76,2 % O; 10,1 % H; 11,1 % C; 2,6 % N.

NOTE 4 In ICRU Report 47[7], the ICRU has considered the definition of the personal dose equivalent to include the dose equivalent at a depth d in a phantom having the composition of ICRU tissue. Then, $H_\mathrm{p}(10)$ for the calibration of personal dosimeters is the dose equivalent at a depth of 10 mm in a phantom composed of ICRU tissue, but of the size and shape of the phantom used for calibration (a 30 cm × 30 cm × 15 cm parallelepiped).

3.6
neutron-fluence to dose-equivalent conversion coefficient
h_{Φ}
dose equivalent divided by neutron fluence

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

NOTE 1 The unit of the neutron-fluence to dose-equivalent conversion coefficient is Sv·m^2.

NOTE 2 Any statement of a fluence to dose-equivalent conversion coefficient requires the statement of the type of dose equivalent, e.g. ambient or personal dose equivalent.

4 Simulated workplace neutron fields

The neutron fluence spectra for a number of neutron fields have been available for some time[9, 10]. Neutron fluence spectra, measured at workplaces and in simulated workplace calibration fields, are included in a catalogue resulting from work sponsored by the European Commission[11]. This catalogue also contains response functions for common detectors and dosimeters in addition to fluence to dose-equivalent conversion coefficients.

Measurements in nuclear power plants[12-15], in the vicinity of transport casks containing spent fuel elements[14, 15], and in factories producing radionuclide neutron sources[15, 16] and reprocessing fuel elements[17] have demonstrated that neutron energy spectra in such environments can be described as a superposition of the following components: a high-energy component representing the uncollided neutrons, a scattered component with an approximately $1/E_{\mathrm{n}}$ dependence (where E_{n} is the neutron energy), and a thermal-neutron component. For these types of spectra, the design of simulated workplace neutron fields requires a knowledge and consideration of the components mentioned above because the relative fractions of these components can be very different in different situations.

Other radiation environments may contain neutrons having much higher energies. For example, neutrons with energies greater than 10 MeV, contributing 30 % to 50 % of the ambient dose equivalent and personal dose equivalent, have been found in the vicinity of high-energy particle accelerators[18, 19] and in aircraft flying at altitudes of 10 km to 15 km[20].

Because of the characteristics of available neutron dosimeters and survey meters, it is difficult to obtain proper measurements in the workplace based on the calibration sources specified in ISO 8529-1 when the workplace spectrum differs markedly from the calibration source spectrum. This can result in an inaccurate estimate of dose equivalent when such devices are used. At least two possibilities exist for improving the situation. First, the neutron spectrum of the workplace field can be measured, and a correction factor calculated to normalize the energy-dependent response of the detector. Secondly, a facility can be constructed to produce a neutron field that simulates the energy spectrum found in the workplace. When this field has been properly characterized, it can be used for the direct calibration of personal dosimeters and survey meters. This latter approach has been employed at a number of laboratories, and this International Standard gives guidance for producing and characterizing simulated workplace neutron spectra for the purpose of calibrating dosimeters and survey meters.

The establishment of simulated workplace neutron spectra in the calibration laboratory is necessary because the laboratory setting offers the possibility of controlling most influence quantities. The environmental parameters, such as temperature and humidity, can be maintained at a constant level. The materials used in the construction of the various pieces of equipment can also be specified and controlled in the laboratory. The general layout as well as the sources of neutron scatter can also be controlled, or at least maintained constant, in the calibration laboratory.

Simulated workplace neutron spectra that have been established in the calibration laboratory can be used to study the effects of changes in the neutron spectrum on the responses of personal dosimeters and survey meters. Dosimeter algorithms may also be tested with such sources used in conjunction with the other radionuclide sources recommended in ISO 8529-1. For these reasons, simulated workplace neutron fields should be provided for the investigation and calibration of neutron personal dosimeters and survey meters that are used in any of the workplace locations mentioned above.

5 General requirements for the production of simulated workplace neutron spectra

There are three basic methods for the production of simulated workplace neutron spectra. Irradiation facilities can be developed by making use of radionuclide neutron sources, accelerators and reactors. In each case, a variety of absorbing and scattering material can be placed between the primary source and detector in order to modify the initial source spectrum and thus simulate a workplace neutron spectrum. In order to characterize the neutron fields generated in such facilities, it is necessary to measure and calculate the energy spectrum, and to determine the spectral and angular neutron fluence and dose equivalent rates at the reference positions.

The field uniformity in the volume containing the detector is also to be determined. In some cases, this determination may be more amenable to calculational, rather than experimental, techniques. The intensity of sources that are expected to vary with irradiation time (such as accelerators or reactors) shall be monitored. This monitoring shall intercept a known portion of the neutron field, measure an unused portion of the field or measure a parameter that has been proven to be directly proportional to the neutron output (such as charged particle beam current or the fluence rate of associated particles accompanying the reaction). If the fluence rate of the neutron field can be varied over a large range, as is often the case when using an accelerator or reactor, it may be necessary to have more than one monitoring device available in order to ensure good counting statistics at low fluence rates while avoiding problems with dead-time losses at higher rates. Relationships shall then be established between the monitor reading and the dose equivalent at the reference position.

The neutron fluence rate can be determined either by absolute measurements or, in some instances, by determining the emission rate from the primary source of neutrons and knowing the effect of the scattering material used to modify the spectrum. The dose equivalent rate at the calibration position can then be determined from the neutron energy spectrum and the neutron fluence rate at this position by using the fluence to dose-equivalent conversion coefficient for the spectrum (see Table 1). If $H_p(10)$ is the quantity to be determined, the field directional characteristics are required. This information may also be needed for survey instruments in order to take into account any non-isotropy of their response characteristics.

The characterization of the simulated workplace neutron field should preferably also include the determination of the proportion of contaminating photons present since these photons may affect the reading of the survey meter or personal dosimeter being exposed. In addition, the relative fraction of photon dose equivalent present in the calibration field may differ from the fraction in the actual workplace neutron field. Methods for the measurement of the photon dose equivalent fraction include the use of multi-element thermoluminescent dosimeters (TLDs), paired ionization chambers, Geiger-Müller counters, recombination chambers and tissue-equivalent proportional counters, that can discriminate between neutron and photon events[13, 14, 30].

6 Characterization of simulated workplace neutron fields

6.1 Calculational methods

Monte Carlo computer codes are used in the design, production and characterization of simulated workplace neutron sources used for calibration purposes[21]. There are some guidelines for the use of computational methods that should be followed. First, it is recommended that only internationally tested computer codes, or those that have been compared favourably to direct measurements, be used. The version, or update number, of the code should be indicated. Second, it is important to document the initial conditions that are used to define the problem. This facilitates the intercomparison of results between laboratories. Since evaluated nuclear data files are periodically updated, it is also important to note the version of the cross-section data set used. Following these guidelines will help to foster consistency in the computation and reporting of calculated neutron spectra. It is also prudent to intercompare calculations to those performed with other commonly used codes.

It is difficult to estimate the overall uncertainty associated with Monte Carlo calculations. However, it is important to attempt a quantification of the uncertainty for a particular calculation, especially if the calculated spectrum is to be used to compute reference data such as fluence to dose-equivalent coefficients. The statistical uncertainty can be quite small, if enough histories are accumulated, but a small value for the statistical uncertainty does not necessarily indicate a small overall uncertainty. Clause 8 deals with the sources of uncertainties.

Table 1 — Ambient and personal dose equivalent per unit neutron fluence, $h^*(10)$ and $h_{p,\ slab}(10,\ \alpha)$, in units of pSv·cm^2, for monoenergetic neutrons incident on the ICRU sphere and ICRU tissue slab phantom

Energy (MeV)	$h^*_\Phi(10)$	$h_{p\Phi}(10, 0°)$	$h_{p\Phi}(10, 15°)$	$h_{p\Phi}(10, 30°)$	$h_{p\Phi}(10, 45°)$	$h_{p\Phi}(10, 60°)$	$h_{p\Phi}(10, 75°)$
$1{,}00 \times 10^{-9}$	6,60	8,19	7,64	6,57	4,23	2,61	1,13
$1{,}00 \times 10^{-8}$	9,00	9,97	9,35	7,90	5,38	3,37	1,50
$2{,}53 \times 10^{-8}$	10,6	11,4	10,6	9,11	6,61	4,04	1,73
$1{,}00 \times 10^{-7}$	12,9	12,6	11,7	10,3	7,84	4,7	1,94
$2{,}00 \times 10^{-7}$	3,5	13,5	12,6	11,1	8,73	5,21	2,12
$5{,}00 \times 10^{-7}$	13,6	14,2	13,5	11,8	9,40	5,65	2,31
$1{,}00 \times 10^{-6}$	13,3	14,4	13,9	12,0	9,56	5,82	2,40
$2{,}00 \times 10^{-6}$	12,9	14,3	14,0	11,9	9,49	5,85	2,46
$5{,}00 \times 10^{-6}$	12,0	13,8	13,9	11,5	9,11	5,71	2,48
$1{,}00 \times 10^{-5}$	11,3	13,2	13,4	11,0	8,65	5,47	2,44
$2{,}00 \times 10^{-5}$	10,6	12,4	12,6	10,4	8,10	5,14	2,35
$5{,}00 \times 10^{-5}$	9,90	11,2	11,2	9,49	7,32	4,57	2,16
$1{,}00 \times 10^{-4}$	9,40	10,3	9,85	8,64	6,74	4,10	1,99
$2{,}00 \times 10^{-4}$	8,90	9,84	9,41	8,22	6,21	3,91	1,83
$5{,}00 \times 10^{-4}$	8,30	9,34	8,66	7,66	5,67	3,58	1,68
$1{,}00 \times 10^{-3}$	7,90	8,78	8,20	7,29	5,43	3,46	1,66
$2{,}00 \times 10^{-3}$	7,70	8,72	8,22	7,27	5,43	3,46	1,67
$5{,}00 \times 10^{-3}$	8,00	9,36	8,79	7,46	5,71	3,59	1,69
$1{,}00 \times 10^{-2}$	10,5	11,2	10,8	9,18	7,09	4,32	1,71
$2{,}00 \times 10^{-2}$	16,6	17,1	17,0	14,6	11,6	6,64	2,11
$3{,}00 \times 10^{-2}$	23,7	24,9	24,1	21,3	16,7	9,81	2,85
$5{,}00 \times 10^{-2}$	41,1	39,0	36,0	34,4	27,5	16,7	4,78
$7{,}00 \times 10^{-2}$	60,0	59,0	55,8	52,6	42,9	27,3	8,10
$1{,}00 \times 10^{-1}$	88,0	90,6	87,8	81,3	67,1	44,6	13,7
$1{,}50 \times 10^{-1}$	132	139	137	126	106	73,3	24,2
$2{,}00 \times 10^{-1}$	170	180	179	166	141	100	35,5
$3{,}00 \times 10^{-1}$	233	246	244	232	201	149	58,5
$5{,}00 \times 10^{-1}$	322	335	330	326	291	226	102
$7{,}00 \times 10^{-1}$	375	386	379	382	348	279	139
$9{,}00 \times 10^{-1}$	400	414	407	415	383	317	171
$1{,}00 \times 10^{0}$	416	422	416	426	395	332	180
$1{,}20 \times 10^{0}$	425	433	427	440	412	335	210
$2{,}00 \times 10^{0}$	420	442	438	457	439	402	274
$3{,}00 \times 10^{0}$	412	431	429	449	440	412	306
$4{,}00 \times 10^{0}$	408	422	421	440	435	409	320
$5{,}00 \times 10^{0}$	405	420	418	437	435	409	331
$6{,}00 \times 10^{0}$	400	423	422	440	439	414	345
$7{,}00 \times 10^{0}$	405	432	432	449	448	425	361
$8{,}00 \times 10^{0}$	409	445	445	462	460	440	379
$9{,}00 \times 10^{0}$	420	461	462	478	476	458	399
$1{,}00 \times 10^{1}$	440	480	481	497	493	480	421
$1{,}20 \times 10^{1}$	480	517	519	536	599	523	464
$1{,}40 \times 10^{1}$	520	550	552	570	561	562	503
$1{,}50 \times 10^{1}$	540	564	565	584	575	579	520
$1{,}60 \times 10^{1}$	555	576	577	597	588	593	535
$1{,}80 \times 10^{1}$	570	595	593	617	609	615	561
$2{,}00 \times 10^{1}$	600	600	595	619	615	619	570
$3{,}00 \times 10^{1}$	515	—	—	—	—	—	—
$5{,}00 \times 10^{1}$	400	—	—	—	—	—	—
$7{,}50 \times 10^{1}$	330	—	—	—	—	—	—
$1{,}00 \times 10^{2}$	285	—	—	—	—	—	—
$1{,}25 \times 10^{2}$	260	—	—	—	—	—	—
$1{,}50 \times 10^{2}$	245	—	—	—	—	—	—
$1{,}75 \times 10^{2}$	250	—	—	—	—	—	—
$2{,}01 \times 10^{2}$	260	—	—	—	—	—	—

6.2 Spectrometric measurement methods

In order to cover the large range of neutron energy values normally encountered, it is necessary to use a spectrometer system that covers the energy range present. An example is the multisphere spectrometer system. This system is capable of performing measurements over a large energy range, but there are major limitations, such as limited energy resolution and uncertainty in data analysis. It has been found that the values of integral quantities, such as $H^*(10)$, agree quite well with other measurements and calculations. Multisphere spectrometer systems may be augmented by the use of hydrogen-filled proportional counters and scintillation detectors for specific measurement applications[23, 24]. In order to verify the consistency of spectrometric determinations, it is good practice to compare measurements from a number of laboratories. Such comparisons have been performed by several European laboratories[14, 24, 25].

The response functions of these systems must be carefully determined and it is preferable to perform a Monte Carlo simulation with a realistic detector model along with experimental calibrations using monoenergetic neutrons[26, 27]. In order to extend the range of the spectrometry to neutron energies above 20 MeV, additional detectors are necessary[28, 30].

7 Fluence to dose-equivalent conversion coefficients

This clause contains data used to calculate the ambient and personal dose equivalents at the point of test for the simulated workplace neutron spectra produced by methods given in this International Standard. In the case of $H_p(10)$, values for conversion coefficients are given as a function of angle, with the reference object being the ICRU slab phantom. It should be noted that the angular distribution of neutron fluence must be considered in the evaluation of $H_p(10)$. Table 1, adapted from ICRP Publication 74[3], is provided to aid in the calculation of the spectrum-averaged conversion coefficients for simulated workplace neutron spectra.

The response, or calibration factor, of a personal dosimeter or survey meter shall be obtained by determining the reading and the neutron fluence, both of which shall be corrected for unwanted contributions, and then applying the appropriate fluence to dose-equivalent conversion coefficient (refer to ISO 8529-2 and ISO 8529-3). The fluence to dose-equivalent conversion coefficient for a neutron spectrum can be calculated using the following equation:

$$\overline{h_{\Phi}} = \frac{\int h_{\Phi}(E)\Phi_E(E)\ \mathrm{d}E}{\int \Phi_E(E)\ \mathrm{d}E}$$

8 Sources of uncertainty

This clause describes the components expected to contribute to the overall uncertainty of fluence or dose equivalent. The numerical values given are approximations for the purposes of illustration and guidance only. Actual values of the uncertainties shall be calculated when developing specific simulated workplace neutron sources. All uncertainties should preferably be expressed in the form of standard deviations.

Characterization and optimization of the simulated workplace neutron field makes use of computer programmes for the calculation of neutron energy spectra. Various aspects of the calculations performed with these programmes can contribute to the uncertainties. The degree to which the initial conditions of a programme simulate the actual irradiation geometry can contribute to the uncertainties. The uncertainties in the nuclear cross-sections also contribute, and the statistical uncertainties should preferably be given as a contribution to the overall uncertainty. It is expected that calculations of integral neutron fluence and dose equivalent for simulated workplace neutron fields will agree with experimental determinations of these quantities to within approximately ± 20 %.

Measurements of neutron energy spectra are subject to uncertainties due to the response functions of spectrometers and the influence of various parameters used in the analysis codes.

Uncertainties in the corrections made for wall effects in proportional counters and the efficiency of scintillators as a function of neutron energy can contribute to the overall uncertainty. It is expected that the uncertainty in spectrometric measurements of integral neutron fluence or dose equivalent in reference simulated workplace neutron fields will be approximately 10 % to 20 %.

