IAEA SAFETY STANDARDS SERIES

Assessment of Occupational Exposure Due to External Sources of Radiation

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INTERNATIONAL ATOMIC ENERGY AGENCY AND THE
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SAFETY GUIDE

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IAEA SAFETY STANDARDS

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ASSESSMENT OF OCCUPATIONAL EXPOSURE DUE TO EXTERNAL SOURCES OF RADIATION

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The Agency's Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world''.

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FOREWORD

by Mohamed ElBaradei

Director General

One of the statutory functions of the IAEA is to establish or adopt standards of safety for the protection of health, life and property in the development and application of nuclear energy for peaceful purposes, and to provide for the application of these standards to its own operations as well as to assisted operations and, at the request of the parties, to operations under any bilateral or multilateral arrangement, or, at the request of a State, to any of that State's activities in the field of nuclear energy.

The following advisory bodies oversee the development of safety standards: the Advisory Commission on Safety Standards (ACSS); the Nuclear Safety Standards Advisory Committee (NUSSAC); the Radiation Safety Standards Advisory Committee (RASSAC); the Transport Safety Standards Advisory Committee (TRANSSAC); and the Waste Safety Standards Advisory Committee (WASSAC). Member States are widely represented on these committees.

In order to ensure the broadest international consensus, safety standards are also submitted to all Member States for comment before approval by the IAEA Board of Governors (for Safety Fundamentals and Safety Requirements) or, on behalf of the Director General, by the Publications Committee (for Safety Guides).

The IAEA's safety standards are not legally binding on Member States but may be adopted by them, at their own discretion, for use in national regulations in respect of their own activities. The standards are binding on the IAEA for application in relation to its own operations and to operations assisted by the IAEA. Any State wishing to enter into an agreement with the IAEA for its assistance in connection with the siting, design, construction, commissioning, operation or decommissioning of a nuclear facility or any other activities will be required to follow those parts of the safety standards that pertain to the activities to be covered by the agreement. However, it should be recalled that the final decisions and legal responsibilities in any licensing procedures rest with the States.

Although the safety standards establish an essential basis for safety, the incorporation of more detailed requirements, in accordance with national practice, may also be necessary. Moreover, there will generally be special aspects that need to be assessed by experts on a case by case basis.

The physical protection of fissile and radioactive materials and of nuclear power plants as a whole is mentioned where appropriate but is not treated in detail; obligations of States in this respect should be addressed on the basis of the relevant

instruments and publications developed under the auspices of the IAEA. Non-radiological aspects of industrial safety and environmental protection are also not explicitly considered; it is recognized that States should fulfil their international undertakings and obligations in relation to these.

The requirements and recommendations set forth in the IAEA safety standards might not be fully satisfied by some facilities built to earlier standards. Decisions on the way in which the safety standards are applied to such facilities will be taken by individual States.

The attention of States is drawn to the fact that the safety standards of the IAEA, while not legally binding, are developed with the aim of ensuring that the peaceful uses of nuclear energy and of radioactive materials are undertaken in a manner that enables States to meet their obligations under generally accepted principles of international law and rules such as those relating to environmental protection. According to one such general principle, the territory of a State must not be used in such a way as to cause damage in another State. States thus have an obligation of diligence and standard of care.

Civil nuclear activities conducted within the jurisdiction of States are, as any other activities, subject to obligations to which States may subscribe under international conventions, in addition to generally accepted principles of international law. States are expected to adopt within their national legal systems such legislation (including regulations) and other standards and measures as may be necessary to fulfil all of their international obligations effectively.

PREFACE

Occupational exposure to ionizing radiation can occur in a range of industries, medical institutions, educational and research establishments and nuclear fuel cycle facilities. Adequate radiation protection of workers is essential for the safe and acceptable use of radiation, radioactive materials and nuclear energy.

In 1996, the Agency published Safety Fundamentals on Radiation Protection and the Safety of Radiation Sources (IAEA Safety Series No. 120) and International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources (IAEA Safety Series No. 115), both of which were jointly sponsored by the Food and Agriculture Organization of the United Nations, the IAEA, the International Labour Organization, the OECD Nuclear Energy Agency, the Pan American Health Organization and the World Health Organization. These publications set out, respectively, the objectives and principles for radiation safety and the requirements to be met to apply the principles and to achieve the objectives.

The establishment of safety requirements and guidance on occupational radiation protection is a major component of the support for radiation safety provided by the Agency to its Member States. The objective of the Agency's Occupational Protection Programme is to promote an internationally harmonized approach to the optimization of occupational radiation protection, through the development and application of guidelines for restricting radiation exposures and applying current radiation protection techniques in the workplace.

Guidance on meeting the requirements of the Basic Safety Standards for occupational protection is provided in three interrelated Safety Guides, one giving general guidance on the development of occupational radiation protection programmes and two giving more detailed guidance on the monitoring and assessment of workers' exposures due to external radiation sources and from intakes of radionuclides, respectively. These Safety Guides together reflect the current internationally accepted principles and recommended practices in occupational radiation protection, with account taken of the major changes that have occurred over the past decade.

The three Safety Guides on occupational radiation protection are jointly sponsored by the IAEA and the International Labour Office. The Agency gratefully acknowledges the contribution of the European Commission to the development of the present Safety Guide.

The present Safety Guide addresses the assessment of exposure due to external sources of radiation in the workplace. Such exposure can result from a number of sources within a workplace, and the monitoring of workers and the workplace in such situations is an integral part of any occupational radiation protection programme. The assessment of exposure due to external radiation sources depends critically upon knowledge of the radiation type and energy and the conditions of exposure. The

present Safety Guide reflects the major changes over the past decade in international practice in external dose assessment.

EDITORIAL NOTE

An appendix, when included, is considered to form an integral part of the standard and to have the same status as the main text. Annexes, footnotes and bibliographies, if included, are used to provide additional information or practical examples that might be helpful to the user.

The safety standards use the form 'shall' in making statements about requirements, responsibilities and obligations. Use of the form 'should' denotes recommendations of a desired option.

The English version of the text is the authoritative version.

CONTENTS

1. INTRODUCTION

BACKGROUND

1.1. Occupational exposure to radiation can occur as a result of various human activities. These include work associated with the different stages of the nuclear fuel cycle, the use of radioactive sources and X ray machines in medicine, scientific research, education, agriculture and industry, and occupations that involve the handling of materials containing enhanced concentrations of naturally occurring radionuclides. In order to control this exposure, it is necessary to be able to assess the magnitude of the doses involved.

1.2. The IAEA Safety Fundamentals publication Radiation Protection and the Safety of Radiation Sources [1] presents the objectives, concepts and principles of radiation protection and safety. Requirements designed to meet the objectives and apply the principles specified in the Safety Fundamentals, including requirements for the protection of workers exposed to sources of radiation, are established in the International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources (commonly referred to as the Basic Safety Standards or BSS), jointly sponsored by the IAEA and five other international organizations [2].

1.3. Three interrelated Safety Guides prepared jointly by the IAEA and the International Labour Office (ILO) provide guidance on fulfilling the requirements of the Basic Safety Standards with respect to occupational exposure. The Safety Guide [3] gives general advice on the exposure conditions for which monitoring programmes should be set up to assess radiation doses arising from external irradiation and from intakes of radionuclides by workers. The present Safety Guide gives more specific guidance on the assessment of doses from external sources of radiation while Ref. [4] deals with intakes of radioactive materials.

1.4. Recommendations related to occupational radiation protection have also been developed by the International Commission on Radiological Protection (ICRP) [5]. These and other current recommendations of the ICRP [6] and the International Commission on Radiation Units and Measurements (ICRU) [7–10] have been taken into account in preparing this Safety Guide.

OBJECTIVE

1.5. The purpose of this Safety Guide is to provide comprehensive guidance for regulatory authorities on meeting the requirements for conducting effective

assessments of occupational exposure to external sources of ionizing radiation. The Safety Guide will also be useful to those concerned with the planning and management of occupational monitoring programmes, to those responsible for the operation of individual monitoring services, and to those involved in the design of dosimeters and equipment for use in personal dosimetry and workplace monitoring.

SCOPE

1.6. This Safety Guide contains guidance on establishing monitoring programmes for external exposure: the appropriate dosimetry to be used for individual and workplace monitoring, the interpretation of results, record keeping and quality assurance. The overall objectives of personal dosimetry systems and services are discussed, with particular attention being paid to the quantities to be measured and the precision and accuracy necessary in making such measurements. Guidance on the type testing and performance testing of dosimeters is given, together with the necessary dosimetric data to carry out this work.

1.7. The subject of workplace monitoring is discussed only to the extent that such monitoring is used in the assessment of individual dose. External exposure from contamination on the skin is discussed in the Appendix, but the monitoring of contamination on workplace surfaces is addressed in the related Safety Guide on internal exposure [4]. Specialized dosimetry for accident situations in which doses significantly exceed occupational dose limits is outside the scope of this publication.

STRUCTURE

1.8. The relationship between the protection quantities and operational quantities of dose is presented in Section 2. Section 3 outlines the objectives and use of monitoring for external radiation exposure. Section 4 presents the essential features of monitoring programmes and the roles of individual and workplace monitoring. The dosimetric specifications for both personal dosimeters and workplace monitoring are described in Section 5, including accuracy, uncertainties and performance specifications. Type testing of personal dosimeters and workplace monitors is outlined in Section 6. Calibration and performance testing are discussed in Sections 7 and 8 respectively. Section 9 covers record keeping and Section 10 deals with quality assurance.

1.9. Additional information is provided in an appendix and annexes. The Appendix addresses the question of skin dosimetry. Annex I gives the recommended values of

radiation weighting factors and the relationship between quality factor and linear energy transfer. Annexes II and III give an overview of instrumentation for individual and workplace monitoring respectively. Annex IV describes reference and standard test conditions specified by the International Electrotechnical Commission (IEC). Annex V provides the dose conversion coefficients recommended by the ICRP and ICRU, as well as details of the radiation fields recommended by the International Organization for Standardization (ISO) for calibration purposes. Annex VI provides examples of the IEC standards for radiation monitoring equipment.

2. DOSIMETRIC QUANTITIES

INTRODUCTION

2.1. The dosimetric quantities recommended for radiological protection purposes, and in which the dose limits are expressed in the BSS, are the effective dose E and the equivalent dose H_T in tissue or organ T. The basic physical quantities include the particle fluence ϕ, the kerma K and the absorbed dose D.

2.2. The ICRU introduced operational quantities for practical use in radiological protection where exposure to external sources is concerned [7]. These quantities were later defined in ICRU Report 51 [10]. The operational quantities for area monitoring are the ambient dose equivalent $H^*(d)$ and the directional dose equivalent $H'(d,\Omega)$, and the quantity for individual monitoring is the personal dose equivalent $H_p(d)$. These quantities are briefly discussed in the related Safety Guide [3] and formally defined in the BSS [2]. A detailed evaluation of the numerical relationship between the physical, protection and operational quantities has been conducted by a Joint Task Group of the ICRP and ICRU [11]. The conceptual relationship between those quantities is illustrated in Fig. 1 [11].

2.3. The determination of equivalent dose, and hence of effective dose, involves the use of radiation weighting factors w_R as multipliers of absorbed dose, to reflect the greater detriment resulting from a given absorbed dose when it is delivered by high linear energy transfer (LET) radiation rather than low LET radiation. Recommended values of w_R are based on a review of published biological information and are listed in Table I–I (Annex I).

2.4. Radiation quality factors Q are used in determining the operational quantities, and are based on a Q–LET relationship. Quality factors are also used as approximate

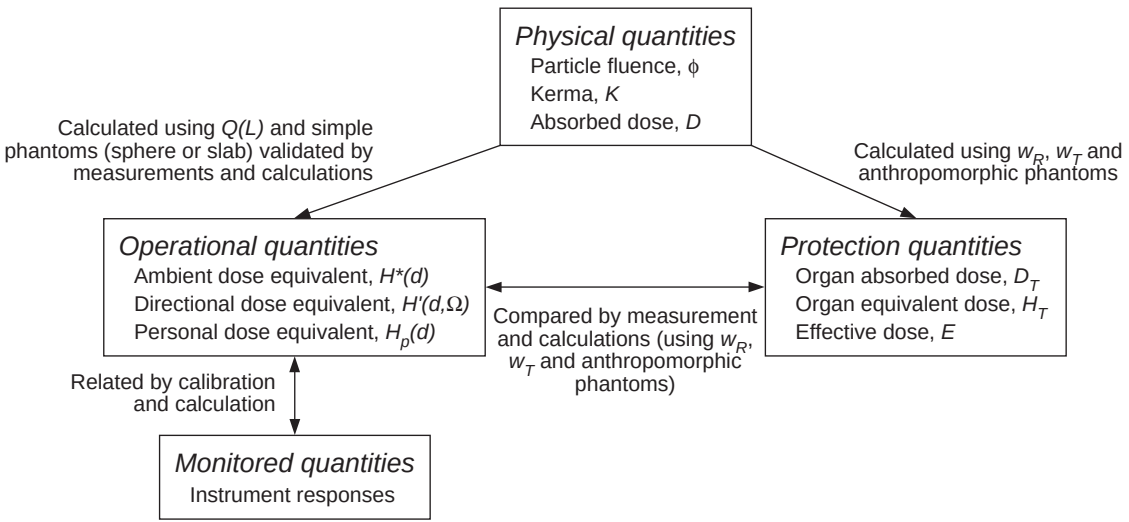

FIG. 1. Relationship of quantities for radiological protection purposes [11].

values of w_R for radiation types not included in Table I–I. The recommended Q–LET relationship is given in Table I–II.

OPERATIONAL QUANTITY FOR INDIVIDUAL MONITORING

2.5. The operational dosimetric quantity recommended in the BSS for individual monitoring is the personal dose equivalent $H_p(d)$ [9–10]. This is the dose equivalent in soft tissue below a specified point on the body, at an appropriate depth d. One possible approach to measuring $H_p(d)$ would be to use a detector worn at the surface of the body and covered with an appropriate thickness of tissue substitute. However, other approaches may be acceptable, provided that the necessary variation of response with energy is achieved.

2.6. Any statement of personal dose equivalent should include a specification of the reference depth d. For weakly penetrating and strongly penetrating radiation (see para. 2.14), the recommended depths are 0.07 mm and 10 mm respectively, although other depths may be appropriate in particular cases, for example 3 mm for the lens of the eye. In order to simplify the notation, d is assumed to be expressed in millimetres and hence the personal dose equivalents at the two recommended depths mentioned above are denoted by $H_p(0.07)$ and $H_p(10)$.

2.7. $H_p(10)$, i.e. the personal dose equivalent at 10 mm depth, is used to provide an estimate of effective dose that avoids both underestimation and excessive overestimation. The sensitive cells of the skin are considered to be between 0.05 and 0.1 mm below the skin surface, and therefore $H_p(0.07)$ is used to estimate the equivalent dose to skin. $H_p(0.07)$ should also be used for extremity monitoring, where the skin dose is the limiting quantity.

2.8. The calibration of dosimeters is performed under simplified conventional conditions (standard test conditions, see Section 5), on an appropriate phantom. The quantity $H_p(d)$ may be used to specify the dose equivalent at a point in a phantom representing the body. If a dosimeter measures $H_p(d)$ correctly at a point in such a phantom, it is assumed that it measures $H_p(d)$ with sufficient accuracy in the body of any person.

QUANTITIES FOR WORKPLACE MONITORING

2.9. The operational quantities recommended for workplace monitoring are defined in a phantom known as the ICRU sphere [10]. This is a sphere of tissue equivalent

material with a diameter of 30 cm, a density of 1 g/cm^3 and an elemental composition (by mass) of 76.2% oxygen, 11.1% carbon, 10.1% hydrogen and 2.6% nitrogen.

2.10. The two quantities recommended by the ICRU for area monitoring [10] are the ambient dose equivalent $H^*(d)$ and the directional dose equivalent $H'(d,\Omega)$. These are appropriate for monitoring strongly penetrating and weakly penetrating radiation fields (see para. 2.14) respectively.

2.11. The ambient dose equivalent $H^*(d)$ at a point in a radiation field is the dose equivalent 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.

2.12. The expanded field is one in which the fluence, and its angular and energy distribution, are the same throughout the volume of interest as in the actual field at the point of reference. In the expanded and aligned field, the fluence and its energy distribution are the same as in the expanded field, but the fluence is unidirectional.

2.13. Any statement of ambient dose equivalent should include a specification of the reference depth d. For strongly penetrating radiation (see para. 2.14), the recommended depth is 10 mm. As for personal dose equivalent, d should be expressed in millimetres, so $H^*(10)$ is the ambient dose equivalent for a depth of 10 mm. It is necessary for measuring $H^*(d)$ that the radiation field be uniform over the sensitive volume of the instrument, and that the instrument have an isotropic response.

2.14. Weakly penetrating and strongly penetrating radiation are defined as follows [7]. If, for a given orientation of the body in a uniform and unidirectional radiation field, the equivalent dose received by any small area of the sensitive layer of the skin is more than ten times larger than the effective dose, the radiation is said to be weakly penetrating. If the equivalent dose is less than ten times larger than the effective dose, then the radiation is said to be strongly penetrating.

2.15. The directional dose equivalent $H'(d,\Omega)$ at a point in a radiation field is the dose equivalent that would be produced by the corresponding expanded field in the ICRU sphere, at a depth d on a radius in a specified direction Ω. Any statement of directional dose equivalent should include a specification of the reference depth d and the direction Ω of the radiation. For weakly penetrating and strongly penetrating radiation, the recommended depths are 0.07 mm and 10 mm respectively. Again, d should be expressed in millimetres.

2.16. If the field is unidirectional, the direction Ω is specified as the angle between the radius opposing the incident field and the specified radius. When the specified

radius is parallel to the radiation field (i.e. when $\Omega = 0°$) the quantity $H'(d,0)$ may be written simply as $H'(d)$. Furthermore, in a unidirectional field, $H'(d) = H^*(d)$. It is necessary for measuring $H'(d,\Omega)$ that the radiation field be uniform over the dimensions of the instrument and that the instrument have the appropriate directional response. For weakly penetrating radiation, an instrument which determines the dose equivalent at the recommended depth in a plane slab of tissue equivalent material will adequately determine $H'(0.07)$ provided that the slab surface is perpendicular to the direction of the radiation field.

3. MONITORING PROGRAMMES

GENERAL OBJECTIVE

3.1. The general objective of operational monitoring programmes is the assessment of workplace conditions and individual exposures. The assessment of doses to workers routinely or potentially exposed to external sources of radiation constitutes an integral part of any radiation protection programme and helps to ensure acceptably safe and satisfactory radiological conditions in the workplace.

3.2. Measures to meet the general requirements for the radiation protection of workers are described in the related Safety Guide [3]. The specific aspects of monitoring that relate to exposure to external radiation are described below.

3.3. Radioactive contamination of workplace surfaces can contribute to external exposure of workers. However, in many contamination situations (and particularly where alpha activity is significant), internal exposure pathways dominate. For this reason, the topic of surface contamination monitoring is addressed in the related Safety Guide [4].

3.4. Contamination of the skin will lead to external exposure, and sometimes even to internal exposure, depending upon the radionuclide(s) involved, the chemical form(s) present, and the activity level. The Appendix addresses the assessment of doses resulting from skin contamination.

3.5. Additional information on instrumentation for monitoring individuals and the workplace is presented in Annexes II and III respectively.

INDIVIDUAL DOSE ASSESSMENT

3.6. In most circumstances, doses due to external irradiation can be readily assessed by the systematic individual monitoring of workers. In cases where individual monitoring is unable to provide an adequate indication of the doses to workers, results of workplace monitoring may be used for personal dose assessment. It may be appropriate to derive an assessment of exposure from the results of workplace monitoring when:

(a) No effective method of individual monitoring is available and a method based on workplace monitoring has been shown to be acceptable;
(b) Doses are relatively constant and can be reliably assessed by other means (for example, in research laboratories using small controlled sources); or
(c) The workers concerned are regularly employed in a supervised area, or only occasionally enter controlled areas (see paras 5.17–5.31 of Ref. [3]).

3.7. Individual monitoring is normally required for persons who routinely work in areas that are designated as controlled areas because of the external radiation hazard. An individual monitoring programme for external radiation exposure is intended to provide information for the optimization of protection, to demonstrate that the worker's exposure has not exceeded any dose limit or the level anticipated for the given activities, and to verify the adequacy of workplace monitoring.

3.8. For supervised areas where individual monitoring is not required, it may be simpler to use a limited number of individual dosimeters than to adopt a comprehensive programme of monitoring of the workplace. In any case, individual monitoring for the purpose of dose records may be considered good practice for all workers in a supervised area.

Design of a monitoring programme

3.9. Where individual monitoring of workers is used, personnel should each be provided with an integrating dosimeter. Where dose equivalent rates encountered in the workplace may vary by more than a factor of ten, an additional, direct reading dosimeter and/or a warning device should be issued for dose control purposes (see para. 3.24).

3.10. An individual monitoring service approved by the regulatory authority should be used. The regulatory authority should require such a service to supply dosimeters capable of measuring $H_p(10)$ and $H_p(0.07)$ with adequate accuracy for all relevant

radiation types. The regulator should also require that the service be staffed with adequately qualified and trained personnel, and have suitable processing equipment and other relevant facilities. The regulatory authority should inspect the service, and should require that processing and dose reporting take place within a prescribed timescale, and that an adequate quality assurance (QA) system be in operation.

3.11. Measurement of $H_p(10)$ is often sufficient to assess a worker's exposure. However, if the radiation field contains significant amounts of weakly penetrating radiation (such as beta particles, or photons of energy below 15 keV), $H_p(0.07)$ may be comparable with, or significantly larger than, $H_p(10)$; for such fields, the dosimeter should be capable of measuring the dose equivalent at a depth of 0.07 mm.

3.12. Where the dose equivalent to the lens of the eye is to be determined, the personal dose equivalent $H_p(3)$ can normally be assessed with sufficient accuracy from the measurements of $H_p(10)$ and $H_p(0.07)$. If $H_p(10)$ and $H_p(0.07)$ are below the respective dose limits, it can be shown that, in the vast majority of cases, the value of $H_p(3)$ will also be below the dose limit for the lens of the eye (150 mSv).

3.13. In most cases, a single dosimeter worn on the trunk is adequate. For strongly penetrating radiation, this dosimeter should be placed in a position at which the highest exposure on the surface of the trunk is expected. For radiation incident primarily from the front, or when the incidence is expected to be rotationally symmetrical or isotropic, the dosimeter should be worn on the front of the torso, between the shoulders and the waist. Dosimeters to assess doses to the lens of the eye should be worn near the eyes (e.g. on the forehead or a cap).

3.14. In order to obtain a better assessment of the effective dose received in an inhomogeneous radiation field, it is useful for workers to wear additional dosimeters on other parts of the body. In some special situations — for example in medical radiology, where protective clothing such as lead aprons is used — it is advisable to use one dosimeter under the protective apron and one on an unshielded part of the body. The purpose of the two dosimeters is to determine the effective dose received by the shielded and unshielded parts of the body. These can be combined to give the total effective dose by the use of suitable algorithms; the available methods have been reviewed, and recommendations made, by the US National Council of Radiation Protection (NCRP) [12].

3.15. In cases where the maximum dose to extremities is expected to be at least ten times greater than the dose to the surface of the whole body (compare the factor of ten between the single year effective dose limit of 50 mSv for the whole body and the equivalent dose limit of 500 mSv to the extremities), one or more extremity

dosimeters should be worn, in positions that will measure the dose to the area(s) expected to receive the highest dose.

3.16. In routine operations, each monitored worker should usually have two dosimeters; the worker wears one while the other (which was worn previously) is being processed and evaluated. The frequency of dosimeter exchange should be established by the dosimetry service depending on the type of work being performed (see paras 3.17–3.29) and the anticipated exposure associated with the work, the characteristics of the dosimeters and the overall limit of detection of the dosimetry system. The fading characteristics of photographic film, for example, usually dictate a shorter exchange period for film dosimeters than for thermoluminescent dosimeters (TLDs). Exchange frequencies can range from daily, in special operations, to every six months, if the exposure is expected to be very low, but exchange periods of one to three months are typical. For routine operations using direct reading dosimeters, other approaches are possible and have been adopted. It is possible to issue a dosimeter to each individual on a daily basis (not necessarily the same one each day), or to issue a dosimeter to an individual for a period as long as a year, with readings being taken periodically. Both of these options (and there may be others) reduce the number of dosimeters needed to a little over one per person (standby dosimeters will, of course, be needed to cover dosimeter failure and maintenance).

Choice of personal dosimeter

Routine monitoring

3.17. The choice of a personal dosimeter will depend not only on the type of radiation but also on the information that is needed in addition to $H_p(d)$. In practice, the following types of dosimeter may be used:

(a) Photon dosimeters, giving information only on the personal dose equivalent $H_p(10)$;
(b) Beta–photon dosimeters, giving information on the personal dose equivalents $H_p(0.07)$ and $H_p(10)$;
(c) Photon dosimeters of the discriminating type giving, in addition to $H_p(10)$, some indication of the radiation type and effective energy, and detection of high energy electrons;
(d) Extremity dosimeters, giving information on $H_p(0.07)$ for beta–photon radiation (and for neutrons if neutron sources are being handled);
(e) Neutron dosimeters, giving information on $H_p(10)$.

3.18. In radiation fields where only photon radiation is important, it is usually sufficient to measure only $H_p(10)$. A simple dosimeter (of the type described in (a) above) is therefore adequate in most practical situations. For a wide range of photon energies, TLDs, radiophotoluminescent (RPL) glass or photographic film dosimeters can be used, provided that they exhibit an adequate energy dependence. In addition, many electronic dosimeters are available which measure $H_p(10)$ directly, above a threshold of 20–80 keV (depending on type). Furthermore, optically stimulated luminescence (OSL) has reached an advanced stage of development, and is currently used by at least one major commercial dosimetry service.

3.19. When it is likely that beta radiation may contribute significantly to the radiation field, dosimeters of the type described in (b) should be used. These may be TLDs or photographic film dosimeters with two or more thermoluminescent elements or films under filters of different materials and thicknesses, or electronic dosimeters. However, when a significant portion of the beta dose is likely to come from low energy beta particles, electronic dosimeters of the current designs may not be appropriate.

3.20. For extremity dosimetry, especially of the hand, a simple single-element TLD should be sufficient if it is placed on the most highly exposed finger and is facing the source. For the best accuracy in measuring low energy beta radiation, the detector should be thin and filtered by a thickness of tissue substitute such that the dose at a nominal depth of 7 mg/cm^2 (or 0.07 mm)[1] can be assessed (e.g. a measurement in a tissue equivalent detector with a thickness of 5 mg/cm^2 — corresponding to an effective thickness of 3 mg/cm^2 — beneath a tissue equivalent filter with a thickness of approximately 4 mg/cm^2 would suffice).

3.21. Simple types of neutron dosimeter cannot provide information on neutron dose equivalents over the whole energy range of interest, and therefore extra effort is needed if individual monitoring for neutrons is necessary. However, neutron dose equivalents are often small compared with the dose equivalent limit, and with contributions from gamma radiation. As gamma radiation is always present in neutron fields, a photon dosimeter should always be worn with a neutron dosimeter. In some neutron fields, the ratio of neutron to gamma dose equivalent has been found to vary by orders of magnitude. Neutron dose equivalents cannot, therefore, be derived with

[1] In discussing the measurement and effects of beta radiation, 'thicknesses' of material are often expressed in units of mg/cm^2 to allow direct comparisons between materials of different densities. For tissue equivalent material, the density is 1 g/cm^2, so 7 mg/cm^2 corresponds to a depth of 0.07 mm.

sufficient accuracy from gamma dose equivalent measurements by assuming a constant ratio for a given workplace.

3.22. Doses from thermal, intermediate and high energy neutrons can be assessed by a system such as an albedo dosimeter (see Annex II). For high energy neutrons, however, the dose equivalent response of albedo dosimeters varies widely with energy, and other methods, such as solid state track detectors, are more suitable (see Annex II). Bubble polymer detectors — a type of direct reading neutron dosimeter — are very sensitive to neutrons, with a detection capability of a few microsieverts, and are completely insensitive to photons. However, each of the three types of dosimeter has only a limited neutron energy range.

3.23. Special individual monitoring systems for neutrons are described in Annex II, and examples of their application to individual monitoring at reprocessing plants, hospitals and reactor areas are given in Ref. [13].

3.24. For controlling individual exposure on a day to day basis, it may be necessary to use supplementary dosimeters of the direct reading (electronic) type, which can provide estimates of an individual's dose with a frequency greater than that provided by typical routine dosimetry. These dosimeters should be used for dose control purposes only, and not as replacements for the dosimeter designated by the regulatory authority for record keeping purposes (the dosimeter of record). However, an electronic dosimeter considered by the regulatory authority to be of a suitable design for use as the dosimeter of record (having an adequate energy range, sensitivity, linearity, precision, etc.) could effectively serve both purposes.

Task related monitoring

3.25. For dose control in situations in which the radiation field experienced by a worker could increase significantly and unexpectedly (para. 3.9), supplementary dosimeters should be worn which can give early information on short term changes of the radiation field in the working environment. Examples of dosimeters of this type include directly readable pen dosimeters, which can be read during operation and after the working day, and active warning electronic dosimeters, which provide an audible or visual alarm if a certain level of dose or dose rate is exceeded. Most of these warning instruments use Geiger–Müller counters or silicon diode detectors, and are suitable for photon dosimetry above a threshold of 20–80 keV, depending on type. However, these instruments can be misleading in circumstances where weakly penetrating or pulsed radiation is encountered in fields where the dose rate is quite high. In some situations, ambient electromagnetic fields may cause false readings with some designs of electronic dosimeter.

3.26. For operations of short duration in high radiation fields, special monitoring programmes should be designed, including the use of warning devices. In highly non-uniform radiation fields, additional body and extremity dosimeters should be worn (e.g. on the fingers, ankles, knees or head).

Special monitoring

3.27. In situations where individual doses could greatly exceed those expected under normal working conditions, special attention should be paid to the capabilities of dosimeters and to the application of measurements and calculation methods needed for the assessment of effective dose or organ doses.

3.28. In order to avoid the use of a special additional accident dosimeter, the routine personal dosimeter should be capable of providing information on absorbed doses from photons of up to at least 10 Gy [14]. However, it is recognized that certain dosimeters, such as film dosimeters, may not be capable of achieving this at all energies. The wearing of warning dosimeters (or dose rate meters) will usually prevent serious exposures and may help in considerably reducing the dose incurred in the event of accidents. Warning dosimeters need not be very accurate, but should be very reliable, especially in high dose rate fields.

3.29. The subject of dosimetry in the event of criticality accidents involving fissile materials is highly specialized and beyond the scope of this publication. This subject is treated in Ref. [14].

Interpretation of results

Individual monitoring

3.30. For radiation protection purposes the measured operational quantities $H_p(10)$ and $H_p(0.07)$ are interpreted in terms of the protection quantities effective dose E and equivalent dose to the skin and extremities H_T. To do this, realistic assumptions have to be made with respect to the type and uniformity of the radiation field and the orientation of the worker within the field [15]. Under these conditions, the dosimeter reading gives a good estimate of the worker's exposure without underestimating or severely overestimating the relevant protection quantity.

3.31. In cases where the worker moves about the workplace, three types of multidirectional field should generally be considered: (a) with radiation incident predominantly from the front half space (anterior–posterior, or AP geometry) or

(b) from the rear half space (posterior–anterior, or PA), or (c) with radiation incident symmetrically from all directions perpendicular to the body (rotational, or ROT). (A fourth type of geometry, in which radiation is incident isotropically from all directions including above and below (ISO), is rarely encountered in occupational exposure situations.) If the radiation is expected to come from the rear (e.g. for the driver of a vehicle transporting radioactive materials), the dosimeter should be worn on the back. For strongly penetrating radiation it may be assumed that $H_p(10)$ measured by a personal dosimeter worn on the chest approximates the effective dose sufficiently accurately, at least for radiation which is incident from the front or is cylindrically symmetrical (ROT). Thus, one dosimeter worn on the front (or rear) of the trunk generally provides a satisfactory assessment of the effective dose. However, if the dose approaches the relevant limit, an appropriate correction factor should be applied for AP, PA or ROT geometry, based on a knowledge of the radiation and the conditions of exposure. More detailed guidance on the interpretation of dosimeter results obtained under various geometric exposure conditions is available in Ref. [16].

3.32. When further interpretation of personal dose equivalents in other cases is necessary, the following procedures are recommended:

(a) In cases when the procedure discussed in para. 3.31 is not applicable, because information about the uniformity of the radiation field and the movement of the worker cannot be analysed with sufficient accuracy, an investigation using several dosimeters on a phantom may indicate whether an appropriate correction factor applied to the results from a single dosimeter is sufficient, or whether the use of several dosimeters is necessary to meet the objectives of routine individual monitoring. A similar procedure may be used for the reconstruction of an exposure following an accident.
(b) If the radiation fields are markedly inhomogeneous and the expected doses or dose rates are significant, then several dosimeters should be worn.
(c) When multiple dosimeters are used, the equivalent dose can be determined by using algorithms published in Ref. [12]. Additional guidance on the use of multiple dosimeters has been published by the American National Standards Institute (ANSI) [17]. Complex exposure geometries may necessitate a series of calculations with mathematical models to determine the relationship between the dosimeter readings and effective or equivalent dose.

3.33. The uncertainty in estimating effective dose from individual dosimeter readings depends on a number of factors, such as the uncertainty in the measurement of $H_p(10)$, as discussed in Section 5, and in the relationship between $H_p(10)$ and E, as reviewed by a Joint Task Group of the ICRP and ICRU [11].

Workplace monitoring

3.34. Where doses are assessed on the basis of routine workplace monitoring results, that monitoring should be continuous and representative of all working areas within the workplace. The basis for a programme of routine monitoring for external radiation in workplaces should be a comprehensive survey, conducted when any new installation is put into service, or when any substantial changes have been made in an existing installation. The frequency of routine monitoring of the workplace depends on the expected changes in the radiation environment:

(a) Where no substantial alterations to the protective shielding or to the process conducted in the workplace are expected, routine monitoring should be used only occasionally for checking purposes.
(b) Where changes of the radiation field in the workplace are expected which are not likely to be rapid or severe, periodical or occasional checks, mainly at pre-established points, will usually give sufficient and timely warning of deteriorating conditions; alternatively, the results of individual monitoring may be used.
(c) Where radiation fields may increase rapidly and unpredictably to serious levels, a system of warning instruments, either located in the workplace and/or worn individually by workers, will be needed in addition to the personal dosimeters. In these situations, only such warning instruments can reliably prevent the accumulation of large dose equivalents within short working periods.

3.35. For mixed beta–gamma fields in which the relative contributions of beta and gamma to the dose equivalent rate can change substantially as a consequence of minor changes in the operations, it may be necessary to use two types of instrument. Alternatively, one instrument may be used, provided that it is capable of measuring both the ambient dose equivalent $H^*(10)$ and the directional dose equivalent $H'(0.07, \Omega)$.

3.36. If appropriately designed and accurately calibrated instruments are used, it may be assumed that a quantity measured in the workplace can, along with appropriate occupancy data, provide the basis for an adequate estimation of the effective dose to a worker or of the equivalent dose in the organs and tissues of a worker. The operational dose quantities $H^*(10)$ and $H'(0.07,\Omega)$ defined for area monitoring will provide an adequate estimate of effective dose and skin dose. Instruments for area monitoring which are designed to measure quantities defined in free air (e.g. kerma) generally do not have the correct energy response for the measurement of $H^*(10)$.

3.37. It should be noted that the quantity $H^*(10)$ may significantly overestimate the value of $H_p(10)$, as measured with a dosimeter on an individual (and hence effective dose), especially if the field is isotropic. This is because instruments for measuring $H^*(10)$ have an isotropic response, whereas the quantities $H_p(10)$ and E are dependent on the angle of incidence.

3.38. For situations in which the extremities, the unprotected skin of the body or the eyes may be locally exposed to weakly penetrating radiation, the directional dose equivalent $H'(d,\Omega)$ provides an adequate estimation of the equivalent dose to the worker. For multidirectional fields, the instrument should be rotated in the radiation field and the maximum value of dose indicated by the instrument used in order to prevent underestimation of the skin or eye dose. The operator should be aware of the possible existence of point sources or narrow beams which could give rise to misleading readings.

3.39. Survey instruments are calibrated in radiation fields that irradiate the detector volume uniformly, with the centre of the volume used as a reference point. However, many operational fields irradiate the detector in a non-uniform manner (e.g. close to point sources or narrow beams). These situations need special attention and it may be necessary to establish a correction factor that can be applied to the readings to give a corrected dose rate. These factors may be in excess of 100 [18]. One technique is to use a matrix of point sources to simulate source geometries of interest [18].

3.40. In many cases, workplace monitoring is used to provide an upper limit of the equivalent dose received by workers so that no further restrictions on movement within the workplace are needed. In these cases it is assumed that a person will be located for the entire working time in that part of the workplace where the dose equivalent rate is highest. However, for the purpose of dose assessment and records, realistic estimates of occupancy should be obtained and used. In those cases where dose rates may vary significantly with time, occupancy in the workplace should be recorded, so that periods of occupancy can be applied to the relevant dose rate to assess exposure. Additional information on workplace monitoring can be found in the companion Safety Guide [4] and a related ICRP report [5].

Accidental exposure assessment

3.41. As noted in para. 3.29, guidance on the specialized techniques for assessing accidental exposures which significantly exceed the occupational dose limits is outside the scope of this Safety Guide. Particular examples of situations involving acute high level exposure include those associated with criticality accidents or

accidents at industrial irradiation facilities. Assessment of these exposures may begin by using data from personal and workplace monitors, but other sophisticated and highly specialized retrospective dosimetry techniques, such as chromosome aberration analysis, electron spin resonance, accident simulation and computer modelling, may also be necessary.

4. DOSIMETRIC SPECIFICATIONS

GENERAL

4.1. The essential dosimetric performance specifications for personal dosimeters are based on the objectives of individual monitoring [3] (see also Section 3). General guidance on these specifications (e.g. in relation to the dose quantities that should be measured, the overall accuracy that should be obtained, and the degree of monitoring that should be exercised) is given in the companion Safety Guide [3]. Additional information is provided by the ICRP [5, 6], the ICRU [7–9] and the Joint Task Group of the ICRP and ICRU [11]. The OECD Nuclear Energy Agency (OECD/NEA) and the European Radiation Dosimetry Group (EURADOS-CENDOS) have developed guidance on specific problems that have been identified in individual monitoring and have been found to need more clarification [19–22].

4.2. A basic objective of personal dosimetry is to provide a reliable measurement of the operational quantities $H_p(0.07)$ and $H_p(10)$ for almost all practical situations, independent of the type, energy and direction of incidence of the radiation, and with a prescribed overall accuracy. Other dosimeter characteristics which are important from a practical point of view include size, shape, weight and identification. Of particular importance to the measurement of $H_p(0.07)$ and $H_p(10)$ is the dependence of the dosimeter response on the energy and direction of the radiation [23].

4.3. Area monitors used for dose assessment should be type tested and calibrated in terms of the operational quantities $H^*(d)$ and $H'(d)$, and should operate within prescribed criteria for overall accuracy, taking into account the dependence on radiation energy, direction of incidence, temperature, radiofrequency interference and other influence quantities. As with personal dosimeters, the energy and direction dependencies of the response are particularly important.

SPECIFICATIONS FOR PERSONAL DOSIMETERS

Accuracy

4.4. In practice, the overall accuracy criteria for personal dosimeters can be met by establishing criteria for a number of the parameters that influence the performance of the dosimeter, for example its response to radiation type, spectral and directional distribution and environmental influences. This section provides guidance on performance criteria for personal dosimeters for individual monitoring in practical radiation conditions involving exposure to beta, gamma and neutron radiation.

4.5. Information concerning the uncertainties that can be expected in making measurements with individual dosimeters in the workplace is given in para. 251 of ICRP Publication No. 75 [5], which states that:

> "The Commission has noted that, in practice, it is usually possible to achieve an accuracy of about 10% at the 95% confidence level for measurements of radiation fields in good laboratory conditions (Para. 271, Publication 60). In the workplace, where the energy spectrum and orientation of the radiation field are generally not well known, the uncertainties in a measurement made with an individual dosemeter will be significantly greater. Non-uniformity and uncertain orientation of the radiation field will introduce errors in the use of standard models. The overall uncertainty at the 95% confidence level in the estimation of effective dose around the relevant dose limit may well be a factor of 1.5 in either direction for photons and may be substantially greater for neutrons of uncertain energy and for electrons. Greater uncertainties are also inevitable at low levels of effective dose for all qualities of radiation."

4.6. Although it is not explicitly stated by the ICRP, this is generally interpreted to mean that, for a large group of workers using a particular dosimetry system, 95% of the reported annual doses should fall within the indicated limits of acceptable uncertainty. The ICRP statement should be taken to mean that, for doses of the order of the annual dose limits, the apparent annual doses to an individual — $H_p(0.07)$ and $H_p(10)$, as indicated by a number of basic dosimeters, issued regularly during the year and worn on the surface of the body — should not differ by more than –33% or +50% (at the 95% confidence level) from the dose equivalents that would be indicated by an ideal dosimeter worn at the same point at the same times.

4.7. The ICRP has also prescribed a value for the recording level, i.e. the dose above which recording of the doses should be required. It is stated that:

> "The Commission now considers that the recording level for individual monitoring should be derived from the duration of the monitoring period and an annual effective dose of no lower than 1 mSv or an annual equivalent dose of about 10% of the relevant dose limit." (Ref. [5], para. 232)[2]

Doses just below this recording level will not be included in assessments of a worker's dose, and this therefore indicates that an absolute uncertainty R (in terms of dose) given by:

$$R = L \times \frac{\text{Monitoring period in months}}{12} \tag{1}$$

is acceptable, where L is 1 mSv or 10% of the relevant annual equivalent dose limit, as appropriate. This sets a realistic accuracy criterion for the measurement of doses in the low dose range.

4.8. Thus the ICRP recommendations [5] indicate acceptable levels of uncertainty at two dose levels:

(a) In the region near the relevant dose limit, a factor of 1.5 in either direction is considered acceptable;
(b) In the region of the recording level, an acceptable uncertainty of ±100% is implied.

This formulation of acceptable uncertainties leads to a step function, and a smoothing procedure is therefore desirable. To assist in this procedure, a recommendation on acceptable uncertainties in the intermediate dose range is taken from an earlier ICRP publication [24]. This publication recommends that a factor of two in either direction is an acceptable uncertainty for doses of about one-fifth of the relevant dose limit. On this basis, the allowable accuracy interval can be smoothed as a function of dose level [25]. The upper limit R_{UL} is given by:

$$R_{UL} = 1.5 \times \left(1 + \frac{H_0}{2H_0 + H_1}\right) \tag{2}$$

[2] Although this definition of recording level is useful for specification of the necessary accuracy, the ICRP acknowledges that: "In practice, little use is made of recording levels in individual monitoring for external exposure because the measured dose is usually entered directly as a measure of the effective dose. The minimum level of detection should then be used as the recording level with results below that level being deemed to be zero." (Ref. [5], para. 233).

where H_1 is the conventional true dose and H_0 is the lowest dose that needs to be measured, i.e. the recording level (which is equal to R in Eq. (1)). The lower limit R_{LL} is given by:

$$R_{LL} = \begin{cases} 0 & \text{for } H_1 < H_0 \\ \dfrac{1}{1.5}\left(1 + \dfrac{2H_0}{H_0 + H_1}\right) & \text{for } H_1 \geq H_0 \end{cases} \qquad (3)$$

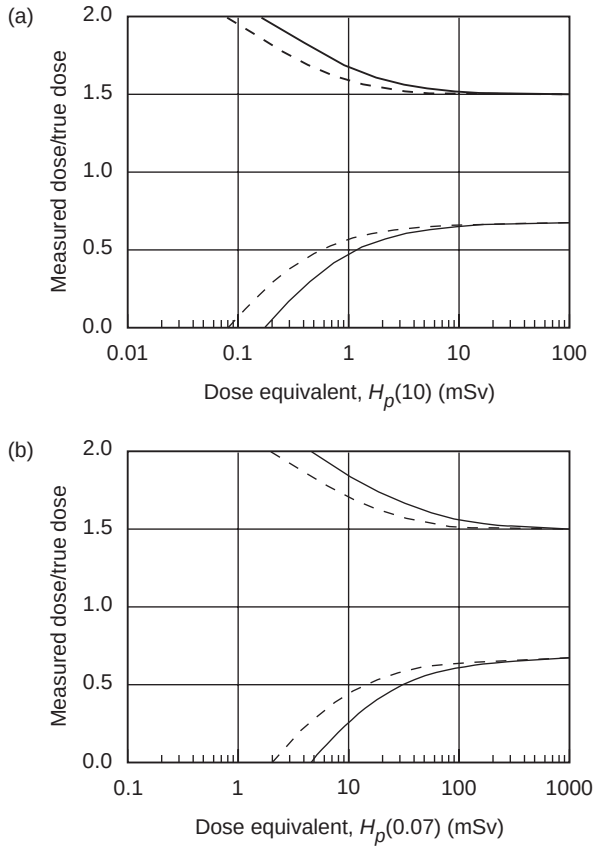

FIG. 2. Acceptable upper and lower limits for the ratio measured dose/conventional true dose as a function of dose: (a) for $H_p(10)$; and (b) for $H_p(0.07)$. (Broken lines: monthly monitoring periods; solid lines: two-month monitoring periods.)

For $H_p(10)$, with monitoring periods of one month or two months, H_0 is 0.08 mSv or 0.17 mSv respectively (using 1 mSv in Eq. (1)). For $H_p(0.07)$, H_0 is 4.2 mSv and 8.3 mSv for one-month and two-month periods respectively (based on 10% of the annual limit of 500 mSv for extremities or the skin). The accuracy intervals are represented graphically in Fig. 2. It should be noted that any changes in the value of the recording level will influence the shape of the trumpet curve in the low dose region. It should also be noted that greatly reduced uncertainties at the lower dose levels are achievable with active (direct reading) dosimeters; for this type of dosimeter, the factor of 1.5 could apply at all relevant dose levels.

Analysis of uncertainties

4.9. The overall uncertainty of a dosimetric system is determined from the combined effects of the two types of uncertainty (Type A, random, and Type B, systematic — see Ref. [26]).

4.10. The standard uncertainty of Type A, U_A, is identified with the standard deviation $\sigma(\bar{x})$ of a series of measurements with observed values x (which form a random distribution with mean $\bar{x}$). Type A uncertainties are those which can, in principle, be reduced by increasing the number of measurements. Typical sources of Type A uncertainty are:

(a) Inhomogeneity of detector sensitivity;
(b) Variability of detector readings due to limited sensitivity and background;
(c) Variability of detector readings at zero dose.

4.11. Type B uncertainties, U_B, are those which cannot be reduced by repeated measurements. The following sources are usually considered to cause uncertainties of Type B:

(a) Energy dependence;
(b) Directional dependence;
(c) Non-linearity of the response;
(d) Fading, dependent on ambient temperature and humidity;
(e) Effects due to exposure to light;
(f) Effects due to exposure to types of ionizing radiation that are not intended to be measured by the dosimeter;
(g) Effects from mechanical shock;
(h) Calibration errors;
(i) Variation in local natural background.

4.12. The effects of Type B uncertainties often appear with a certain probability distribution and behave like Type A uncertainties. For example, for irradiation at a certain angle of incidence, a personal dosimeter will incur a systematic error due to its variation of response with angle. However, when the same dosimeter is worn by an individual working within the individual's radiation environment, it is irradiated from a range of angles and the resulting uncertainty behaves more like one of Type A. It is recommended by ISO [26] that Type B uncertainties be characterized by standard deviations and variances, and that Type A and Type B uncertainties be combined by addition in quadrature to obtain an overall uncertainty. As the total uncertainty includes both random (Type A) and systematic (Type B) uncertainties, it is a necessary assumption in doing this that there is no group of workers, even if consisting of only a few per cent of a large group, for whom the conditions of the workplace imply that the systematic uncertainties exceed the random uncertainties mentioned above.

4.13. The combined uncertainty U_C may then be expressed in the form:

$$U_C = \sqrt{U_A^2 + U_B^2} \tag{4}$$

To obtain a numerical value for U_B, one must evaluate the separate uncertainties $U_{B,i}$ for each individual uncertainty i. U_B can then be obtained from:

$$U_B = \sqrt{\sum_i U_{B,i}^2} \tag{5}$$

4.14. By convention, it is often assumed that Type B uncertainties can be represented by a rectangular probability density distribution, from which the standard uncertainty can be obtained by:

$$U_{B,i} = \frac{a_i}{\sqrt{3}} \tag{6}$$

where a_i is the half-range of values that parameter i is assumed to take.

4.15. Equations (4), (5) and (6) then give:

$$U_c = \sqrt{\left(U_A^2 + \frac{1}{3}\sum_i a_i^2 \right)} \tag{7}$$

4.16. The combined standard uncertainty thus still has the character of a standard deviation. If, in addition, it is believed to have a Gaussian (normal) probability density, then one standard deviation each side of the mean corresponds to confidence limits of about 66%. Therefore, it is often necessary to multiply the combined standard uncertainty by a suitable factor, called the coverage factor *k*, to yield an expanded uncertainty (also known as the 'overall uncertainty'). Typical values of the coverage factor would be 2 or 3, corresponding to confidence limits of approximately 95% or 99% respectively. The numerical value taken for the coverage factor should be clearly indicated.

Performance criteria

4.17. The performance criteria presented in paras 4.18–4.20 should be used for demonstrating compliance with the ICRP recommendation on overall accuracy. They are fully consistent with those recommended by the European Commission [22]. However, it is recognized that national requirements may make it necessary to adopt other criteria, which may be more stringent or have more mathematical rigour, for purposes of accreditation and performance testing.

4.18. Equation (4) can be used to determine a single value of the overall uncertainty of a dosimetry system that can be used for demonstrating compliance with the ICRP's recommendation on overall accuracy (i.e. an uncertainty interval of –33% to +50% for doses near the dose limit). The equation may also be used to define the performance criteria necessary to satisfy the ICRP's accuracy criteria. An allowable uncertainty of –33% to +50% of the dose being measured can be met at the 95% confidence level (corresponding to a coverage factor of 1.96) if:

$$1.96\, U_c \leq 0.5 \times (0.33 + 0.50) \tag{8}$$

and, accordingly from Eq. (4):

$$U_C = \sqrt{U_A^2 + U_B^2} \leq 0.21 \tag{9}$$

where U_A and U_B should be expressed in terms of the performance quotient $(H_m - H_t)/H_t$, with H_m and H_t indicating the measured and conventional true doses respectively. Thus, the acceptance of a dosimetry system does not imply compliance with specific criteria for each uncertain parameter separately, but only that the combined effects from the uncertainties are within a certain limit.

4.19. In practice, the uncertainties caused by the energy and angular dependence of the response of the dosimeter receive more attention than any other source of error, because the effects from all other uncertainty components are assumed to be much smaller. Therefore, it is convenient to differentiate between the Type B uncertainty due to the energy and angular dependence, characterized by the resultant standard deviation $U_{B(E,\alpha)}$, and the uncertainties due to all other Type B uncertainties, characterized by the resultant standard deviation $U_{B(0)}$. Equation (5) gives:

$$U_B = \sqrt{U_{B(E,\alpha)}^2 + U_{B(0)}^2} \tag{10}$$

and furthermore, from Eq. (9):

$$\sqrt{U_A^2 + U_{B(E,\alpha)}^2 + U_{B(0)}^2} \leq 0.21 \tag{11}$$

4.20. From Eq. (11), Δ, the maximum allowable value for $U_{B(E,\alpha)}$, can be calculated if U_A and $U_{B(0)}$ are known. Hence, for doses near the dose limit:

$$\Delta = \sqrt{0.21^2 - U_A^2 - U_{B(0)}^2} \tag{12}$$

For example, if it is assumed that $U_A = U_{B(0)} = 0.10$, then the maximum allowable uncertainty for the combined energy and angular response at a 95% confidence level equals $\pm 1.96\Delta$, and the range ($\pm 1.96\Delta$) equals ± 0.30.

Other criteria

4.21. In addition to the numerical criteria for the performance of personal dosimeters, criteria concerning their use in practice and economic factors should be considered. Criteria of this kind include, but are not limited to:

(a) Low cost;
(b) Low weight, convenient size and shape, convenient and reliable clips;
(c) Adequate mechanical strength and dust tightness;
(d) Unambiguous identification;
(e) Ease of handling;
(f) Reliable readout systems;
(g) Reliable supplier who will continue to provide dosimeters over long periods;
(h) Adaptability to various applications, e.g. measurement of body dose and extremity dose;
(i) Suitability for automatic processing.

4.22. In particular, for extremity dosimetry, attention should be paid to the mechanical strength of the dosimeters and to their resistance to environments with extreme temperatures and humidity, as these dosimeters are often used under extreme working conditions. Where the extremities, for example fingertips, come into close proximity with the source, large variations in dose rate occur over the surface of the hand and it is essential to support the detector at the front surface of the finger. Small detectors that can be fixed to the finger with tape or kept in finger covers or finger rings should be used for this purpose.

SPECIFICATIONS FOR WORKPLACE MONITORING

4.23. Assessments of individual doses from exposure to external radiation should in general be made using personal dosimeters. This will be the normal method of complying with national regulatory requirements.

4.24. As discussed in para. 3.6, however, there may be cases in which such doses need to be assessed from workplace monitoring results. In such circumstances, it may be necessary to demonstrate the correlations between values of dose rate and individual or group occupancy data. Detailed records of occupancy may be needed for areas in which dose rates vary significantly with time.

4.25. The uncertainties that are acceptable in workplace monitoring and record keeping depend on the scope and purpose of the monitoring programme. In the following sections, some information is given on acceptable uncertainties and record keeping for dose assessment purposes.

Accuracy and performance criteria

4.26. To satisfy the criteria defined in Section 3 for interpreting workplace monitoring results in terms of $H^*(d)$ and $H'(d)$, criteria for a number of parameters influencing the performance of the monitor (e.g. the dependence of response on radiation type, spectral and directional distribution and environmental influences) have to be established. For guidance on performance criteria for personal dosimeters see paras 4.17–4.20. It is normally considered that the uncertainty of area monitors needs to be within ±30%. This value applies to performance under laboratory test conditions (standard test conditions), and may not be achievable under normal operational conditions. However, certain parameters should be considered differently for the analysis of uncertainties. As an example, the response of an instrument designed to measure ambient dose equivalent should be isotropic, whereas an instrument designed to measure directional dose equivalent should have the same angular response as H'.

Other criteria

4.27. In addition to the energy and angular response, several factors can influence the accuracy and reliability of measurements. The following should be assessed as part of a type test (this list may not be exhaustive):

(a) Ability to withstand shock and vibration;
(b) Independence of response to atmospheric pressure;
(c) Dust tightness;
(d) Water resistance;
(e) Independence of response to dose rate;
(f) Correctness of response in pulsed fields (as applicable);
(g) Insensitivity to electric and magnetic fields;
(h) Stability under extremes of temperature and humidity;
(i) Insensitivity to radiation types not to be measured;
(j) Response time;
(k) Stability of response over time (minimal drift);
(l) Sensitivity and coefficient of variation.

Other features should be considered as appropriate, including weight, cost, ease of handling and reading, and the need for reliable and continuing maintenance/support.

Operational use of workplace monitors

4.28. Workplace monitors should be appropriate for the intended use. Care should be taken to verify that the instrument is appropriate for the type of radiation to be measured and that its results are not seriously affected by other radiation types that might be encountered. Provision should be made for the continuous monitoring of radiological hazard levels in areas where sudden unexpected increases might result in a significant dose to an individual. These provisions will include permanently installed monitoring devices. The following important characteristics of area monitors should be considered:

(a) Monitors normally indicate the dose equivalent rate (although additional functions are sometimes performed, such as the calculation of the accumulated dose or the safe occupancy time remaining);
(b) The dose rate range of the instrument should be adequate to cover the range of dose rates that could reasonably be encountered in practice;
(c) When a monitor is exposed beyond its range, the indication should remain high and off-scale.

4.29. Battery checks, zeroing and tests to demonstrate an adequate response should be carried out frequently as part of a quality assurance programme to ensure that the equipment is still functioning satisfactorily and has suffered no obvious damage.

4.30. Fixed monitors should be fitted with appropriate audio and/or visual alarms to warn of unacceptable conditions.

4.31. Area monitoring may also be performed with passive dosimeters such as TLDs, which provide a wide dynamic range. However, they give no information about the time dependence of the radiation field and so are not ideally suited to dose assessment applications, particularly where dose rates might vary significantly with time. Spectrometers are a useful supplement to dosimetry, and are needed when a lack of information about the radiation spectrum might give reason to doubt the performance of area monitoring.

Siting of workplace monitors

4.32. Careful consideration should be given to the selection of sites for workplace monitoring and to the number of instruments deployed. If the radiation field is well characterized, is uniform in space and does not vary significantly with time, it may be possible to justify the installation of only a few or even a single workplace monitor. In contrast, more radiation monitoring instruments will be needed if the dose rate varies rapidly in time and space. The use of portable instruments may be helpful, provided that adequate supporting documentation is maintained to define the place and time of measurements. Sites selected for workplace monitoring should be representative of worker occupancy as determined on the basis of expected operational activities.

5. TYPE TESTING

GENERAL

5.1. Type testing of a dosimetry system involves testing the performance characteristics of the system as a whole under a series of irradiation and storage conditions. In particular, those sources of uncertainty discussed in Section 4 should be quantified. This largely involves investigation of the variation of dosimeter response with the

energy and the direction of incidence of the radiation beam. However, it also includes consideration of other dosimetric characteristics, such as the linearity of response, the range of measurable doses, the ability of the system to perform satisfactorily over a reasonable range of temperature and humidity conditions, and the ability to respond properly in high dose rates and in pulsed radiation fields. Type testing also includes tests of a more general nature, such as the ability of the system to operate satisfactorily in a reasonable range of electric and magnetic fields, and its ability to withstand mechanical shock and vibration. The results of type testing should be analysed in terms of performance criteria (see paras 4.17–4.20), and are intended to demonstrate whether these can be met in practice, bearing in mind the range of values of the various factors at the facility in which the dosimeters or instruments are to be used.

5.2. Type testing of workplace monitoring instruments is necessary to demonstrate the suitability of an instrument to perform adequate measurements in the workplace environment.

5.3. Type testing may be undertaken by secondary standards laboratories whose measurements are traceable to primary standards.

5.4. The International Electrotechnical Commission (IEC) specifies in all of its standards the test conditions to be used for type testing (see, for example, Ref. [27]). Parameters other than the influence quantity under investigation should be set to the fixed values specified in the reference conditions. The IEC reference and standard test conditions are listed in Annex IV. Detailed recommendations on calibration procedures for individual and workplace monitors are given in Ref. [28].

TYPE TESTING OF PERSONAL DOSIMETERS

Type testing for energy and angular response

5.5. The response with respect to radiation energy and angle of incidence is a crucial characteristic of a personal dosimeter (see Section 4). Dosimeters should be tested to determine how well they conform to the energy and angular response characteristics demanded by the quantity or quantities to be measured.

5.6. Because the definition of the operational quantity for individual monitoring $H_p(d)$ specifies the measurement of dose equivalent within the body, dosimeters should be type tested on an appropriate phantom to emulate backscatter from and attenuation by the person's body. This assumes that if the dosimeter performs adequately on the phantom, it would also do so on an individual's body.

5.7. The current ICRU guidance states that personal dosimeters should, for the purpose of type testing, be irradiated on a slab phantom 30 cm × 30 cm square and 15 cm thick, made of tissue substitute. The appropriate energy and angular response is determined by the calculation of $H_p(d)$ for various energies and angles of incidence. The results are used to relate the response needed for $H_p(10)$ and $H_p(0.07)$ to that needed for one of the physical quantities, such as absorbed dose to air or air kerma, by means of sets of conversion coefficients. Conversion coefficients for monoenergetic photons for the 30 cm × 30 cm × 15 cm ICRU tissue equivalent slab phantom are compiled in Tables V–1 and V–2 (Annex V) [11]. The International Organization for Standardization (ISO) has specified the conditions and characteristics of standard X ray fields to be used for calibration purposes and the phantoms that should be used with these radiation types [29–31]. The specifications of those fields are summarized in Table V–3. Conversion coefficients to be used for the ISO reference photon radiations are presented in Table V–4 [31] and those for the type testing of neutron dosimeters are shown in Table V–5 [11].

5.8. Conversion coefficients for electrons are presented in Table V–6. The use of computed conversion coefficients for dosimeter type testing is less relevant for beta radiation, because the dose rate in calibration beams either is known for secondary standard sources, or is measured with an extrapolation chamber (in terms of the dose equivalent rate at a depth of 0.07 mm — and at 10 mm for the more energetic beta emitters — in a tissue equivalent medium which provides the same backscatter and attenuation as soft tissue). The results obtained are virtually identical to those that would be obtained in the ICRU tissue slab because the range of electrons from common beta emitters is relatively short. Hence, the values can be taken as a measurement of $H_p(0.07)$ and $H_p(10)$. Extrapolation chambers may be used, therefore, as primary or secondary standard instruments for measuring these quantities for beta radiation.

5.9. A practical problem arises because ICRU tissue substitute cannot be produced exactly as specified. Appropriate backscatter phantoms specified by the ISO should therefore be used during irradiation of the dosimeters for the whole body (slab[3]), arm or leg (pillar[4]) and finger (rod[5]) [31]. The backscatter characteristics of these

[3] A 30 cm × 30 cm × 15 cm waterfilled container with polymethylmethacrylate (PMMA) walls 1 cm thick. One 30 cm × 30 cm entrance window is 2.5 mm thick.

[4] A 30 cm long water filled PMMA cylinder, 73 mm outside diameter with a 2.5 mm thick wall.

[5] A 30 cm long PMMA solid rod, 19 mm in diameter.

phantoms are acceptably close to ICRU tissue for both photon and neutron radiations. More detailed guidance on the use of these phantoms for calibration purposes is given in Ref. [28].

5.10. The definition of $H_p(10)$ and $H_p(0.07)$ implies a response which varies with angle if the radiation is described in terms of particle fluence. This is a result of the increase with angle of the attenuation in the material overlying the point where the quantity is defined (because radiation incident at an angle will pass through more material to reach a given depth than radiation incident normally to the surface). This extra attenuation is small for $H_p(0.07)$, except for beta particles, but is substantial for $H_p(10)$ for both photons and neutrons, especially at lower energies. The appropriate variations in response with angle are described by observing the variation of $H_p(10)$ with angle. In Fig. V–1, the ratios $H_p(10,\alpha)/H_p(10,0°)$ and $H_p(0.07,\alpha)/H_p(0.07,0°)$ are plotted against energy for photons for a number of representative angles α. The ratios $H_p(10,\alpha)/H_p(10,0°)$ for neutrons are plotted in Fig. V–2 [11].

5.11. The type testing procedure can be summarized as follows, using as an example the irradiation of dosimeters with photons to measure the quantity $H_p(10)$:

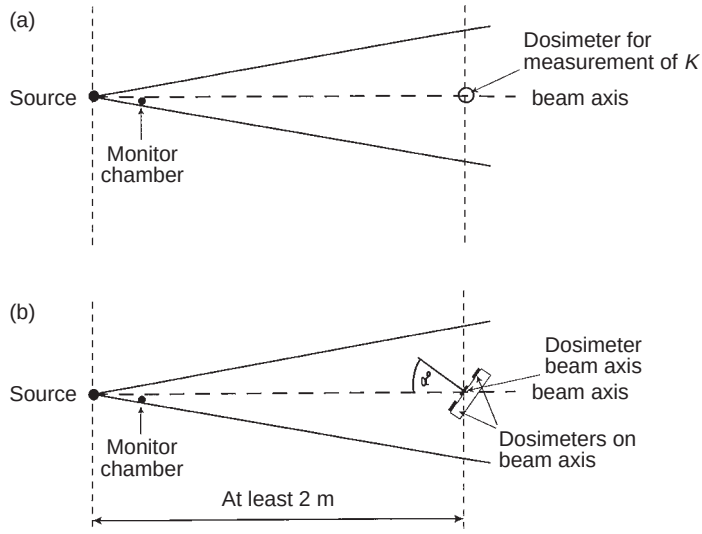

FIG. 3. Exposure arrangement for dosimeter type testing.

(1) Choose the mean photon energy from the ISO reference radiations given in Table V–3 (Annex V), and set up the radiation beam, together with a monitor chamber (Fig. 3(a));

(2) Design the collimation such that the monitor chamber, the slab and the dosimeters can be completely enveloped by the beam at a distance of at least 2 m;

(3) In the absence of the slab and dosimeters, and for a given indication D on the monitor chamber, measure the air kerma (K_a) using an instrument such as an ion chamber, at the position to be occupied by the reference point of the dosimeter [27] when it is placed on the phantom during the actual irradiation. This point should be at least 2 m from the source (Fig. 3(a));

(4) Multiply the measured air kerma by the appropriate conversion coefficient (C) for $H_p(10,\alpha)$ from Table V–1, i.e. the value of $H_p(10,\alpha)$ is given by $K_a \times C$ for a monitor indication of D. Each unit on the monitor chamber thus corresponds to an $H_p(10,\alpha)$ value of $(K_a \times C)/D$;

(5) Place the slab phantom and dosimeters in the beam such that the beam is incident on the dosimeters at angle α, with the reference point of the dosimeter on the beam axis[6] at the position at which the air kerma was measured in (3) above (Fig. 3(b));

(6) Choose the dose equivalent H to be delivered to the dosimeters. Irradiate the arrangement until the monitor chamber indicates the desired value of $(H \times D)/(K_a \times C)$;

(7) Process the dosimeters and compare their readings with the conventional true dose equivalent H for $H_p(10,\alpha)$.

5.12. Performance criteria for the energy and angular response of a personal dosimeter are usually specified for each parameter separately, for example for the energy response at normal radiation incidence and for the angular response at specific energies. However, the effects of these two parameters on the uncertainty are interrelated, and therefore criteria should also be specified for their combined effect. One approach is to specify criteria for the angular response which should be met for the whole range of energies to be monitored. In practice, some averaging over different angles of radiation incidence will occur during a monitoring period. It is therefore considered satisfactory to specify criteria for the mean value of the responses for a number of angles of incidence, provided that the overall performance criteria satisfy Eq. (9).

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[6] If multiple dosimeters are irradiated simultaneously in the above manner, a correction for the non-uniform distance to the source may be necessary for those positioned off the beam axis. It is suggested that the phantom be turned halfway through the exposure such that the dosimeters are irradiated at angle $-\alpha$.

5.13. The following procedure may be used to determine experimentally the combined energy and angular response of a personal dosimeter (see para. 4.4). Energy response curves should be established for both $H_p(0.07)$ and $H_p(10)$ at incident angles of 0°, ±20°, ±40° and ±60° from the normal. Separate measurements should be made for each angle for both the horizontal and vertical rotation planes unless the dosimeter is cylindrically symmetrical. Measurements should be made using the reference radiations specified in ISO standards, within the energy ranges:

(a) 15 keV to 1.5 MeV for photons;
(b) 0.2 MeV to 3.5 MeV (E_{max}) for beta particles;
(c) thermal to 15 MeV for neutrons.

The conversion coefficients for the photon and beta energies to be included in the measurements should be selected from those listed in Annex V, taking into account the intended use of the dosimeter. These measurements are usually made on a rectangular, water filled backscatter phantom [31]. However, if angles greater than ±60° are of particular concern, testing may need to be performed using a more realistic phantom (for example, an elliptical cylinder).

5.14. To calculate the mean energy response over the four angles 0°, 20°, 40° and 60°, for a truly isotropic radiation field it would strictly be necessary to weight the results for each angle by the solid angle subtended at the dosimeter. In practice, however, the irradiation conditions are more likely to be rotationally symmetrical, in which case the responses at each angle should have equal weighting. Thus, a response curve can be constructed for each type of radiation by calculating and plotting the average angular response for each energy ε [32, 33]:

$$\bar{R}_\varepsilon = 0.25(R_{\varepsilon,0} + R_{\varepsilon,20} + R_{\varepsilon,40} + R_{\varepsilon,60}) \quad (13)$$

where $R_{\varepsilon,\alpha}$ is the response at energy ε and incident angle α, obtained from:

$$R_{\varepsilon,\alpha} = \frac{\left(H_{\varepsilon,\alpha}\right)_m}{\left(H_{\varepsilon,\alpha}\right)_t} \quad (14)$$

where $(H_{\varepsilon,\alpha})_m$ is the measured dose and $(H_{\varepsilon,\alpha})_t$ is the conventional true value.

5.15. If $\bar{R}_\varepsilon$ is assumed to represent the average response at energy ε for the range of angles of incidence of radiation during the monitoring period, the values $\pm\left|\bar{R}_\varepsilon - 1\right|$ may be taken as an indication of the uncertainty of the energy response.

5.16. From Eq. (11), the allowable limits ±1.96Δ are evaluated for the combined uncertainty (at the 95% confidence level) related to the combined energy and angular response of the dosimeter. A dosimeter may therefore be considered to perform satisfactorily if the condition:

$$\left| \bar{R}_{\varepsilon} - 1 \right| \leq 1.96\Delta \tag{15}$$

is fulfilled for all of the irradiation energies prescribed for the test and the overall performance criteria satisfy Eq. (9). It should be recognized that other appropriate approaches to the assessment of the angular response of dosimeters have been adopted by national standards laboratories.

Type testing for other important characteristics

5.17. In addition to its response to radiation energy and angle of incidence, there are a number of other characteristics of a dosimetry system that should be considered in type testing. The suitability of a dosimetry system should be demonstrated by analysing the results of the type tests using Eq. (12). Methods of testing for these characteristics have been published by national and international standards organizations [27, 34, 35]. Tests should be made for the characteristics listed in para. 4.11.

TYPE TESTING OF WORKPLACE MONITORS

5.18. Procedures for the measurement of the energy and angular responses of workplace monitoring instruments are similar to those for type testing individual dosimeters, except that radiation exposures in workplace monitoring would normally be free-in-air (i.e. without phantom). Information on type testing and instrument performance can be found in the references listed in Annex VI. The conversion coefficients to be used for $H^*(d)$ and $H'(d)$ are given in Tables V–7 and V–8 [11].

5.19. The IEC issues standards for most types of radiation protection monitoring equipment. Examples of these standards are given in Annex VI. These standards not only give the performance specifications to be met but also describe the methods of type testing to be undertaken. Tests are prescribed for determining the radiological performance (e.g. linearity, energy dependence, angular response) and the environmental, electrical and mechanical performance.

6. PRE-USE AND PERIODIC TESTING

6.1. Instruments should be tested before they are first used to ensure that they conform to type test data. This testing should be designed to identify credible faults such as miscalibration or incorrect assembly of the detector. Pre-use testing also provides a baseline for subsequent routine testing. It is normally possible to select a restricted series of tests which can provide adequate confidence in an instrument's performance. Detailed recommendations are presented in Ref. [28]. The organization carrying out such tests should be recognized by the regulatory authority as competent to do so.

6.2. Periodic testing of workplace monitoring or survey instruments should be carried out at least once a year, and should involve a subset of the tests used in pre-use testing, selected to indicate any deterioration in an instrument's performance. Examples of reference radiations that may be used are:

(a) For photon dose rate monitors, the 0.662 MeV gamma from ^{137}Cs;
(b) For neutron dose rate monitors, ^{241}Am–Be neutrons;
(c) For beta dose rate monitors, the 0.662 MeV gamma from ^{137}Cs plus a low energy beta source;
(d) For beta contamination monitors, betas at or below the minimum energy for which the monitor is to be used.

6.3. Following testing, a sticker should be attached to the instrument giving relevant information, including the organization performing the test, the test certificate number, and the date of the test or date when the next test is due, as appropriate. Tests should be carried out by an organization that maintains reference radiation fields traceable to the national standards body.

6.4. Testing should cover the range of dose rates that could reasonably be encountered. Ranges for which an instrument has not been tested should be clearly identified and documented.

7. PERFORMANCE TESTING

GENERAL

7.1. In addition to the type testing of a personal dosimetry system, in which the whole performance of the system is carefully analysed in order to verify that it meets the accuracy criteria (Section 4), it is necessary to demonstrate that this standard of

performance is maintained continuously. Three categories of testing should be carried out regularly for this purpose, as follows:

(a) Approval performance testing is a means of demonstrating that the overall dosimetric performance standard is maintained;
(b) Routine testing or calibration is a means by which the sensitivity, precision and accuracy are verified for a single radiation type and energy;
(c) Testing connected to the QA programme.

The initial approval of a dosimetry service by the regulatory authority should include a combination of type testing and approval performance testing.

APPROVAL PERFORMANCE TESTING

7.2. Performance testing as part of approved procedures is carried out to demonstrate that the essential performance specifications are routinely maintained. The results should confirm the type testing data.

7.3. An approval performance testing programme may be subdivided into different irradiation categories to suit different classes of dosimeter design, i.e. based on the radiation types and energy ranges covered by the dosimeters. Each test may include a range of different energies and angles of incidence of the radiation, and an appropriate distribution of doses over the range from 0.2 mSv to at least 100 mSv to test the overall performance of the system. An extended dose range may be necessary if the dosimeter is used for approved dosimetry in accident situations. The results of this testing should satisfy the overall accuracy criteria specified by the ICRP, so that 95% of the results fall within the acceptable accuracy band defined in Section 4 (Fig. 2).

7.4. Approval performance tests should be carried out at regular intervals, in accordance with regulatory requirements, by an external test facility, and may be used in the initial and/or ongoing approval for the operation of the dosimetry service.

ROUTINE PERFORMANCE TESTING

7.5. The purpose of routine performance testing in individual monitoring is to test the accuracy and precision of the dosimetry system for measurement of doses at a single energy, usually that of the calibration source, e.g. ^{137}Cs or ^{60}Co gammas for photon dosimeters. The precision (given by the standard deviation of a single measurement), and the accuracy (the average deviation of the readings from the

TABLE I. SUMMARY OF TESTING FOR INDIVIDUAL DOSIMETERS

Type of test	Test performed by	Frequency of testing
Type	Manufacturer or authorized type testing organization	Once, typically prior to marketing to end users
Approval	Organization authorized by regulatory authority	Annually
Routine	End user or service	Monthly
QA	End user or service	Daily, prior to startup of dosimeter processing

TABLE II. SUMMARY OF TESTING FOR WORKPLACE OR SURVEY INSTRUMENTS

Type of test	Test performed by	Frequency of testing
Type	Manufacturer or authorized type testing organization	Once, typically prior to marketing to end users
Pre-use	Manufacturer, end user or authorized testing organization	Once, prior to placing instrument into service
Periodic	End user or authorized calibration organization	Annually or more frequently, dependent upon stability of instrument and intended use
Performance	Authorized performance testing organization	As specified by regulatory authority, typically every 2–3 years

conventional true value), should be tested at different dose levels. The results of the tests should at least fulfil the accuracy criteria given in Eqs (2) and (3) and shown in Fig. 2. This type of test also serves to normalize the overall sensitivity of the system. Routine performance tests are normally carried out by the service itself, and should be repeated at regular intervals, preferably once per month. In contrast, QA tests to monitor specific aspects of system performance are generally performed daily.

7.6. Workplace monitoring instruments should be frequently source-checked to ensure proper functioning. The choice of the source and ranges tested should be appropriate for the type of monitoring being conducted.

7.7. Summaries of the recommended testing programmes for individual dosimeters and for workplace instruments are given in Tables I and II respectively.

8. DOSE RECORD KEEPING AND REPORTING

GENERAL

8.1. Dose record keeping is the setting up and keeping of individual dose records for radiation workers. It is an essential part of the process of monitoring the exposure of individuals to radiation and supports the overall objectives of monitoring (Section 3). General guidance on record keeping and reporting is given in a related Safety Guide [3]. Further information that relates specifically to doses from external radiation is given below.

8.2. Records should provide support for decision making, demonstrate and facilitate regulatory compliance, provide for the reconstruction of results at any later time, and facilitate co-ordination with other records, such as those for internal monitoring and area monitoring. They should therefore be easily retrievable and be protected against loss. Such protection is usually attained by maintaining duplicate sets of records in well separated locations, so that both copies cannot be destroyed in a single incident. Records should be consolidated for each monitored individual, identified by site, purpose, date and originator, and should be legible and intelligible to a qualified person, complete and accurate. Consideration should be given to any applicable national requirements or international agreements concerning the privacy of individual data records.

RECORD KEEPING FOR INDIVIDUAL MONITORING

8.3. The purpose of record keeping, the nature and scope of the records and the extent of record keeping systems depend on national requirements. The records should include the results of individual monitoring for both external radiation and intakes of radioactive material.

8.4. Modern individual monitoring services for external radiation, particularly the larger ones, have adopted a high degree of automation, often using fully integrated systems linking the dose record keeping to the labelling and issuing of dosimeters and their subsequent dose assessment. Such automated systems, especially if the

dosimeters are labelled with the wearer's name (thus providing a further line of defence against errors in issuing dosimeters), offer a high degree of integrity — and hence quality — to the service being provided.

8.5. When recording individual occupational doses, it is customary not to include doses due to situations that can reasonably be regarded as being outside the responsibility of the operating management, i.e. exposure which has been excluded or which arises from sources that have been exempted by the regulatory authority. However, doses due to work with materials containing significant levels of naturally occurring radionuclides are regarded as the responsibility of the operating management and should therefore be included in an individual's recorded occupational dose [6].

8.6. Because it is virtually impossible when evaluating the readings of personal dosimeters to distinguish between photon and beta radiation, it is not sensible to attempt to identify (and report) the beta and gamma components of $H_p(0.07)$ separately. However, because the different types of high LET radiation have different quality factors, it is advisable when monitoring in terms of $H_p(10)$ to record neutron doses separately. It should be remembered that photon, neutron and beta doses are to be combined to determine the total personal dose equivalent.

8.7. If a dose assessment is not available for a period when a radiation worker was (or should have been) monitored — which may happen when a dosimeter has been damaged or lost, or recorded a dose that, on investigation, is declared invalid — the record keeping system should allow the introduction of doses estimated or assessed by an authorized person. These dose estimates should be marked in such a way that they can be distinguished from official dose measurements made by the approved monitoring service.

8.8. For those individuals who need to use extremity dosimeters, separate records should be kept for the exposure of each extremity. The procedure of record keeping is made more complicated, however, when the extremity dosimeters are worn only for certain periods during the year. In these situations, in order to achieve a complete dose record for each extremity, the records should contain the readings of the extremity dosimeters for those periods when they were worn, and the reading of $H_p(0.07)$ from the body dosimeter for those periods when extremity dosimeters were not worn.

RECORD KEEPING FOR WORKPLACE MONITORING

8.9. Records documenting the designation and location of controlled and supervised areas should be kept. Records should also be kept of radiation surveys, including the

date, time and location, and the radiation levels measured, and any comments relevant to the measurements made. Records should identify the instrument(s) used and the individual performing the survey.

8.10. A suitable record of the calibration of monitoring equipment should include identification of the equipment, the calibration accuracy over its range of operation for the type(s) of radiation that it is intended to monitor, the date of the test, identification of the calibration standards used, the frequency of calibration, and the name and signature of the qualified person under whose direction the test was carried out.

REPORTING OF INFORMATION TO MANAGEMENT

8.11. The procedures and criteria to be used for reporting individual and workplace monitoring results should be clearly specified by the management or regulatory authority. Information reported should be clearly identifiable and understandable. Normally, only final results are reported.

8.12. In accident situations, or for an exposure that may be close to or above a regulatory limit, interim results should be supplied so that appropriate administrative and other response actions can be instituted. The results should include the result of the measurement and the implied exposure, based on appropriate conversion coefficients. Recommendations for follow-up monitoring and for workplace restrictions may be made if appropriate. The source of the information reported should be clearly identified, as should a point of contact for any additional information. Finally, the uncertainty in the measured and computed values should always be reported, accompanied by a statement of which sources of variability have been considered, quantified and propagated in the quoted uncertainty.

9. QUALITY ASSURANCE

REQUIREMENTS

9.1. The continued effectiveness of any radiation protection programme relies on those in charge implementing its various components, including the adoption of an effective QA programme. General QA requirements related to occupational exposure are given in the BSS [2] and general guidance is given in the related Safety Guide [3].

The following section deals more specifically with issues related to the assessment of exposure to external radiation.

IMPLEMENTATION AND MANAGEMENT

9.2. The nature and extent of the QA programme should be consistent with the number of workers monitored, and the magnitude and likelihood of exposures expected in the workplaces covered by the monitoring programme. Of particular importance is the ISO/IEC Guide 25 [36], which is used by many regulatory bodies to accredit testing and calibration programmes.

9.3. All persons involved in the external exposure assessment programme are responsible for its quality, and therefore for implementing its QA programme and quality control (QC) procedures. Responsibility for the quality of a particular operation should be delegated to the person actually performing the operation. Such persons should be actively involved in the development of QC procedures, and trained in methods of detecting non-compliance. Management should encourage staff to detect, report and correct non-compliance. Quality assurance built into a programme from the bottom up is more effective than QA imposed from the top down. For the QA programme to be effective, all personnel must be confident that management expects and encourages performance that meets its objectives.

9.4. The dosimetry service should have a designated QA representative. This representative should monitor QC procedures, perform internal audits of the programme, and be responsible for training all personnel in QA, both in general terms and in the specific quality aspects of their individual work.

9.5. Implementation of a QA programme and QC procedures requires an understanding of the complete dosimetry system, from the manufacturing of equipment and materials to the use of dosimeters in the workplace.

9.6. National regulations may require that facilities concerned with measurement and external dose assessment be accredited. Such accreditation programmes will include specifications of the QA and QC measures to be implemented. Details of the QA system management, organization and administration may be related to national legislation and may depend on the nature of the service, for example:

(a) The number of dosimeters issued;
(b) The number of customers served;
(c) The categories of dosimeter used (basic, discriminating, neutron, etc.);

(d) The dosimetric method(s) applied (film, TLD, RPL, track etch, etc.);
(e) The choice of issuing periods offered;
(f) The level of automation.

Documentation

9.7. Essential components of the quality system, including all methods and procedures set up to control the various processes within the service, should be documented. Documentation should include the results of all tests that relate to the quality of the dose assessment process, such as type testing of dosimetry systems and validation of equipment performance.

9.8. An important part of this documentation is a quality handbook, which should cover all aspects of the established quality system in a concise and practical way. Appropriate parts of the documentation should be made available to staff members.

Training of personnel

9.9. Adequate training of dosimetry service personnel is essential to ensure that they can perform their jobs reliably. Such training should include:

(a) Their particular responsibility within the quality system;
(b) The basic philosophy and strategy of external dose assessment;
(c) The principles and details of the methods and procedures used, and their limitations;
(d) The technical details and potential problems of the processes in which they are involved;
(e) The relation their work has to other parts of the programme;
(f) Guidance on recognition and reporting of problems that may arise;
(g) Knowledge of the overall quality system and its objectives.

Laboratory facilities

9.10. It is difficult to achieve quality results in substandard environments. Adequate laboratory and office space should be available to accommodate the necessary equipment and personnel. Equipment should be reliable, stable and appropriate to the task for which it is intended, and procedures should be in place to prevent contamination of measurement equipment with radionuclides. A preventive maintenance programme should be instituted to minimize the chance that equipment will fail at a critical time, such as in an emergency. Activities that are not directly related to the performance of

dosimetry service operations should be separated to avoid unnecessary interference. The general safety of working conditions should also be considered.

9.11. Special consideration should be given to the background radiation level in the laboratory, in particular at locations where dosimeters are kept for significant amounts of time before dispatch or evaluation. This level should never be significantly above normal local background. The background levels should be assessed regularly (e.g. using control dosimeters); these levels may be used in the routine monitoring programme to determine net doses by subtracting the background contribution. Dosimetry service records should include the results of routine background measurements.

9.12. The workplace controls should be adequate to ensure that no equipment or dosimeters are subjected to conditions likely to affect their performance. Factors that should be controlled include temperature, humidity, light levels, dust and reactive chemical vapours.

9.13. A stable power supply is needed so that the voltage and AC frequency remain within the specifications of the equipment in use. Stray electric and magnetic fields should be minimized to avoid affecting equipment and dosimeters.

PERFORMANCE ASSESSMENT

9.14. The characteristics of equipment and dosimetric materials may change as a function of either time or usage. Readout instruments should be checked at least daily. The sensitivity of TLDs may change, so it is necessary to evaluate their sensitivity on a regular basis. For film dosimetry, reproducibility of the developing and readout process should be determined with each batch.

9.15. Monitoring services need access to adequate calibration facilities. Radiation sources, capable of producing the radiation fields needed to evaluate the performance of the dosimetry system, should be available. Secondary standard instruments should be available to measure the intensity of the radiation beams in terms of the necessary quantities. The measuring equipment and radioactive sources should be calibrated and traceable to the national standard or, if such a standard is not available, to a primary standard of another country.

9.16. A system should be established to provide a quality indicator of overall dosimetry service performance. One method is the establishment of a ‘dummy’ user or client. On a regular basis, dosimeters are exposed to known doses either in the laboratory or

by some external test facility, and the dosimeters are submitted for processing with fictitious client or employee numbers, so that they will be processed in the normal manner. The reported values of doses should then be compared with the conventional true values, and the results interpreted using the method described in Section 4. Valuable information on dosimetry system performance may also be obtained through the service participating in national or international dosimetry intercomparison programmes.

CONTRACTING FOR A MONITORING SERVICE

9.17. It may be necessary for many operators (registrants or licensees) to obtain dosimetry services for external radiation under contract from commercial suppliers. This is especially true for operators with small workforces, such as medical practitioners, dentists and small hospitals, who may have limited knowledge and/or experience in radiation protection and dosimetry. However, in contracting for commercial dosimetry services, operators should ensure that there is adequate communication and understanding between them and the suppliers to ensure an effective dosimetry programme. The following items should be considered:

(a) Regulatory requirements;
(b) Radiation type(s) to be measured and types of dosimeter (e.g. basic dosimeters, providing information on the dose recorded, or discriminating dosimeters, providing additional information on the radiation type and its energy);
(c) Quality records, references or certificates for equipment and services;
(d) Dosimeter issuing periods;
(e) Details of where to wear and how to handle dosimeters;
(f) Dosimetric method(s) used;
(g) System of identification of dosimeters and wearers;
(h) Dose record keeping, reporting of results, customer dose entries, accessibility and confidentiality;
(i) Interpretation of results (quantities, dose limits, natural background, net dose, lower and upper limit of detection of the dosimetry system, etc.);
(j) Issuing and returning procedures;
(k) Procedures for ordering, changing and cancelling subscriptions;
(l) Information needed from the operator;
(m) Costs;
(n) The amount of time needed to make an order (or cancellation) effective;
(o) Information on routine and/or special services provided, such as immediate reporting by telephone or telex in the event of unusually high doses, emergency processing and advice on technical, scientific and legal matters.

Appendix

MONITORING FOR SKIN CONTAMINATION AND ASSESSMENT OF SKIN DOSE

PRINCIPAL OBJECTIVES

A.1. The principal objectives for the monitoring and assessment of skin irradiation and contamination can be summarized as follows:

(a) To determine compliance with dose limits, and hence in particular to ensure the avoidance of deterministic effects;
(b) In the case of overexposures, to initiate and/or support any appropriate medical examinations and interventions.

GENERAL CONSIDERATIONS

Strongly penetrating radiation

A.2. For strongly penetrating radiation types, the limitation on effective dose provides sufficient protection for the skin against stochastic effects. In virtually all situations, therefore (except those involving hot particles — see para. A.5), no further consideration of skin monitoring is necessary.

Weakly penetrating radiation

A.3. For weakly penetrating radiation, an additional limit is needed for skin exposure in order to prevent deterministic effects. The ICRP has recommended an annual equivalent dose limit of 500 mSv averaged over 1 cm^2, regardless of the area exposed. The nominal depth of measurement is 0.07 mm (7 mg/cm^2). A principal contribution to irradiation of the skin in this context is that from skin contamination.

MONITORING OF SKIN CONTAMINATION

A.4. Skin contamination is never uniform and occurs preferentially on certain parts of the body, notably the hands. For routine control purposes, it is adequate to regard the contamination as being averaged over areas of about 100 cm^2. Routine monitoring for skin contamination should therefore be interpreted on the basis of the average

dose equivalent over an area of 100 cm^2. In most monitoring for skin contamination, the reading is compared with a derived limit — a level expressed in units of, say, Bq/cm^2 which is considered to be capable of causing exposure equal to the relevant dose limit, and is usually established taking account of all potential exposure pathways (not just skin irradiation) — and the contamination is reduced when practicable. No attempt is routinely made to assess equivalent doses if these secondary limits are not exceeded. Sometimes, however, the contamination persists or is initially very high, and some estimation of equivalent dose becomes necessary. In such cases the dose should be averaged over an area of 1 cm^2 which includes the contamination. These estimates are often extremely imprecise, especially if the radiation from the contaminant may be absorbed below the surface layer of the skin. Uncertainties of two orders of magnitude are not uncommon. Such estimates are therefore regarded as qualitative procedures and considered separately from conventional monitoring for external radiation. However, where an estimate of equivalent dose is made that exceeds one-tenth of the appropriate equivalent dose limit, it should be included in the individual's personal record. Some of the contamination may also be transferred into the body, causing internal exposure. Monitoring for any associated intake of radioactive material into the body is discussed in the related Safety Guide on internal dose assessment [4].

A.5. Situations may arise in which exposure to 'hot particles' is possible. This can lead to spatially non-uniform exposure from discrete radioactive sources with dimensions of up to 1 mm. While compliance with dose limits is a principal objective, the ICRP has noted [37] that acute ulceration is a particular endpoint to be prevented. This implies that the average dose delivered within a few hours over a skin area of 1 cm^2, measured at depths of 10–15 mg/cm^2, should be restricted to 1 Sv. Detection of hot particles within an ambient radiation field in a workplace can be difficult, because of the very localized nature of the radiation from the particle. Emphasis should be given to identifying and controlling those operations which could give rise to such particles.

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Annex I

SUMMARY OF RECOMMENDED RADIATION WEIGHTING FACTORS AND Q–L RELATIONSHIPS

I–1. Values of radiation weighting factors w_R for the calculation of equivalent dose, recommended by the ICRP [I–1] and adopted in the Basic Safety Standards [I–2], are listed in Table I–1. The calculation of dose equivalent for the operational quantities

TABLE I–1. RADIATION WEIGHTING FACTORS[a] [I–1, I–2]

Type and energy range[b]	Radiation weighting factor, w_R
Photons, all energies	1
Electrons and muons, all energies[c]	1
Neutrons[d], energy:	
< 10 keV	5
10 keV to 100 keV	10
> 100 keV to 2 MeV	20
> 2 MeV to 20 MeV	10
> 20 MeV	5
Protons, other than recoil protons, energy > 2 MeV	5
Alpha particles, fission fragments, heavy nuclei	20

[a] All values relate to the radiation incident on the body or, for internal sources, emitted from the source.

[b] The choice of values for other radiations is discussed in Annex A of Ref. [I–1].

[c] Excluding Auger electrons emitted from radionuclides bound to DNA, for which special microdosimetric considerations apply.

[d] To assist in providing consistency in calculations, a smooth fit to the w_R values for neutrons as a function of energy is given as a mathematical relationship:

$$w_R = 5 + 17e^{-[\ln(2\varepsilon)]^2/6}$$

where ε is the energy in MeV. See Annex A of Ref. [I–1].

TABLE I–2. SPECIFIED Q–L RELATIONSHIPS [I–1, I–2]

Unrestricted linear energy transfer, L in water (keV/μm)	$Q(L)$
< 10	1
10–100	$0.32L - 2.2$
> 100	$300/\sqrt{L}$

$H_p(d)$, $H^*(d)$ and $H'(d,\Omega)$ uses quality factors Q rather than radiation weighting factors. The relationship between Q and the linear energy transfer L recommended by the ICRP [I–1] and adopted in the Basic Safety Standards [I–2] is given in Table I–2.

REFERENCES TO ANNEX I

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Annex II

INSTRUMENTATION FOR INDIVIDUAL MONITORING

INTRODUCTION

II–1. In this annex some general information is given on methods and systems used for individual monitoring. Reference is made to the literature where more detailed information is given.

DOSIMETERS FOR PHOTON AND BETA RADIATION

Photographic film dosimeters

II–2. Photographic film dosimeters are used for determining personal exposure to photon, beta and thermal neutron radiations. They commonly consist of a photographic film which is placed inside a suitable holder containing appropriate filters. Such assemblies are often referred to as film badges [II–1].

II–3. The emulsion of the film is made of silver bromide crystals which are suspended in a gelatinous medium. A thin layer of this emulsion is coated uniformly onto a thin plastic base. The action of ionizing radiation on the grains in the emulsion produces a latent image. In subsequent development, the silver ions in the latent image produce permanent blackening. The optical density is measured with a densitometer, and is a function of the film type and developing process as well as the type and energy of the radiation being measured. The optical density does not vary linearly with dose. Photographic films are used most widely for photon and beta monitoring, but they will respond to the ionizing effects of any radiation which imparts enough energy to produce silver ions in the emulsion. Film is often used for indirect measurements of thermal neutrons, through the capturing of the neutrons with a Cd filter and use of the blackening of the film produced by the resulting gamma radiation as an indication of the neutron dose.

II–4. A complicating factor which is of much concern in practical photon dosimetry is the energy dependence of the film relative to human tissue. Compensation for the energy dependence of the film dosimeter is achieved by using one or more filters of appropriate material and thickness. Although the use of one filter is adequate for photons of energy higher than about 0.1 MeV, the use of a multiple filter system (e.g.

copper, tin, lead and plastic filters and open windows) is necessary for lower energy photons. The type of incident radiation and the dose can be estimated from the responses behind different filters.

II–5. Type testing is necessary whenever a new type of film is proposed for use or changes are made to the developing process. Film badges are generally used for issue periods up to one month and are suitable for use in controlled areas. When a longer issue period is used, special attention should be paid to the problem of fading. It is necessary to calibrate film dosimeters by irradiating identical films with known doses and processing these 'standards' simultaneously with the dosimeters.

II–6. Photographic film dosimeters can be used as discriminating dosimeters, giving qualitative information in addition to dose. This technique can be very economical, depending upon the degree of automation adopted. Film dosimeters are susceptible to temperature and humidity, resulting in fading of the latent image. The energy dependence of the response may necessitate a complex filter system. This type of dosimeter can readily be designed to measure $H_p(10)$ and $H_p(0.07)$ for photons and beta radiation with energies (ε_{max}) in excess of 0.5 MeV.

Thermoluminescent dosimeters

II–7. Thermoluminescence is the emission of light when a material that has been exposed to ionizing radiation is heated. This light results from the release of electrons that were excited and trapped when the material was irradiated, and the amount of light released is directly related to the radiation dose received by the material. The random release of trapped electrons before readout is called fading, and may result from thermally or optically stimulated release of the electrons. In thermoluminescent dosimetry (TLD), the relationship between the relevant signal and the dose equivalent to be measured must be determined by calibration.

II–8. In using this phenomenon for dosimetry, the thermoluminescent material is observed by a photomultiplier or other light sensitive device during the heating process. A plot of the luminescent light output against temperature is called the 'glow curve'. The shape of the glow curve depends on the type and amount of impurities and lattice defects present in the material, as well as on the thermal history and treatment of the material. The photomultiplier tube has high sensitivity, a high signal to noise ratio and a large dynamic range. The area under the glow curve is used as a measure of dose. The thermoluminescent material is discharged by the reading process and is then ready to register a new exposure (although some materials must be annealed before reissue).

II–9. The mechanism of thermoluminescence is complex, and although general theoretical models have been postulated, each thermoluminescent phosphor is unique, and models which correspond to particular materials display very different characteristics.

II–10. TLD has found increasing application with the progress made in the development of solid thermoluminescent dosimeters and instrumentation for reading them. TLD is now commercially available, and is widely used in routine personal dosimetry, environmental monitoring and clinical radiation dosimetry.

II–11. TLD is increasingly accepted for radiation protection dosimetry for the following reasons:

(a) The existence of nearly tissue equivalent thermoluminescent materials;
(b) Sufficiently high sensitivity and accuracy for both personal and environmental monitoring;
(c) Commercial availability as small sized solid detectors adaptable for both manual and automatic processing;
(d) Suitability for beta skin and extremity dosimetry;
(e) Availability of materials with excellent long term stability under varying environmental conditions;
(f) Ease of processing;
(g) Reusability;
(h) Linearity of response with dose and dose rate over a large range.

Some general characteristics of the most common thermoluminescent materials used for radiation protection dosimetry are listed in Table II–1.

II–12. The dosimeters currently used for personal monitoring of beta doses suffer from an energy threshold problem, because the detector and its cover are too thick. Thin and ultrathin detectors are commercially available but may be difficult to use on a large scale for routine monitoring. In recent years, several types of thermoluminescent detectors have been developed to measure the operational quantities $H_p(0.07)$ and $H_p(10)$ [II–2 to II–4].

II–13. The response of thermoluminescent materials to neutrons depends on the detector composition, on TLD encapsulation and, strongly, on neutron energy. A number of phosphors have high sensitivity to thermal neutrons but little response to fast neutrons. Various techniques have been investigated to increase the fast neutron response of TLDs, such as using the body as a moderator to thermalize the neutrons. This has achieved practical importance in personal albedo dosimeters (see below).

TABLE II–1. GENERAL CHARACTERISTICS OF SOME COMMERCIALLY AVAILABLE THERMOLUMINESCENT DOSIMETERS

TLD type	Effective atomic number Z_{eff}	Main peak (°C)	Emission maximum (nm)	Relative sensitivity	Fading (at 25°C)
LiF:Ti,Mg	8.3	200	400	1	5%/year[a]
LiF:Na,Mg	8.3	200	400	1	5%/year[a]
LiF:Mg,Cu,P	8.3	210	400	25	5%/year
$Li_2B_4O_7$:Mn	7.3	220	605	0.20[b]	4%/month
$Li_2B_4O_7$:Cu	7.3	205	368	2[b]	10%/2 months[a]
MgB_4O_7:Dy	8.4	190	490	10[b]	4%/month[a]
BeO	7.1	190	200–400	0.20[b]	8%/2 months
$CaSO_4$:Dy	14.5	220	480–570	30[b]	1%/2 months
$CaSO_4$:Tm	14.5	220	452	30[b]	1–2%/2 months
CaF_2:Mn	16.3	260	500	5[b]	16%/2 weeks
CaF_2 (natural)	16.3	260	380	23	very slight
CaF_2:Dy	16.3	215	480–570	15[b]	8%/2 months[a]
Al_2O_3	10.2	360	699	4[b]	5%/2 weeks[a]

[a] Fading in the dark (after using a post-irradiation annealing of 15 min at 100°C) related to 1 day storage.

[b] Light sensitive.

Photoluminescent dosimeters

II–14. Photoluminescence is based on the formation of induced luminescent centres in silver-doped phosphate glasses when they are exposed to ionizing radiation. When the glasses are subsequently exposed to ultraviolet radiation, visible light is emitted with an intensity that is linearly related to the absorbed dose from the ionizing radiation. Unlike thermoluminescence, the effects of the ionizing radiation — the centres — are not destroyed by the normal reading process, and are extremely stable, so that fading at room temperature is negligible over a period of several years and the dose information can be obtained at any time during long term dose accumulation [II–1].

II–15. Phosphate glasses can be produced on a large scale with good reproducibility and constant sensitivity. Thus, calibration of individual detectors is not needed. The application of commercially available pulsed ultraviolet laser readers reduces the ‘pre-dose’ — the apparent reading from unirradiated glasses — to a value of about 10 μSv [II–5]. This eliminates some of the drawbacks of the older, conventional

readout technique, which needed glass cleaning and subtraction of the pre-dose in order to measure doses below 100 μSv.

II–16. Because of the high atomic number of some glass materials, energy compensating filters have to be used. Later glass dosimeters give an energy dependence within ±15% for photon energies above 15 keV [II–6]. A complete phosphate glass dosimetry system with an automatic readout using ultraviolet laser excitation can be used in large scale systems for personal monitoring.

II–17. Phosphate glass dosimeters have been routinely applied in personal and environmental monitoring for measuring $H_p(10)$ and $H_p(0.07)$ at levels of dose ranging from environmental levels to those of interest for accident situations.

II–18. The advantages of photoluminescent dosimeters include permanent and long term integration of dose information, good accuracy, negligible fading and the possibility of repeating a dosimeter reading if necessary.

Electronic dosimeters

II–19. Electronic dosimeters have been developed for personal dosimetry based on Geiger–Müller devices that detect photons above 30 keV, and on silicon diode detectors.

II–20. An electronic dosimetry system has recently become commercially available that is based on the use of three silicon diode detectors, suitable for the simultaneous measurement of $H_p(10)$ and $H_p(0.07)$ for photons and beta radiation (above a mean energy of 250 keV). This device is suitable for use by workers in controlled areas, provided that the dose contribution from low energy beta radiation is not significant. Dosimetry services in a number of countries have successfully applied to their regulatory authorities for approval to use it as an official or legal dosimeter [II–7].

II–21. Details of a credit card sized routine individual dosimeter, involving a silicon diode detector, have recently been published [II–8]. This dosimeter measures dose equivalent and dose equivalent rate from photons, has an adjustable alarm, and stores the daily integrated dose for the last 12 months in its memory.

II–22. Electronic devices can give the worker an instant indication of both accumulated dose and dose rate. Preset visual and audible alarms are also provided, so that these devices can be used simultaneously as an integrating dosimeter and as an alarm dosimeter.

Pocket dosimeters

II–23. Quartz fibre dosimeters are still in use for individual monitoring, although their use has declined. They comprise a small ion chamber with a fibre, and the deflection of the fibre is proportional to the dose received. Readout is made optically, by looking through the dosimeter and noting the fibre deflection on a scale. These devices are simple and of low cost. However, their sensitivity in relation to the levels needed for current radiation protection purposes is poor. Moreover, they have a limited useful dose range (approximately a factor of 20).

II–24. Suitable direct reading pocket dosimeters can be chosen, depending on the expected maximum dose and the radiation qualities to be encountered. The main operational problems are the influence of zeroing and the charge leakage, both of which limit the minimum measurable dose.

NEUTRON DOSIMETERS

Nuclear track emulsions

II–25. Nuclear track emulsions are suitable for fast neutron dosimetry. The neutrons interact with hydrogen nuclei in the emulsion and surrounding materials, producing recoil protons by elastic collisions. The ionizing particles pass through the emulsion to create a latent image, which leads to darkening of the film along the particle track after processing [II–9].

II–26. Nuclear track emulsions typically have an energy threshold of about 0.7 MeV, and have a poor energy response and a limited dose range. This type of dosimeter saturates at about 50 mSv.

II–27. Neutrons with energies below 10 eV can be detected through interaction with the nitrogen nuclei of the gelatine resulting in the production of recoil protons. If sensitivity to thermal neutrons is undesirable, the dosimeter should be kept under a filter of material such as cadmium that absorbs thermal neutrons.

II–28. A microscope with a magnification of 1000× may be used for counting recoil tracks in the emulsion. Counting can be facilitated by using a microscope fitted with a television camera and monitor. The accuracy of the measured dose depends on the skill of the operator in recognizing the tracks in the emulsion.

II–29. One disadvantage of nuclear track emulsion is its high rate of fading. The fading is accelerated by high humidity and temperature, and can be as much as 75% per week. The problem can be controlled if the films are dried in a controlled atmosphere and sealed in a moisture-proof pouch prior to use.

II–30. Another serious problem with emulsions is that photon radiation can darken the film following exposure and development, making it very difficult to distinguish the proton tracks. Because of these disadvantages, including the high neutron energy threshold, nuclear track emulsions are increasingly being replaced in personal dosimetry by other methods such as TLD albedo dosimeters and/or solid state track detectors.

Solid state nuclear track detectors

II–31. Strongly ionizing particles such as fission fragments, alpha particles or neutron induced recoil particles produce structural damage along their path in many materials such as minerals, glass and different plastics [II–10]. By etching the surface of the detector with suitable reagents, the damage zone along the particle track can be removed and the etch pits enlarged to become visible under an optical microscope. The application of electrochemical etching greatly enlarges the track size, and track densities can easily be counted in a single detector field of 1 cm^2 using a low magnification (e.g. 20×) microscope or other optical reader.

II–32. The size and shape of the etched track depend on the type, energy and angle of incidence of the particle, the type of detector material, and the etching conditions (i.e. the etchant concentration and temperature and the etching time). These parameters should be optimized for each material and particular application.

II–33. For neutron dosimetry, three detector types have been commonly used, namely fission track, recoil track and (n,α) track detectors. These are described briefly below; a comprehensive discussion of track detection measurement techniques can be found in Refs [II–11, II–12].

Fission track detectors

II–34. A radiator or converter of fissionable material emits fission fragments following exposure to neutrons. The fission fragments are detected with a solid state track detector such as polycarbonate. Fission reactions have either an energy threshold (e.g. 0.6 MeV for ^{237}Np, 1.3 MeV for ^{232}Th, 1.5 MeV for ^{238}U) or an extremely high cross-section for thermal neutrons (e.g. ^{235}U). The use of fissionable materials

in dosimeters is now restricted or prohibited in certain countries because of their radioactivity.

Recoil track detectors

II–35. The elastic scattering of neutrons with the nuclei of the plastic detectors may produce charged recoil particles such as protons or atoms of carbon, oxygen and nitrogen. These recoils produce latent tracks which can be made visible by etching. Chemical or electrochemical etching is used to enlarge the tracks. The track density, which is proportional to the neutron exposure, can be counted with a microfiche reader or an automatic particle counter [II–11, II–12]. Because of the LET of recoil protons and the short range of the heavier particles, different types of plastic have different sensitivities to neutrons, and the response also depends on the neutron energy. For each detector material or combination of radiator, absorber and detector material, the etching technique should be optimized, and the energy response curves should be established by experiment. The most common detector materials are polycarbonate, cellulose nitrate and CR-39. A number of dosimetry services based on the use of CR-39 are now in operation with the approval of the regulatory authorities.

Track detectors based on (n,α) reactions

II–36. Neutrons interact with ^{6}Li or ^{10}B in an external radiator. The alpha particles produced by (n,α) reactions have maximum alpha energies of about 2.5 MeV (^{6}Li) and 1.5 MeV (^{10}B) for neutrons below several hundred keV. The reaction cross-sections are high for thermal neutrons and decrease as the neutron energy increases in inverse proportion to the neutron velocity. Most commercially available plastic detectors can detect the emitted alpha particles. The detection efficiency depends on the type of material and the etching conditions.

TLD albedo dosimeters

II–37. Albedo dosimetry is based on the detection of low energy neutrons (albedo neutrons) which emerge from the body of a person exposed to neutrons of various energies. Any thermal neutron detector placed on the surface of the body may therefore serve as an albedo detector.

II–38. Albedo dosimeters usually use thermoluminescent detectors such as ^{6}LiF in boron-loaded plastic encapsulations which separate the albedo neutrons from incident thermal neutrons. Because of the photon sensitivity of TLDs, the neutron dose reading is given by the difference between ^{6}LiF and ^{7}LiF detector readings.

II–39. Albedo dosimeters have been designed with a high and nearly constant response for neutrons in the energy range from thermal to 10 keV. However, the response decreases rapidly above 10 keV. In stray neutron fields, the relative energy response of an albedo detector has been found to vary by a factor of as much as 20.

II–40. A two-component albedo dosimeter designed for automatic read-out in various TLD systems has been found to be suitable for routine monitoring [II–13]. This dosimeter type incorporates the albedo detector and an additional thermal neutron detector.

II–41. The neutron response depends on the neutron spectrum. Neutron spectra vary widely in workplaces. However, site specific correction factors can be used to correct for this, provided that the neutron spectrum is known and remains constant.

II–42. The energy dependence of albedo detectors can be compensated for in dosimeters used in fast neutron fields by the addition of a nuclear track detector, such as polycarbonate, for separate measurement of fast neutrons. In such a detector combination, the albedo detector serves as the basic neutron detector that can be read automatically using a normal TLD reader. The track detector then only needs to be processed if a significant exposure is indicated by the TLD.

Bubble detectors

II–43. Bubble detectors are a new type of direct reading neutron dosimeter [II–14]. The detector is prepared by suspending superheated droplets in a firm elastic polymer; the passage of neutrons through the material causes visible vapour bubbles, which are trapped at the sites of formation [II–14]. The number of bubbles gives a measure of the neutron dose. This detector is a completely passive device which can be stored until needed for use. It does not require any electronic apparatus for measurement or reading. However, an automatic reader which is computer controlled can be used to perform the reading if a large number of detectors is used routinely.

II–44. The detector is extremely sensitive to neutrons, detecting down to the millisievert range, and is completely insensitive to gamma radiation. The detectors can be made to have different neutron energy thresholds, from 100 keV to several MeV, so that a set of bubble detectors with different thresholds can be used for crude neutron spectrometry. However, these detectors suffer from their significant dependence on ambient temperature, and both their energy and dose ranges are limited, so that a number of dosimeters with different sensitivities may be necessary to cover the required dose range.

Personal alarm neutron dosimeters

II–45. Personal alarm neutron dosimeters can provide an indication of the neutron dose equivalent to the wearer. These detectors are based on several techniques, including:

(a) A counter for measuring recoil protons;
(b) A ^{3}He detector in a small polyethylene moderator with a shield against thermal neutrons;
(c) The Rossi counter principle, with a microprocessor to convert counts into either absorbed dose or dose equivalent [II–15];
(d) A silicon surface barrier detector to detect recoil ions from polyethylene and ^{10}B radiators [II–16].

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Annex III

INSTRUMENTATION FOR WORKPLACE MONITORING

INTRODUCTION

III–1. Workplace monitors are primarily intended to provide information on the dose rates within the working area to permit decisions to be made on its occupancy. It is necessary to know the dose equivalent rates in the various working areas to assess and control occupational exposure. This is true while the workers occupy a particular area or before they are admitted to it. Usually the dose rate is monitored, although this might not be necessary where dose rates do not vary significantly with time.

III–2. Fixed area monitors are often equipped with remote displays and audible alarms. Apart from some engineering differences, their detectors and operating methods are similar to those of portable survey meters. From the practical viewpoint, instruments for area monitoring can be divided into the following categories:

(a) Instruments for photons;
(b) Instruments for beta particles and low energy photons;
(c) Instruments for neutrons;
(d) Passive gamma monitors;
(e) Passive neutron survey meters;
(f) Spectrometry systems.

A comprehensive discussion of monitoring methods can be found in Refs [III–1, III–2].

INSTRUMENTS FOR PHOTONS (GAMMA AND X RAYS)

Ionization (ion) chambers

III–3. Hand held survey meters and some installed instruments use chambers that have walls of low atomic number material and that are filled with air in equilibrium with the atmosphere. In the past, such units were designed to measure exposure, but most designs are now intended to measure ambient dose equivalent $H^*(10)$, and often directional dose equivalent $H'(0.07)$.

III–4. These instruments are based mainly on measurement of the quantity exposure. Instrument designs have been modified by the addition of aluminium within the

chamber to enhance the response below 150 keV, and of aluminium on the slide or cover to provide an appropriate reduction in response below about 40 keV [III–3].

III–5. Hand held instruments for use at normal occupational dose levels (i.e. a few μSv/h) generally have chamber volumes in the range 300–700 cm^3. Installed instruments designed for use where beta and low energy photons are not expected to be a problem often have large (of the order of 5 L) steel walled chambers filled with argon at high pressure. These have a large useful dynamic range, from about 0.1 μSv/h to as much as 1 Sv/h.

Geiger–Müller (GM) counters

III–6. GM counters are popular for use in X ray and gamma fields. They produce large pulses which can be counted and processed easily. Their dynamic range is, however, limited by dead time losses at high count rates. Care should also be taken to ensure at overload rates that the dose rate indication does not fall back on scale; this is a fundamental test that should be performed during type testing.

III–7. GM counters have a photon detection efficiency, typically about 0.5%, that is effectively constant over a wide energy range. This means that the ambient dose equivalent response is energy dependent. Effective filters can be designed which allow good energy and angular performance for $H^*(10)$ above about 50 keV for steel walled detectors and from 15 keV for end window detectors.

III–8. It should be noted that the use of GM counters in pulsed radiation fields may lead to serious underestimates of the measured radiation quantity. For this reason extreme caution is needed when GM counters — or, indeed, any pulse counting detectors — are used in such situations.

Scintillation instruments

III–9. Organic scintillators, when used to measure exposure rate or air kerma rate, are sufficiently close to air in effective atomic number to require little correction for energy dependence, except at energies below about 0.1 MeV. In anthracene, for example, the response per unit kerma falls, primarily because only the outer layers of the crystal are irradiated. Incorporation of a small amount of material with high atomic number in front of the crystal can partially offset this drop, and commercially available survey meters allow the measurement of photons above 20 keV.

III–10. Scintillation instruments [III–4] may be used for all types of X ray and gamma survey. In relatively weak radiation fields, although the electronic parts of the

instruments cause their overall size to be similar to that of ion chambers, the detecting volume can be much smaller. Although a 1 cm^3 crystal is often adequate, the higher sensitivity of larger crystals permits their use for measurements of natural background dose rates.

III–11. NaI(Tl) crystals, widely used in gamma spectroscopy, make very sensitive detectors. However, their response is very energy dependent. For this reason, simple units cannot be used for making accurate measurements of dosimetric quantities. However, instruments using spectrometric techniques can be used and are very sensitive.

Proportional counters

III–12. The sensitivity of proportional counters is higher than that of ion chambers, because of gas multiplication. Proportional counters can be used either as pulse detectors or as continuous current detectors, allowing the measurement of photon dose rates from 1 mSv/h to 10 Sv/h. The main advantages of commercial proportional counters are their high sensitivity, large dose rate range and low energy dependence. However, they require a stable high voltage supply and are much more expensive than ion chamber or GM based instruments.

Semiconductors

III–13. Dose rates can be measured with silicon diodes used as pulse generators (at lower dose rates) or as photocurrent generators (at high rates). Silicon has a higher atomic number than tissue and hence it is necessary, in both pulse and current modes, to provide an energy compensation filter appropriate to the quantity of interest. These filters inevitably limit the low energy threshold.

INSTRUMENTS FOR BETA RADIATION AND LOW ENERGY PHOTON RADIATION

Ionization chambers

III–14. It is important to be able to measure the dose equivalent rates from beta radiation (or low energy X rays) and from photons. Measurement can be made with a single detector. In this case, the detector (ion chamber) is fitted with a window which can be opened or shut. When it is shut, the strongly penetrating component (i.e. photons with energies above approximately 20 keV) can be measured. With the window open, both components are measured and the weakly penetrating component

(beta particles and low energy photons) of the dose equivalent is estimated by subtraction.

III–15. Most survey measurements for beta radiation (and low energy photons) are made with small, portable ion chambers which can also be used for X ray and gamma surveys. One side of the chamber comprises a thin conducting plastic sheet that is covered, when measuring photons, with a piece of material equivalent to 1 cm of tissue. The thick cover is removed for measuring beta radiation [III–5]. Another type of beta survey meter has an entire thin wall. Such a chamber may not be appropriate for the measurement of the directional dose equivalent.

III–16. The walls of an ion chamber to be used for beta radiation measurement should be made of materials similar in composition to tissue. However, the exact composition is not as important for electrons as in the case of ion chambers for X rays or gamma radiation. With electrons, the function of the walls is merely to simulate the absorption and backscattering by the body. The foregoing remarks about size, sensitivity, response time and readout methods for gamma ion chambers also apply to beta radiation measurements [III–5].

GM counters

III–17. Thin walled or thin windowed GM counter survey instruments for photons are sometimes also used for the detection of beta radiation. If the counter is provided with a cover that is sufficiently thick to stop the beta radiation, the difference between readings with and without the cover can be used to distinguish between beta and gamma radiation. Thin end window GM detectors in particular have an acceptable energy dependence for workplace beta dose rate monitoring, and have the additional advantage of small size for a particular minimum useful dose rate.

Scintillators

III–18. A good beta dose rate monitor for $H'(0.07)$ can be made using a thin (3–4 mg/cm^2) scintillator, covered by a light-tight plastic window of similar thickness. It can be used in the pulse counting mode at low dose rates, when it behaves similarly to a GM detector, or in the current mode at high dose rates. These are not for routine use, but for special applications.

Semiconductor detectors

III–19. Semiconductor detectors operating in the mean current mode can be used for the measurement of high dose rates. Their thin detection layer makes them suitable

for beta dosimetry. For beta and low energy photon radiation measurements, thin sensitive layer silicon diodes are suitable for $H'(0.07)$ evaluation, but their response to gamma radiation is higher than their response to beta radiation because the effective atomic number of the detector is too high. Such detectors are not normally used for operational radiation protection.

INSTRUMENTS FOR NEUTRONS

Moderator based survey instruments

III–20. Moderator based survey instruments are the most common instruments for the monitoring of neutron fields [III–6, III–7]. They consist of a hydrogenous moderator which moderates the neutrons and detects the thermalized neutrons using detectors such as proportional counters filled with BF_3 or ^{3}He gas or ^{6}LiI scintillators. The neutrons are detected by the $^{10}B(n,\alpha)^7Li$, $^3He(n,p)^3H$ or $^6Li(n,\alpha)^3He$ reactions, which have such high Q values that good discrimination against gamma radiation can be achieved. By choosing an appropriate thickness for a moderating shield, or by varying the wall thickness and the gas mixture and pressure, the response to neutrons can be adjusted to give an output which is roughly proportional to the dose equivalent or to the dose. Crude neutron spectrometry can be achieved by mathematically analysing the responses of a set of moderated spheres with different diameters [III–8]. The responses for several moderated neutron instruments to operational neutron fields have been calculated [III–9].

III–21. By thermalizing the neutrons in a hydrogenous moderator, Andersson and Braun [III-10] produced an instrument with an approximately energy independent dose equivalent response for neutrons up to 10 MeV. The instrument used a BF_3 proportional counter surrounded by a perforated cadmium shield in a cylindrical moderator and suffered from some anisotropy in response (a factor of two or more). This anisotropy has been largely overcome by the use of a spherical moderator of polyethylene of diameter 20–30 cm, but at the expense of the energy response. Detectors such as ^{6}LiI scintillators and ^{3}He proportional counters have been used as alternatives to the proportional counters. The main characteristic of all these instruments is an over-response to intermediate energy neutrons.

III–22. Another instrument [III–11] uses two moderating spheres (107 and 64 mm in diameter) in a single case to produce an instrument weighing 3 kg that covers the dose equivalent range from 20 to 200 mSv/h, with an energy response of ±30% over the energy range from thermal to 10 MeV. The response of the larger sphere is corrected

using the ratio of the count rates in the two spheres, which varies from 0.15 to 0.8 for observed neutron spectra. The correction — which varies from 1 to 30 over this range — is automatically made in the instrument.

Ionization chambers

III–23. Ionization chambers were first developed to measure exposure to X rays and gamma radiation. However, if hydrogen is introduced into the walls and the gas, they can be made more sensitive to neutrons. However, they are also sensitive to photons, and so it is necessary to provide a second chamber which is relatively insensitive to neutrons (e.g. with graphite walls and a CO_2 gas mixture, or aluminium walls and argon gas) to correct for the gamma radiation which is always associated with neutrons. Tissue equivalent ionization chambers measure the neutron absorbed dose, not the dose equivalent. Because their response to gamma radiation per unit dose is similar to that for neutrons, it is not possible to discriminate efficiently between the two radiation types and so ionization chambers are not particularly useful for neutron monitoring, except where pulsed fields may be a problem. Small tissue equivalent ion chambers may be used in personal alarm dosimeters.

Other neutron instruments

III–24. A number of other neutron detection methods can be used for special applications, but are not generally applicable for routine radiation protection.

Recoil proton proportional counters

III–25. Recoil proton proportional counters are usually lined with polyethylene and filled with either ethylene (C_2H_4) or cyclopropane (C_3H_6) at pressures of the order of 100 kPa. The wall thickness is chosen on the basis of energy and range relationship calculations, so that the system satisfies the requirements of the Bragg–Gray principle. The recoil proton spectra can be analysed mathematically to infer the incident neutron spectrum. This spectral information can then be used to determine the ambient dose equivalent. The practical energy range for these systems is about 10 keV–1.5 MeV.

Rossi proportional counters

III–26. Tissue equivalent proportional counters can be used to measure the LET of the deposited energy, in addition to dose. The LET can then be used with the Q–L relationship defined by the ICRP (see Table I–2) to determine the mean quality factor Q, which can then be incorporated into the electronics of the instrument. Thus, dose

can be converted to dose equivalent. These instruments can also be used for measurements in mixed radiation fields.

Scintillators

III–27. Organic scintillation detectors offer a potentially simple method of neutron dosimetry and spectrometry because they can be made of tissue equivalent materials and are small in volume. There are, however, two major drawbacks. Firstly, the scintillation efficiency for light production is low, with 1–2 keV typically being required to produce a photoelectron at the first stage of a multiplier phototube. Secondly, they are very sensitive to gamma radiation; they require about three times as much energy to produce a photoelectron from a recoil proton as from a gamma photon, and ten times as much for an alpha particle. However, it is possible to use pulse shape discrimination to separate charged particle events from those produced by electrons. There is also a non-linear relationship between the energy of the recoil proton and the magnitude of the light pulse, but this can be corrected for in a neutron spectrometer during the mathematical analysis. These limitations restrict the energy range of the detector to about 0.2–20 MeV.

Semiconductor detectors

III–28. Semiconductor detectors are normally based on silicon and germanium, and are not used directly for neutron measurements. However, they can be used in neutron spectrometers to measure secondary particles such as protons, tritons and alpha particles produced in converter foils of lithium borate, boron, ^{6}LiF, polyethylene and polycarbonate. They are small and sensitive — for example, the ionization yield is about ten times larger than in ionization chambers — and their density is about 1000 times that of the gas in a chamber.

Passive neutron area monitoring

III–29. In the measurement of neutron fields where the gamma dose rate is extremely high, or when the field occurs in intense pulsed mode (such as around an accelerator), active detectors are unsuitable because of electronic saturation. In such applications, passive devices such as track etch detectors, activation foils or TLDs are often used. These detectors are normally used as thermal neutron detectors at the centre of moderators. Track etch detectors and activation foils (e.g. gold or indium) provide excellent gamma discrimination along with high neutron sensitivity.

III–30. A very attractive technique uses polycarbonate foils in contact with boron so that an (n,α) reaction produces tracks, which can be developed by electrochemical

etching. The limit of sensitivity is about 1 mSv and so the technique can be applied to the measurement of background radiation.

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Annex IV

REFERENCE CONDITIONS AND STANDARD TEST CONDITIONS

IV–1. The International Electrotechnical Commission (IEC) specifies reference conditions and standard test conditions for the type testing of dosimetry systems [IV–1]. These are summarized in Table IV–I.

TABLE IV–1. RADIOLOGICAL PARAMETERS

Influence quantity	Reference conditions	Standard test conditions (unless otherwise indicated)
Photon radiation	^{137}Cs[a]	^{137}Cs[a]
Neutron radiation	$^{241}Am/Be$[a]	$^{241}Am/Be$[a]
Beta radiation	$^{90}Sr/^{90}Y$[a]	$^{90}Sr/^{90}Y$[a]
Phantom (only in the case of personal dosimeters)	30 cm × 30 cm × 15 cm slab of ICRU tissue (for whole body dosimeters)	ISO water slab phantom
	Right circular cylinder of ICRU tissue with 73 mm diameter and 300 mm length (for wrist or ankle dosimeters)	ISO water pillar phantom
	Right circular cylinder of ICRU tissue with 19 mm diameter and 300 mm length (for finger dosimeters)	ISO PMMA rod phantom
Angle of radiation incidence	Calibration direction given by manufacturer	Direction given ±5°
Orientation of assembly	To be stated by manufacturer	Stated orientation ±5°
Assembly controls	Set up for normal operation	Set up for normal operation
Contamination by radioactive elements	Negligible	Negligible
Radiation background	Ambient dose equivalent rate $H^*(10)$ 0.1 μSv/h or less if practical	Ambient dose equivalent rate $H^*(10)$ less than 0.25 μSv/h
Ambient temperature	20°C	18–22°C[b,c]
Relative humidity	65%	50–75%[b,c]
Atmospheric pressure	101.3 kPa	86–106 kPa[b,c]

TABLE IV–1. (cont.)

Influence quantity	Reference conditions	Standard test conditions (unless otherwise indicated)
Stabilization time	15 min	>15 min
Power supply voltage	Nominal power supply voltage	Nominal power supply voltage ±3%
Frequency[d]	Nominal frequency	Nominal frequency ±1%
AC power supply	Sinusoidal	Sinusoidal with total wave form harmonic distortion less than 5%[d]
Electromagnetic field of external origin	Negligible	Less than the lowest value that causes interference
Magnetic induction of external origin	Negligible	Less than twice the value of the induction due to the earth's magnetic field
Assembly controls	Set up for normal operation	Set up for normal operation

[a] Another radiation quality may be used if this is more appropriate.
[b] The actual values of these quantities at the time of test shall be stated.
[c] The values in the table are intended for calibrations performed in temperate climates. In other climates, the actual values of the quantities at the time of calibration shall be stated. Similarly, a lower limit of pressure of 70 cpu may be permitted where instruments are to be used at higher altitudes.
[d] Only for assemblies which are operated from the mains electricity supply.

REFERENCE TO ANNEX IV

[IV–1] INTERNATIONAL ELECTROTECHNICAL COMMISSION, Draft Standard: Direct Reading Personal Dose Equivalent and/or Dose Equivalent Rate for X, Gamma and High Energy Beta Radiation, Rep. IEC/SC 45B (CO)94, IEC, Geneva (1989).

Annex V

DATA RELEVANT TO TYPE TESTING OF PERSONAL DOSIMETERS AND AREA MONITORS IN TERMS OF THE OPERATIONAL QUANTITIES

V–1. A range of reference data is needed in the type testing of dosimetry systems to relate the operational dosimetric quantities to physical quantities such as kerma and particle fluence, to correct measurements of the operational quantities according to the angle of incidence of radiation and to specify the characteristics of the reference radiations recommended by the ISO [V–1 to V–3]. A selection of the data referred to in the main text is reproduced in Tables V–1 to V–8 and Figs V–1 and V–2 of this Annex for ease of reference.

TABLE V–1. CONVERSION COEFFICIENTS FROM AIR KERMA TO $H_p(10,0°)$ IN AN ICRU SLAB AND ANGULAR DEPENDENCE FACTORS (PHOTONS) [V–4]

Photon energy (MeV)	$H_p(10,0°)/K_a$ (Sv/Gy)	Ratio $H_p(10,\alpha)/H_p(10,0°)$ for angles α					
		0°	15°	30°	45°	60°	75°
0.010	0.009	1.000	0.889	0.556	0.222	0.000	0.000
0.0125	0.098	1.000	0.929	0.704	0.388	0.102	0.000
0.015	0.264	1.000	0.966	0.822	0.576	0.261	0.030
0.0175	0.445	1.000	0.971	0.879	0.701	0.416	0.092
0.020	0.611	1.000	0.982	0.913	0.763	0.520	0.167
0.025	0.883	1.000	0.980	0.937	0.832	0.650	0.319
0.030	1.112	1.000	0.984	0.950	0.868	0.716	0.411
0.040	1.490	1.000	0.986	0.959	0.894	0.760	0.494
0.050	1.766	1.000	0.988	0.963	0.891	0.779	0.526
0.060	1.892	1.000	0.988	0.969	0.911	0.793	0.561
0.080	1.903	1.000	0.997	0.970	0.919	0.809	0.594
0.100	1.811	1.000	0.992	0.972	0.927	0.834	0.612
0.125	1.696	1.000	0.998	0.980	0.938	0.857	0.647
0.150	1.607	1.000	0.997	0.984	0.947	0.871	0.677
0.200	1.492	1.000	0.997	0.991	0.959	0.900	0.724
0.300	1.369	1.000	1.000	0.996	0.984	0.931	0.771
0.400	1.300	1.000	1.004	1.001	0.993	0.955	0.814

TABLE V–1. (cont.)

Photon energy	H_p (10,0°)/K_a	Ratio $H_p(10,\alpha)/H_p(10,0°)$ for angles α					
(MeV)	(Sv/Gy)	0°	15°	30°	45°	60°	75°
0.500	1.256	1.000	1.005	1.002	1.001	0.968	0.846
0.600	1.226	1.000	1.005	1.004	1.003	0.975	0.868
0.800	1.190	1.000	1.001	1.003	1.007	1.987	0.892
1.0	1.167	1.000	1.000	0.996	1.009	0.990	0.910
1.5	1.139	1.000	1.002	1.003	1.006	0.997	0.934
3.0	1.117	1.000	1.005	1.010	0.998	0.998	0.958
6.0	1.109	1.000	1.003	1.003	0.992	0.997	0.995
10.0	1.111	1.000	0.998	0.995	0.989	0.992	0.966

TABLE V–2. CONVERSION COEFFICIENTS FROM AIR KERMA TO $H_P(0.07,0°)$ IN AN ICRU SLAB AND ANGULAR DEPENDENCE FACTORS (PHOTONS) [V–4]

Photon energy	H_P (10,0°)/K_a	Ratio $H_p(10, \alpha)/H_p(10,0°)$ for angles α					
(MeV)	(Sv/Gy)	0°	15°	30°	45°	60°	75°
0.005	0.750	1.000	0.991	0.956	0.895	0.769	0.457
0.010	0.947	1.000	0.996	0.994	0.987	0.964	0.904
0.015	0.981	1.000	1.000	1.001	0.994	0.992	0.954
0.020	1.045	1.000	0.996	0.996	0.987	0.982	0.948
0.030	1.230	1.000	0.990	0.989	0.972	0.946	0.897
0.040	1.444	1.000	0.994	0.990	0.965	0.923	0.857
0.050	1.632	1.000	0.994	0.979	0.954	0.907	0.828
0.060	1.716	1.000	0.995	0.984	0.961	0.913	0.837
0.080	1.732	1.000	0.994	0.991	0.966	0.927	0.855
0.100	1.669	1.000	0.993	0.990	0.973	0.946	0.887
0.150	1.518	1.000	1.001	1.005	0.995	0.977	0.950
0.200	1.432	1.000	1.001	1.001	1.003	0.997	0.981
0.300	1.336	1.000	1.002	1.007	1.010	1.019	1.013
0.400	1.280	1.000	1.002	1.009	1.016	1.032	1.035
0.500	1.244	1.000	1.002	1.008	1.020	1.040	1.054
0.600	1.220	1.000	1.003	1.009	1.019	1.043	1.057
0.800	1.189	1.000	1.001	1.008	1.019	1.043	1.062
1.000	1.173	1.000	1.002	1.005	1.016	1.038	1.060

TABLE V–3. SPECIFICATION FOR ISO PHOTON REFERENCE RADIATIONS, NARROW SPECTRUM SERIES (X RAYS AND GAMMA RADIATION) [V–1]

(a) Fluorescent radiations

Mean energy (keV)	Tube high voltage (kVp)	Total primary filtration (g/cm^2)	Radiator	Secondary filtration (g/cm^2)
9.9	60	A1 0.135	Germanium	GdO 0.020
17.5	80	A1 0.27	Molybdenum	Zr 0.035
23.2	100	A1 0.27	Cadmium	Ag 0.053
25.3	100	A1 0.27	Tin	Ag 0.071
31.0	100	A1 0.27	Caesium	TeO_2 0.132

(b) Filtered X rays

Mean energy[a] (keV)	Resolution R_e (%)	Constant potential[b] (kV)	Additional filtration[c] (mm)			Half value layer thickness (mm Cu)		Homogeneity coefficient
			Pb	Sn	Cu	First	Second	
33	30	40	–	–	0.21	0.084	0.091	0.92
48	36	60	–	–	0.6	0.24	0.26	0.92
65	32	80	–	–	2.0	0.58	0.62	0.94
83	28	100	–	–	5.0	1.11	1.17	0.95
100	27	120	–	1.0	5.0	1.71	1.77	0.97
118	37	150	–	2.5	–	2.36	2.47	0.96
164	30	200	1.0	3.0	2.0	3.99	4.05	0.99
208	28	250	3.0	2.0	–	5.19	5.23	0.99
250	27	300	5.0	3.0	–	6.12	6.15	1.00

(c) Gamma radiations

(Mean) Energy (keV)	Gamma source	First half value layer (mm Cu)
662	Caesium-137	10.3
1250	Cobalt-60	14.6

[a] The value of the mean energy adopted with a tolerance of ±3%.

[b] The constant potential is measured under load.

[c] The total filtration includes, in each case, the fixed filtration adjusted to 4 mm of aluminium.

TABLE V–4. CONVERSION COEFFICIENTS FROM AIR KERMA FOR $H_p(10)$ AND $H_p(0.07)$ IN AN ICRU SLAB FOR ISO PHOTON REFERENCE RADIATIONS [V–2]

Reference radiation[a]	Mean energy (keV)	$H_p(10,\alpha)/K_a$ for angles α (Sv/Gy)				$H_p(0.07,\alpha)/K_a$ for angles α (Sv/Gy)			
		0°	20°	40°	60°	0°	20°	40°	60°
F-Ge	9.9	–	–	–	–	0.95	0.94	0.94	0.91
F-Mo	17.5	(0.44)[b]	(0.42)[b]	(0.34)[b]	(0.19)[b]	1.01	1.01	1.00	1.00
F-Cd	23.2	0.79	0.77	0.68	0.48	1.09	1.10	1.09	1.07
F-Sn	25.3	0.89	0.87	0.78	0.58	1.14	1.14	1.12	1.09
F-Cs	31.0	1.15	1.13	1.04	0.84	1.25	1.24	1.22	1.18
N-40	33	1.17	1.15	1.06	0.85	1.27	1.26	1.24	1.19
N-60	48	1.65	1.62	1.52	1.27	1.55	1.54	1.50	1.42
N-80	65	1.88	1.86	1.76	1.50	1.72	1.70	1.66	1.58
N-100	83	1.88	1.86	1.76	1.53	1.72	1.70	1.68	1.60
N-120	100	1.81	1.79	1.71	1.51	1.67	1.66	1.63	1.58
N-150	118	1.73	1.71	1.64	1.46	1.61	1.60	1.58	1.54
N-200	164	1.57	1.56	1.51	1.38	1.49	1.49	1.49	1.46
N-250	208	1.48	1.48	1.44	1.33	1.42	1.42	1.43	1.43
N-300	250	1.42	1.42	1.40	1.30	1.38	1.38	1.40	1.40
S-Cs	662	1.21	1.22	1.22	1.19	–	–	–	–
S-Co	1250	1.15	1.15	1.16	1.14	–	–	–	–

[a] F — fluorescent series; N — narrow spectrum series; S — radionuclide sources. Number denotes tube potential.

[b] Numbers in brackets: Care needs to be taken as variations in energy distribution may have a substantial influence on the numerical values of the conversion coefficients.

TABLE V–5. AMBIENT AND PERSONAL DOSE EQUIVALENT PER UNIT NEUTRON FLUENCE, $H^*(10)/\Phi$ AND $H_{P,SLAB}(10,\alpha)/\Phi$ FOR MONOENERGETIC NEUTRONS INCIDENT IN VARIOUS GEOMETRIES ON THE ICRU SPHERE AND SLAB [V–4]

Neutron energy (MeV)	$H^*(10)/\Phi$ (pSv cm^2)	$H_p(10,\alpha)/\Phi$ (pSv cm^2) for angles α					
		0°	15°	30°	45°	60°	75°
1.00×10^{-9}	6.60	8.19	7.64	6.57	4.23	2.61	1.13
1.00×10^{-8}	9.00	9.97	9.35	7.90	5.38	3.37	1.50
2.53×10^{-8}	10.6	11.4	10.6	9.11	6.61	4.04	1.73
1.00×10^{-7}	12.9	12.6	11.7	10.3	7.84	4.70	1.94
2.00×10^{-7}	13.5	13.5	12.6	11.1	8.73	5.21	2.12
5.00×10^{-7}	13.6	14.2	13.5	11.8	9.40	5.65	2.31
1.00×10^{-6}	13.3	14.4	13.9	12.0	9.56	5.82	2.40
2.00×10^{-6}	12.9	14.3	14.0	11.9	9.49	5.85	2.46
5.00×10^{-6}	12.0	13.8	13.9	11.5	9.11	5.71	2.48
1.00×10^{-5}	11.3	13.2	13.4	11.0	8.65	5.47	2.44
2.00×10^{-5}	10.6	12.4	12.6	10.4	8.10	5.14	2.35
5.00×10^{-5}	9.90	11.2	11.2	9.42	7.32	4.57	2.16
1.00×10^{-4}	9.40	10.3	9.85	8.64	6.74	4.10	1.99
2.00×10^{-4}	8.90	9.84	9.41	8.22	6.21	3.91	1.83
5.00×10^{-4}	8.30	9.34	8.66	7.66	5.67	3.58	1.68
1.00×10^{-3}	7.90	8.78	8.20	7.29	5.43	3.46	1.66
2.00×10^{-3}	7.70	8.72	8.22	7.27	5.43	3.46	1.67
5.00×10^{-3}	8.00	9.36	8.79	7.46	5.71	3.59	1.69
1.00×10^{-2}	10.5	11.2	10.8	9.18	7.09	4.32	1.77
2.00×10^{-2}	16.6	17.1	17.0	14.6	11.6	6.64	2.11
3.00×10^{-2}	23.7	24.9	24.1	21.3	16.7	9.81	2.85
5.00×10^{-2}	41.1	39.0	36.0	34.4	27.5	16.7	4.78
7.00×10^{-2}	60.0	59.0	55.8	52.6	42.9	27.3	8.10
1.00×10^{-1}	88.0	90.6	87.8	81.3	67.1	44.6	13.7
1.50×10^{-1}	132	139	137	126	106	73.3	24.2
2.00×10^{-1}	170	180	179	166	141	100	35.5
3.00×10^{-1}	233	246	244	232	201	149	58.5
5.00×10^{-1}	322	335	330	326	291	226	102
7.00×10^{-1}	375	386	379	382	348	279	139
9.00×10^{-1}	400	414	407	415	383	317	171
1.00×10^{0}	416	422	416	426	395	332	180
1.20×10^{0}	425	433	427	440	412	355	210

TABLE V–5. (cont.)

Neutron energy (MeV)	$H^*(10)/\Phi$ (pSv cm^2)	$H_p(10,\alpha)/\Phi$ (pSv cm^2) for angles α					
		0°	15°	30°	45°	60°	75°
2.00×10^0	420	442	438	457	439	402	274
3.00×10^0	412	431	429	449	440	412	306
4.00×10^0	408	422	421	440	435	409	320
5.00×10^0	405	420	418	437	435	409	331
6.00×10^0	400	423	422	440	439	414	345
7.00×10^0	405	432	432	449	448	425	361
8.00×10^0	409	445	445	462	460	440	379
9.00×10^0	420	461	462	478	476	458	399
1.00×10^1	440	480	481	497	493	480	421
1.20×10^1	480	517	519	536	529	523	464
1.40×10^1	520	550	552	570	561	562	503
1.50×10^1	540	564	565	584	575	579	520
1.60×10^1	555	576	577	597	588	593	535
1.80×10^1	570	595	593	617	609	615	561
2.00×10^1	600	600	595	619	615	619	570
3.00×10^1	515						
5.00×10^1	400						
7.50×10^1	330						
1.00×10^2	285						
1.25×10^2	260						
1.50×10^2	245						
1.75×10^2	250						
2.01×10^2	260						

TABLE V–6. REFERENCE CONVERSION COEFFICIENTS FOR ELECTRONS, NORMAL INCIDENCE [V–4]

Electron energy (MeV)	$H'(0.07,0°)/\Phi$ (nSv cm^2)	$H'(3,0°)/\Phi$ (nSv cm^2)	$H'(10,0°)/\Phi$ (nSv cm^2)
0.07	0.221		
0.08	1.056		
0.09	1.527		
0.10	1.661		
0.1125	1.627		
0.125	1.513		
0.15	1.229		
0.20	0.834		
0.30	0.542		
0.40	0.455		
0.50	0.403		
0.60	0.366		
0.70	0.344	0.000	
0.80	0.329	0.045	
1.00	0.312	0.301	
1.25	0.296	0.486	
1.50	0.287	0.524	
1.75	0.282	0.512	0.000
2.00	0.279	0.481	0.005
2.50	0.278	0.417	0.156
3.00	0.276	0.373	0.336
3.50	0.274	0.351	0.421
4.00	0.272	0.334	0.447
5.00	0.271	0.317	0.430
6.00	0.271	0.309	0.389
7.00	0.271	0.306	0.360
8.00	0.271	0.305	0.341
10.00	0.275	0.303	0.330

TABLE V–7. CONVERSION COEFFICIENTS FOR AMBIENT DOSE EQUIVALENT $H^*(10)$ AND DIRECTIONAL DOSE EQUIVALENT $H'(0.07,0°)$ FROM AIR KERMA K_a (PHOTONS) [V–4]

Photon energy (MeV)	$H^*(10)/K_a$ (Sv/Gy))	$H'(0.07,0°)/K_a$ (Sv/Gy)
0.01	0.008	0.95
0.015	0.26	0.99
0.020	0.61	1.05
0.030	1.10	1.22
0.040	1.47	1.41
0.050	1.67	1.53
0.060	1.74	1.59
0.080	1.72	1.61
0.100	1.65	1.55
0.150	1.49	1.42
0.200	1.40	1.34
0.300	1.31	1.31
0.400	1.26	1.26
0.500	1.23	1.23
0.600	1.21	1.21
0.800	1.19	1.19
1	1.17	1.17
1.5	1.15	1.15
2	1.14	1.14
3	1.13	1.13
4	1.12	1.12
5	1.11	1.11
6	1.11	1.11
8	1.11	1.11
10	1.10	1.10

TABLE V–8. CONVERSION COEFFICIENTS FROM AIR KERMA TO $H'(0.07,0°)$ AND ANGULAR DEPENDENCE FACTORS UP TO 180° (PHOTONS) [V–4]

Photon energy (MeV)	$H'(0.07,0°)/K_a$ (Sv/Gy)	Ratio $H'(0.07,\alpha)/H'(0.07,0°)$ for angles α							
		0°	15°	30°	45°	60°	75°	90°	180°
0.005	0.76	1.00	0.96	0.87	0.79	0.41	0.00	0.00	0.00
0.010	0.95	1.00	0.99	0.98	0.98	0.96	0.89	0.19	0.00
0.020	1.05	1.00	1.00	0.99	1.00	1.00	0.98	0.54	0.00
0.030	1.22	1.00	0.99	0.99	0.99	0.98	0.94	0.62	0.00
0.050	1.53	1.00	0.99	0.98	0.98	0.97	0.92	0.69	0.02
0.100	1.55	1.00	0.99	0.99	0.99	0.98	0.94	0.77	0.05
0.150	1.42	1.00	0.99	0.99	0.99	0.99	0.97	0.87	0.07
0.300	1.31	1.00	1.00	1.00	1.00	1.02	1.00	0.89	0.10
0.662	1.20	1.00	1.00	1.00	1.00	1.00	0.98	0.89	0.18
1.25	1.16	1.00	1.00	1.00	1.00	1.00	0.98	0.90	0.30
2	1.14	1.00	1.00	1.00	1.00	1.00	0.98	0.90	0.39
3	1.13	1.00	1.00	1.00	1.00	1.00	0.98	0.90	0.46
5	1.11	1.00	1.00	1.00	1.00	1.00	0.98	0.91	0.54
10	1.10	1.00	1.00	1.00	1.00	1.00	0.98	0.94	0.63

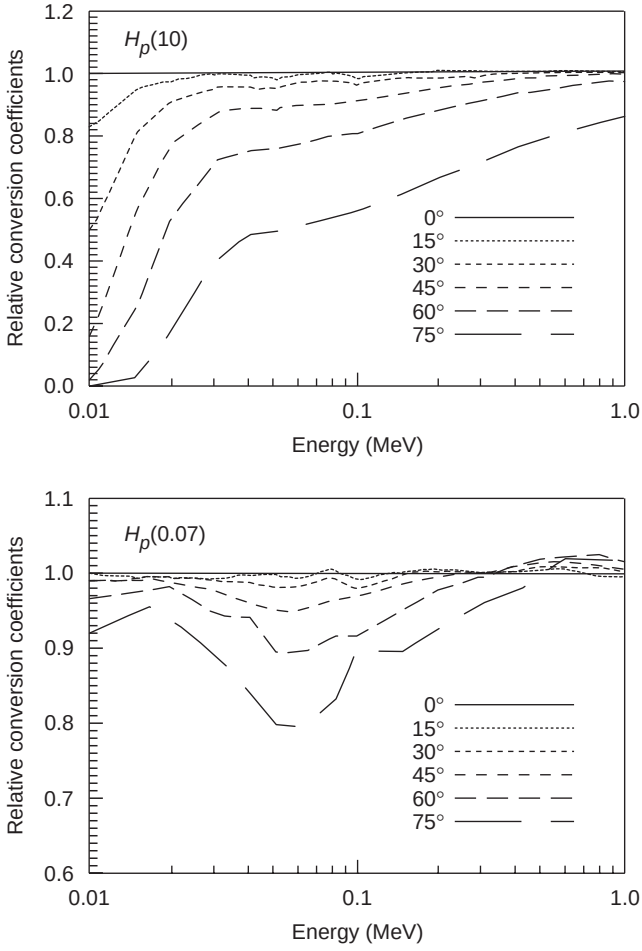

FIG. V–1. Angular dependence of photon conversion coefficients for $H_p(10)$ and $H_p(0.07)$ in an ICRU slab (after Ref. [11]).

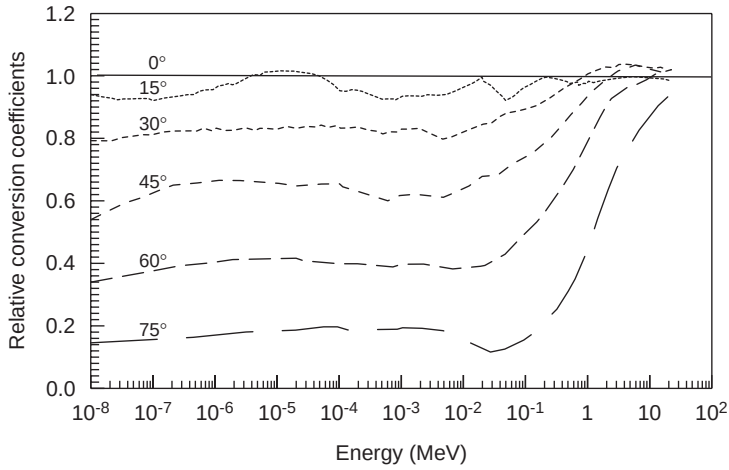

FIG. V–2. Angular dependence of neutron conversion coefficients for $H_p(10)$ in an ICRU slab (after Ref. [11]).

REFERENCES TO ANNEX V

[V–1] INTERNATIONAL ORGANIZATION FOR STANDARDIZATION, X and Gamma Reference Radiations for Calibrating Dosemeters and Doserate Meters and for Determining Their Response as a Function of Photon Energy, ISO 4037/Part 1: Radiation Characteristics and Production Methods, ISO, Geneva (1996).

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Annex VI

EXAMPLES OF IEC STANDARDS ON RADIATION MONITORING EQUIPMENT

Publication number	Equipment
	Photon and beta monitoring equipment
1018	High range beta and photon dose and dose rate portable instruments for emergency radiation protection purposes
532	Installed dose rate meters, warning assemblies and monitors for X or gamma radiation of energy between 50 keV and 7 MeV
846	Beta, X and gamma radiation dose equivalent and dose equivalent ratemeters for use in radiation protection
1017-1	Portable, transportable or installed X or gamma radiation ratemeters for environmental monitoring — Part 1: Ratemeters;
1017-2	Part 2: Integrating assemblies
	Personal dosimetry
1066	Thermoluminescence dosimetry systems for personal and environmental monitoring
	Neutron monitoring equipment
1005	Portable neutron ambient dose equivalent ratemeters for use in radiation protection
—	Direct reading personal dose equivalent and/or dose equivalent rate monitors for neutron radiation

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