IAEA Safety Standards

for protecting people and the environment

Occupational Radiation Protection

Jointly sponsored by

General Safety Guide

No. GSG-7

IAEA
International Atomic Energy Agency

IAEA SAFETY STANDARDS AND RELATED PUBLICATIONS

IAEA SAFETY STANDARDS

Under the terms of Article III of its Statute, the IAEA is authorized to establish or adopt standards of safety for protection of health and minimization of danger to life and property, and to provide for the application of these standards.

The publications by means of which the IAEA establishes standards are issued in the **IAEA Safety Standards Series**. This series covers nuclear safety, radiation safety, transport safety and waste safety. The publication categories in the series are **Safety Fundamentals**, **Safety Requirements** and **Safety Guides**.

Information on the IAEA's safety standards programme is available on the IAEA Internet site

http://www-ns.iaea.org/standards/

The site provides the texts in English of published and draft safety standards. The texts of safety standards issued in Arabic, Chinese, French, Russian and Spanish, the IAEA Safety Glossary and a status report for safety standards under development are also available. For further information, please contact the IAEA at: Vienna International Centre, PO Box 100, 1400 Vienna, Austria.

All users of IAEA safety standards are invited to inform the IAEA of experience in their use (e.g. as a basis for national regulations, for safety reviews and for training courses) for the purpose of ensuring that they continue to meet users' needs. Information may be provided via the IAEA Internet site or by post, as above, or by email to Official.Mail@iaea.org.

RELATED PUBLICATIONS

The IAEA provides for the application of the standards and, under the terms of Articles III and VIII.C of its Statute, makes available and fosters the exchange of information relating to peaceful nuclear activities and serves as an intermediary among its Member States for this purpose.

Reports on safety in nuclear activities are issued as **Safety Reports**, which provide practical examples and detailed methods that can be used in support of the safety standards.

Other safety related IAEA publications are issued as **Emergency Preparedness and Response** publications, **Radiological Assessment Reports**, the International Nuclear Safety Group's **INSAG Reports**, **Technical Reports** and **TECDOCs**. The IAEA also issues reports on radiological accidents, training manuals and practical manuals, and other special safety related publications.

Security related publications are issued in the **IAEA Nuclear Security Series**.

The **IAEA Nuclear Energy Series** comprises informational publications to encourage and assist research on, and the development and practical application of, nuclear energy for peaceful purposes. It includes reports and guides on the status of and advances in technology, and on experience, good practices and practical examples in the areas of nuclear power, the nuclear fuel cycle, radioactive waste management and decommissioning.

OCCUPATIONAL
RADIATION PROTECTION

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''.

IAEA SAFETY STANDARDS SERIES No. GSG-7

OCCUPATIONAL RADIATION PROTECTION

GENERAL SAFETY GUIDE

JOINTLY SPONSORED BY THE
INTERNATIONAL ATOMIC ENERGY AGENCY
AND INTERNATIONAL LABOUR OFFICE

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2018

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email: sales.publications@iaea.org
www.iaea.org/books

Printed by the IAEA in Austria
October 2018
STI/PUB/1785

IAEA Library Cataloguing in Publication Data

Names: International Atomic Energy Agency.
Title: Occupational radiation protection / International Atomic Energy Agency.
Description: Vienna : International Atomic Energy Agency, 2018. | Series: IAEA safety standards series, ISSN 1020–525X ; no. GSG-7 | Includes bibliographical references.
Identifiers: IAEAL 18-01183 | ISBN 978–92–0–102917–1 (paperback : alk. paper)
Subjects: LCSH: Radiation — Safety measures. | Radiation dosimetry. | Industrial safety.
Classification: UDC 614.8.086.5 | STI/PUB/1785

FOREWORD

by Yukiya Amano
Director General

The IAEA's Statute authorizes the Agency to "establish or adopt… standards of safety for protection of health and minimization of danger to life and property" — standards that the IAEA must use in its own operations, and which States can apply by means of their regulatory provisions for nuclear and radiation safety. The IAEA does this in consultation with the competent organs of the United Nations and with the specialized agencies concerned. A comprehensive set of high quality standards under regular review is a key element of a stable and sustainable global safety regime, as is the IAEA's assistance in their application.

The IAEA commenced its safety standards programme in 1958. The emphasis placed on quality, fitness for purpose and continuous improvement has led to the widespread use of the IAEA standards throughout the world. The Safety Standards Series now includes unified Fundamental Safety Principles, which represent an international consensus on what must constitute a high level of protection and safety. With the strong support of the Commission on Safety Standards, the IAEA is working to promote the global acceptance and use of its standards.

Standards are only effective if they are properly applied in practice. The IAEA's safety services encompass design, siting and engineering safety, operational safety, radiation safety, safe transport of radioactive material and safe management of radioactive waste, as well as governmental organization, regulatory matters and safety culture in organizations. These safety services assist Member States in the application of the standards and enable valuable experience and insights to be shared.

Regulating safety is a national responsibility, and many States have decided to adopt the IAEA's standards for use in their national regulations. For parties to the various international safety conventions, IAEA standards provide a consistent, reliable means of ensuring the effective fulfilment of obligations under the conventions. The standards are also applied by regulatory bodies and operators around the world to enhance safety in nuclear power generation and in nuclear applications in medicine, industry, agriculture and research.

Safety is not an end in itself but a prerequisite for the purpose of the protection of people in all States and of the environment — now and in the future. The risks associated with ionizing radiation must be assessed and controlled without unduly limiting the contribution of nuclear energy to equitable and sustainable development. Governments, regulatory bodies and operators everywhere must ensure that nuclear material and radiation sources are used beneficially, safely and ethically. The IAEA safety standards are designed to facilitate this, and I encourage all Member States to make use of them.

PREFACE

Occupational exposure to ionizing radiation can occur in a range of industries, medical institutions, educational and research establishments, and nuclear fuel cycle facilities. Appropriate levels of radiation protection of workers are essential for the safe and justified use of radiation, radioactive material and nuclear energy.

In 2006, the IAEA published the Fundamental Safety Principles, IAEA Safety Standards Series No. SF-1, jointly sponsored by the European Atomic Energy Community (Euratom), the Food and Agriculture Organization of the United Nations (FAO), the IAEA, the International Labour Organization (ILO), the International Maritime Organization, the OECD Nuclear Energy Agency (OECD/NEA), the Pan American Health Organization (PAHO), the United Nations Environment Programme (UNEP) and the World Health Organization (WHO). That publication sets out the safety objective and the principles of protection and safety.

In 2014, the IAEA published Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards, IAEA Safety Standards Series No. GSR Part 3, jointly sponsored by the European Commission, FAO, IAEA, ILO, OECD/NEA, PAHO, UNEP and WHO. This publication establishes the requirements that are designed to meet the fundamental safety objective and to apply the safety principles specified in the Fundamental Safety Principles.

The establishment of safety requirements and guidance on occupational radiation protection is a major element of the support for radiation protection and safety provided by the IAEA to its Member States. The objective of the IAEA's activities on occupational radiation protection is to promote an internationally harmonized approach to occupational radiation protection through the development and application of standards for optimizing protection and safety, restricting exposures and applying current radiation protection techniques in the workplace.

This Safety Guide was prepared jointly by the IAEA and the International Labour Office to provide guidance on fulfilling the requirements of GSR Part 3 with respect to occupational exposure. It provides general guidance on the exposure conditions for which radiation protection programmes are required to be established, including the setting up of monitoring programmes to assess radiation doses to workers arising from exposure due to external sources of radiation and from exposure due to intakes of radionuclides. It also provides specific guidance on the assessment of doses from exposure due to external sources of radiation and from exposure due to intakes of radionuclides.

THE IAEA SAFETY STANDARDS

BACKGROUND

Radioactivity is a natural phenomenon and natural sources of radiation are features of the environment. Radiation and radioactive substances have many beneficial applications, ranging from power generation to uses in medicine, industry and agriculture. The radiation risks to workers and the public and to the environment that may arise from these applications have to be assessed and, if necessary, controlled.

Activities such as the medical uses of radiation, the operation of nuclear installations, the production, transport and use of radioactive material, and the management of radioactive waste must therefore be subject to standards of safety.

Regulating safety is a national responsibility. However, radiation risks may transcend national borders, and international cooperation serves to promote and enhance safety globally by exchanging experience and by improving capabilities to control hazards, to prevent accidents, to respond to emergencies and to mitigate any harmful consequences.

States have an obligation of diligence and duty of care, and are expected to fulfil their national and international undertakings and obligations.

International safety standards provide support for States in meeting their obligations under general principles of international law, such as those relating to environmental protection. International safety standards also promote and assure confidence in safety and facilitate international commerce and trade.

A global nuclear safety regime is in place and is being continuously improved. IAEA safety standards, which support the implementation of binding international instruments and national safety infrastructures, are a cornerstone of this global regime. The IAEA safety standards constitute a useful tool for contracting parties to assess their performance under these international conventions.

THE IAEA SAFETY STANDARDS

The status of the IAEA safety standards derives from the IAEA's Statute, which authorizes the IAEA to establish or adopt, in consultation and, where appropriate, in collaboration with the competent organs of the United Nations and with the specialized agencies concerned, standards of safety for protection of health and minimization of danger to life and property, and to provide for their application.

With a view to ensuring the protection of people and the environment from harmful effects of ionizing radiation, the IAEA safety standards establish fundamental safety principles, requirements and measures to control the radiation exposure of people and the release of radioactive material to the environment, to restrict the likelihood of events that might lead to a loss of control over a nuclear reactor core, nuclear chain reaction, radioactive source or any other source of radiation, and to mitigate the consequences of such events if they were to occur. The standards apply to facilities and activities that give rise to radiation risks, including nuclear installations, the use of radiation and radioactive sources, the transport of radioactive material and the management of radioactive waste.

Safety measures and security measures[1] have in common the aim of protecting human life and health and the environment. Safety measures and security measures must be designed and implemented in an integrated manner so that security measures do not compromise safety and safety measures do not compromise security.

The IAEA safety standards reflect an international consensus on what constitutes a high level of safety for protecting people and the environment from harmful effects of ionizing radiation. They are issued in the IAEA Safety Standards Series, which has three categories (see Fig. 1).

Safety Fundamentals

Safety Fundamentals present the fundamental safety objective and principles of protection and safety, and provide the basis for the safety requirements.

Safety Requirements

An integrated and consistent set of Safety Requirements establishes the requirements that must be met to ensure the protection of people and the environment, both now and in the future. The requirements are governed by the objective and principles of the Safety Fundamentals. If the requirements are not met, measures must be taken to reach or restore the required level of safety. The format and style of the requirements facilitate their use for the establishment, in a harmonized manner, of a national regulatory framework. Requirements, including numbered ‘overarching’ requirements, are expressed as ‘shall’ statements. Many requirements are not addressed to a specific party, the implication being that the appropriate parties are responsible for fulfilling them.

[1] See also publications issued in the IAEA Nuclear Security Series.

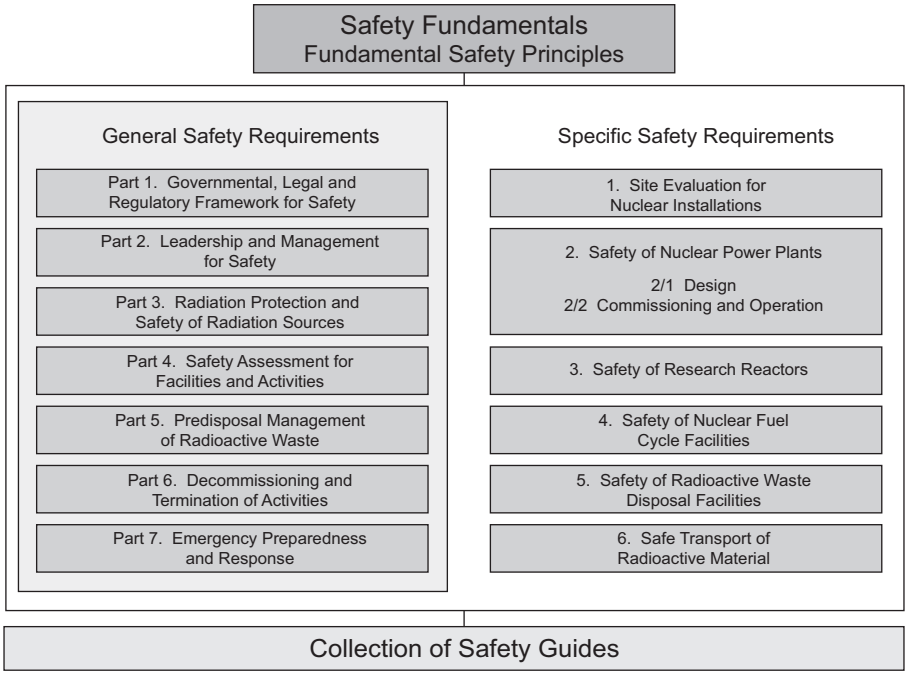

FIG. 1. The long term structure of the IAEA Safety Standards Series.

Safety Guides

Safety Guides provide recommendations and guidance on how to comply with the safety requirements, indicating an international consensus that it is necessary to take the measures recommended (or equivalent alternative measures). The Safety Guides present international good practices, and increasingly they reflect best practices, to help users striving to achieve high levels of safety. The recommendations provided in Safety Guides are expressed as 'should' statements.

APPLICATION OF THE IAEA SAFETY STANDARDS

The principal users of safety standards in IAEA Member States are regulatory bodies and other relevant national authorities. The IAEA safety standards are also used by co-sponsoring organizations and by many organizations that design, construct and operate nuclear facilities, as well as organizations involved in the use of radiation and radioactive sources.

The IAEA safety standards are applicable, as relevant, throughout the entire lifetime of all facilities and activities — existing and new — utilized for peaceful purposes and to protective actions to reduce existing radiation risks. They can be used by States as a reference for their national regulations in respect of facilities and activities.

The IAEA's Statute makes the safety standards binding on the IAEA in relation to its own operations and also on States in relation to IAEA assisted operations.

The IAEA safety standards also form the basis for the IAEA's safety review services, and they are used by the IAEA in support of competence building, including the development of educational curricula and training courses.

International conventions contain requirements similar to those in the IAEA safety standards and make them binding on contracting parties. The IAEA safety standards, supplemented by international conventions, industry standards and detailed national requirements, establish a consistent basis for protecting people and the environment. There will also be some special aspects of safety that need to be assessed at the national level. For example, many of the IAEA safety standards, in particular those addressing aspects of safety in planning or design, are intended to apply primarily to new facilities and activities. The requirements established in the IAEA safety standards might not be fully met at some existing facilities that were built to earlier standards. The way in which IAEA safety standards are to be applied to such facilities is a decision for individual States.

The scientific considerations underlying the IAEA safety standards provide an objective basis for decisions concerning safety; however, decision makers must also make informed judgements and must determine how best to balance the benefits of an action or an activity against the associated radiation risks and any other detrimental impacts to which it gives rise.

DEVELOPMENT PROCESS FOR THE IAEA SAFETY STANDARDS

The preparation and review of the safety standards involves the IAEA Secretariat and five safety standards committees, for emergency preparedness and response (EPReSC) (as of 2016), nuclear safety (NUSSC), radiation safety (RASSC), the safety of radioactive waste (WASSC) and the safe transport of radioactive material (TRANSSC), and a Commission on Safety Standards (CSS) which oversees the IAEA safety standards programme (see Fig. 2).

All IAEA Member States may nominate experts for the safety standards committees and may provide comments on draft standards. The membership of

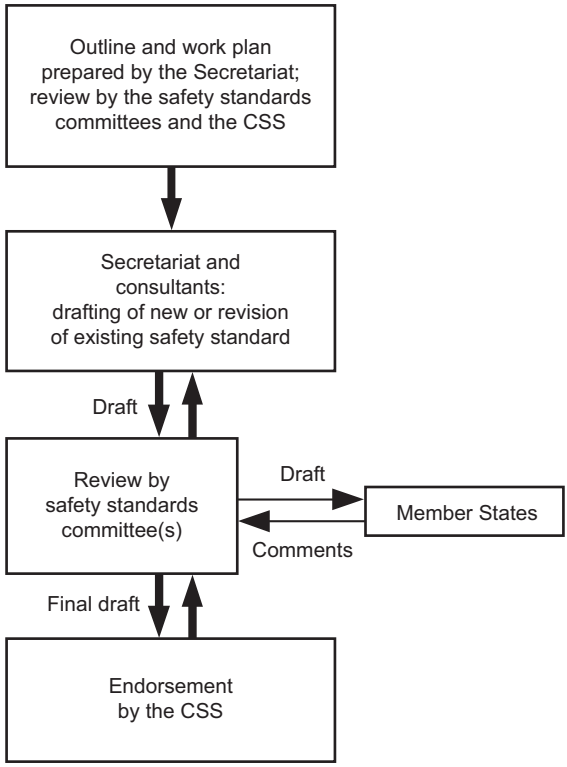

FIG. 2. The process for developing a new safety standard or revising an existing standard.

the Commission on Safety Standards is appointed by the Director General and includes senior governmental officials having responsibility for establishing national standards.

A management system has been established for the processes of planning, developing, reviewing, revising and establishing the IAEA safety standards. It articulates the mandate of the IAEA, the vision for the future application of the safety standards, policies and strategies, and corresponding functions and responsibilities.

INTERACTION WITH OTHER INTERNATIONAL ORGANIZATIONS

The findings of the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) and the recommendations of international

expert bodies, notably the International Commission on Radiological Protection (ICRP), are taken into account in developing the IAEA safety standards. Some safety standards are developed in cooperation with other bodies in the United Nations system or other specialized agencies, including the Food and Agriculture Organization of the United Nations, the United Nations Environment Programme, the International Labour Organization, the OECD Nuclear Energy Agency, the Pan American Health Organization and the World Health Organization.

INTERPRETATION OF THE TEXT

Safety related terms are to be understood as defined in the IAEA Safety Glossary (see http://www-ns.iaea.org/standards/safety-glossary.htm). Otherwise, words are used with the spellings and meanings assigned to them in the latest edition of The Concise Oxford Dictionary. For Safety Guides, the English version of the text is the authoritative version.

The background and context of each standard in the IAEA Safety Standards Series and its objective, scope and structure are explained in Section 1, Introduction, of each publication.

Material for which there is no appropriate place in the body text (e.g. material that is subsidiary to or separate from the body text, is included in support of statements in the body text, or describes methods of calculation, procedures or limits and conditions) may be presented in appendices or annexes.

An appendix, if included, is considered to form an integral part of the safety standard. Material in an appendix has the same status as the body text, and the IAEA assumes authorship of it. Annexes and footnotes to the main text, if included, are used to provide practical examples or additional information or explanation. Annexes and footnotes are not integral parts of the main text. Annex material published by the IAEA is not necessarily issued under its authorship; material under other authorship may be presented in annexes to the safety standards. Extraneous material presented in annexes is excerpted and adapted as necessary to be generally useful.

CONTENTS

1. INTRODUCTION

BACKGROUND

1.1. Occupational exposure to radiation can occur as a result of various human activities, including: work associated with the different stages of the nuclear fuel cycle; the use of radiation in medicine, scientific research, agriculture and industry; and occupations that involve exposure due to natural sources.

1.2. IAEA Safety Standards Series No. SF-1, Fundamental Safety Principles [1], presents the fundamental safety objective and principles of protection and safety. Requirements designed to meet the fundamental safety objective and to apply the safety principles specified in SF-1 [1], including requirements for the protection of workers exposed to sources of radiation, are established in IAEA Safety Standards Series No. GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards [2], jointly sponsored by the IAEA and seven other international organizations.

1.3. This Safety Guide, prepared jointly by the IAEA and the International Labour Office, provides guidance on fulfilling the requirements of GSR Part 3 [2] with respect to occupational exposure. It provides general guidance on the exposure conditions for which radiation protection programmes are required to be established, including the setting up of monitoring programmes to assess radiation doses to workers arising from exposure due to external sources of radiation and from exposure due to intakes of radionuclides. It also provides specific guidance on the assessment of doses from exposure due to external sources of radiation and from exposure due to intakes of radionuclides.

1.4. Recommendations for a system of radiation protection were developed by the International Commission on Radiological Protection (ICRP) [3]. These and other current recommendations of the ICRP and the International Commission on Radiation Units and Measurements (ICRU) have been taken into account in preparing this Safety Guide.

1.5. It is recognized that radiation protection is only one component to be addressed to protect the overall health and safety of the worker. The radiation protection programme has to be established and managed together with other health and safety disciplines, such as industrial hygiene, medical hygiene, industrial safety and fire safety.

1.6. Guidance on meeting the requirements of GSR Part 3 [2] for occupational radiation protection is provided in this Safety Guide. It gives general guidance on the development of occupational radiation protection programmes, in accordance with the requirements of GSR Part 3 [2] and appropriate for the sources of radiation likely to be encountered in the workplaces in question. It also gives more detailed guidance on the monitoring and assessment of workers' exposure due to external radiation sources and from intakes of radionuclides. This Safety Guide reflects the current internationally accepted principles and recommended practices in occupational radiation protection, with account taken of the major changes that have occurred since 1999.

1.7. This Safety Guide updates the guidance given in five previous Safety Guides, which are hereby superseded: Occupational Radiation Protection[1]; Assessment of Occupational Exposure Due to Intakes of Radionuclides[2]; Assessment of Occupational Exposure Due to External Sources of Radiation[3]; Occupational Radiation Protection in the Mining and Processing of Raw Materials[4]; and The Management System for Technical Services in Radiation Safety[5].

OBJECTIVE

1.8. The objective of this Safety Guide is to provide guidance on the control of occupational exposure. The recommendations given are intended for regulatory bodies, but this Safety Guide will also be useful to: employers, licensees and registrants; management bodies and their specialist advisers; and health and safety committees concerned with the radiation protection of workers. The

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[1] INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR OFFICE, Occupational Radiation Protection, IAEA Safety Standards Series No. RS-G-1.1, IAEA, Vienna (1999).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR OFFICE, Assessment of Occupational Exposure Due to Intakes of Radionuclides, IAEA Safety Standards Series No. RS-G-1.2, IAEA, Vienna (1999).

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR OFFICE, Assessment of Occupational Exposure Due to External Sources of Radiation, IAEA Safety Standards Series No. RS-G-1.3, IAEA, Vienna (1999).

[4] INTERNATIONAL LABOUR OFFICE, Occupational Radiation Protection in the Mining and Processing of Raw Materials, IAEA Safety Standards Series No. RS-G-1.6, IAEA, Vienna (2004).

[5] INTERNATIONAL ATOMIC ENERGY AGENCY, The Management System for Technical Services in Radiation Safety, IAEA Safety Standards Series No. GS-G-3.2, IAEA, Vienna (2008).

recommendations may also be used by workers and their representatives to encourage safe working practices.

SCOPE

1.9. This Safety Guide addresses the technical and organizational aspects of the control of occupational exposure. The intention is to provide an integrated approach to the control of exposure, and the control of potential exposure, due to external exposure and internal exposure to radiation from both artificial sources and natural sources.

STRUCTURE

1.10. Section 2 gives an overview of the basic framework for occupational radiation protection, including an explanation of the three types of exposure situation (planned exposure situations, emergency exposure situations and existing exposure situations), the basic principles of radiation protection and their application to the protection of workers, and the dosimetric quantities used. Sections 3–5 provide guidance on meeting the requirements of GSR Part 3 [2] in each of the three types of exposure situation. Section 3 addresses occupational radiation protection in planned exposure situations, including application of the principles of optimization and of dose limitation, the radiation protection programme, and specific guidance on the protection of workers exposed due to natural sources. Section 4 addresses the protection of workers in a nuclear or radiological emergency, including the preparation of an emergency plan, the application of the principles of optimization and dose limitation in emergencies, and the assessment and management of exposures of emergency workers. Section 5 addresses the protection of workers in existing exposure situations, including the establishment of an appropriate protection strategy and legal and regulatory framework. It also provides specific guidance on the protection of workers against exposure due to residual radioactive material from past activities or accidents, radon in workplaces and cosmic radiation in aircraft and spacecraft.

1.11. Sections 6–10 provide guidance on more specific aspects of occupational radiation protection. Section 6 describes the special measures to be taken for the protection of two particular groups of workers: female workers during and after pregnancy and itinerant workers. Section 7 gives detailed guidance on the monitoring and assessment of occupational exposure, including: monitoring programmes, systems and equipment; the estimation of uncertainties; testing and

calibration; the interpretation of the results of monitoring; and the maintenance of records. The guidance covers both individual monitoring and workplace monitoring, addresses external exposure and internal exposure as well as skin contamination, and includes assessment of exposure in emergencies. Section 8 gives guidance on the management system for providers of technical services in occupational radiation protection, in particular calibration, testing and dosimetry services. Section 9 describes engineered and administrative controls for occupational protection and safety, including the maintenance of good air quality, the provision of adequate shielding and the control of contamination. Guidance on the use of personal protective equipment is also provided. Section 10 addresses programmes for workers' health surveillance, including guidance on the medical examination of workers and medical records, as well as on the care of overexposed workers.

1.12. Five appendices and an annex provide additional, more detailed guidance relating to the exposure of workers due to naturally occurring radioactive material, methods and systems for individual monitoring for assessment of external exposure, workplace monitoring instruments for assessment of external exposure, methods for monitoring and assessment of internal exposure (including biokinetic modelling), and techniques for retrospective dosimetry.

2. FRAMEWORK FOR OCCUPATIONAL RADIATION PROTECTION

OCCUPATIONAL EXPOSURE AND TYPES OF EXPOSURE SITUATION

2.1. Occupational exposure is the exposure of workers incurred during the course of their work, regardless of the exposure situation. For the purpose of establishing practical requirements for protection and safety, GSR Part 3 [2] distinguishes between three different types of exposure situation: planned exposure situations, emergency exposure situations and existing exposure situations. As stated in para. 1.20 of GSR Part 3 [2]:

> "(a) A *planned exposure situation* is a situation of exposure that arises from the planned operation of a source or from a planned activity that results in an exposure due to a source. Since provision for protection and safety can be made before embarking on the activity concerned, the associated exposures and their likelihood of occurrence can be

restricted from the outset. The primary means of controlling exposure in planned exposure situations is by good design of facilities, equipment and operating procedures, and by training….

(b) An *emergency exposure situation* is a situation of exposure that arises as a result of an accident, a malicious act or any other unexpected event, and requires prompt action in order to avoid or to reduce adverse consequences. Preventive measures and mitigatory actions have to be considered before an emergency exposure situation arises. However, once an emergency exposure situation actually arises, exposures can be reduced only by implementing protective actions.

(c) An *existing exposure situation* is a situation of exposure that already exists when a decision on the need for control needs to be taken. Existing exposure situations include situations of exposure to natural background radiation. They also include situations of exposure due to residual radioactive material that derives from past practices that were not subject to regulatory control or that remains after an emergency exposure situation."

This Safety Guide gives guidance on the protection of workers in each of these three types of exposure situation.

2.2. As stated in para. 1.21 of GSR Part 3 [2]:

> "The descriptions that are given in para. 1.20 of the three types of exposure situation are not always sufficient to determine unequivocally which type of exposure situation applies for particular circumstances. For instance, the transitions from an emergency exposure situation to an existing exposure situation may occur progressively over time; and some exposures due to natural sources may have some characteristics of both planned exposure situations and existing exposure situations. In these Standards, the most appropriate type of exposure situation for particular circumstances has been determined by taking practical considerations into account."

2.3. Reference is made to potential exposure in para. 1.20(a) of GSR Part 3 [2] as follows:

> "In planned exposure situations, exposure at some level can be expected to occur. If exposure is not expected to occur with certainty, but could result from an accident or from an event or a sequence of events that may occur but is not certain to occur, this is referred to as 'potential exposure'.

…….

> "If an event or a sequence of events that has been considered in the assessment of potential exposure does actually occur, it may be treated either as a planned exposure situation or, if an emergency has been declared, as an emergency exposure situation."

2.4. Some exposures are excluded from the scope of GSR Part 3 [2]. Paragraph 1.42 of GSR Part 3 [2] states that:

> "These Standards apply to all situations involving radiation exposure that is amenable to control. Exposures deemed to be not amenable to control are excluded from the scope of these Standards.[8]
>
> "[8] It is generally accepted, for example, that it is not feasible to control ^{40}K in the body or cosmic radiation at the surface of the Earth."

Guidance is given in Section 3 on the components of exposure due to natural sources of radiation that may be required to be subject to control as occupational exposure.

PRINCIPLES OF RADIATION PROTECTION

2.5. The three general principles of radiation protection, which concern justification, optimization of protection and application of dose limits (limitation of doses), are expressed in Safety Principles 4–6 and 10 of the Fundamental Safety Principles [1]. In terms of Requirement 1 of GSR Part 3 [2], those "**with responsibilities for protection and safety shall ensure that the principles of radiation protection are applied for all exposure situations.**"

Justification

2.6. Paragraphs 2.8 and 2.9 of GSR Part 3 [2] state that:

> "2.8. For planned exposure situations, each party with responsibilities for protection and safety shall ensure, when relevant requirements apply to that party, that no practice is undertaken unless it is justified.
>
> "2.9. For emergency exposure situations and existing exposure situations, each party with responsibilities for protection and safety shall ensure, when

relevant requirements apply to that party, that protective actions or remedial actions are justified and are undertaken in such a way as to achieve the objectives set out in a protection strategy."

2.7. In planned exposure situations, this means that no practice or source within a practice should be authorized unless the practice produces sufficient benefit to the exposed individuals or to society to offset the harm (including radiation detriment) that it might cause; that is, unless the practice is justified, with societal, economic and other relevant factors having been taken into account.

2.8. The process of determining whether a practice is justified involves consideration of all radiation doses received by workers and members of the public. In general, the assumption made in this Safety Guide is that the process of justification has already taken place and that the contribution of occupational exposure to the total radiation detriment has been taken into account. The subject of justification for planned exposure situations is therefore not considered in detail in this Safety Guide. Guidance on justification is given in IAEA Safety Standards Series No. GSG-5, Justification of Practices, Including Non-medical Human Imaging [4].

Optimization

2.9. Paragraph 2.10 of GSR Part 3 [2] states that:

"For all exposure situations, each party with responsibilities for protection and safety shall ensure, when relevant requirements apply to that party, that protection and safety is optimized[9].

"[9] 'Protection and safety is optimized' means that optimization of protection and safety has been applied and the result of that process has been implemented."

2.10. In planned exposure situations, in relation to exposures due to any particular source within a practice, protection and safety is required to be optimized in order that the magnitude of individual doses, the number of people exposed and the likelihood of incurring exposures all be kept as low as reasonably achievable, economic and societal factors being taken into account, with the restriction that the doses to individuals delivered by the source be subject to dose constraints. This principle is of particular importance for the implementation of radiation protection measures in the workplace and therefore underlies much of the more detailed guidance given in Section 3.

Dose limitation

2.11. Paragraph 2.11 of GSR Part 3 [2] states that:

> "For planned exposure situations other than for medical exposure, each party with responsibilities for protection and safety shall ensure that, when relevant requirements apply to that party, specified dose limits are not exceeded."

2.12. Dose limits apply only in planned exposure situations. In such situations, the exposure of individuals should be restricted so that neither the total effective dose nor the total equivalent dose to relevant tissues or organs, caused by possible combinations of exposures due to authorized practices, exceeds any relevant dose limit.

2.13. The limit on effective dose represents the level above which the risk of stochastic effects due to radiation exposure is considered to be unacceptable. For localized exposure of the lens of the eye, the extremities and the skin, this limit on effective dose is not sufficient to ensure the avoidance of deterministic effects. Limits on equivalent dose to these tissues and organs are, therefore, specified for such situations.

2.14. Guidance on the application of the dose limits for occupational exposure is given in Section 3.

RESPONSIBILITIES

The government

2.15. The responsibilities of the government[6] with regard to protection and safety are set out in paras 2.13–2.28 of GSR Part 3 [2]. These include:

(a) Establishing an effective legal and regulatory framework for protection and safety in all exposure situations;
(b) Establishing legislation that meets specified requirements;
(c) Establishing an independent regulatory body with the necessary legal authority, competence and resources;

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[6] Since States have different legal structures, the term 'government' here is to be understood in a broad sense and, accordingly, is interchangeable with the term 'State'.

(d) Establishing requirements for education and training in protection and safety;
(e) Ensuring that arrangements are in place for the provision of technical services and education and training services.

The regulatory body

2.16. The responsibilities of the regulatory body with regard to protection and safety are set out in paras 2.29–2.38 of GSR Part 3 [2]. These include:

(a) Establishing requirements for applying the principles of radiation protection;
(b) Establishing a regulatory system that meets specified requirements;
(c) Ensuring the application of the requirements for education and training in protection and safety;
(d) Ensuring that mechanisms are in place for the dissemination of lessons learned from incidents and accidents;
(e) Setting acceptance and performance criteria for sources and equipment with implications for protection and safety;
(f) Making provision for the establishment and maintenance of records.

2.17. The responsibilities of the regulatory body specific to occupational exposure in planned exposure situations are set out in paras 3.69–3.73 of GSR Part 3 [2]. The regulatory body is responsible for establishing and enforcing requirements for ensuring that protection and safety is optimized, ensuring that applicable dose limits are complied with, and monitoring and recording occupational exposures.

Employers, registrants and licensees

2.18. Requirement 4 of GSR Part 3 [2] states that "**The person or organization responsible for facilities and activities that give rise to radiation risks shall have the prime responsibility for protection and safety.**..." In planned exposure situations, employers, registrants and licensees (hereinafter referred to simply as the 'management') are responsible for ensuring that protection and safety is optimized, that applicable dose limits are complied with, and that appropriate radiation protection programmes are established and implemented. Guidance on the content of the radiation protection programme is given in Section 3.

Compliance by workers

2.19. Requirement 22 of GSR Part 3 [2] states that "**Workers shall fulfil their obligations and carry out their duties for protection and safety.**" This

requirement reflects the fact that workers can, by their own actions, contribute to protection and safety for themselves and for others at work. The requirements on workers in this regard are listed in para. 3.83 of GSR Part 3 [2] and relate to: following rules and procedures; using monitoring equipment and personal protective equipment; cooperating in programmes for workers' health surveillance and programmes for dose assessment; and accepting instruction and training. Workers are also required to provide relevant information to the management and to act in a responsible manner with regard to protection and safety.

GRADED APPROACH

2.20. Paragraph 2.31 of GSR Part 3 [2] provides the basis for the graded approach to the control of exposure:

> "The regulatory body shall adopt a graded approach to the implementation of the system of protection and safety, such that the application of regulatory requirements is commensurate with the radiation risks associated with the exposure situation."

It is required as a general responsibility of the government to ensure that the overall application of the principles of radiation protection is in line with the graded approach (see para. 2.18 of GSR Part 3 [2]).

2.21. Requirement 6 of GSR Part 3 [2] refers to the graded approach in the more specific context of planned exposure situations:

> "**The application of the requirements of these Standards in planned exposure situations shall be commensurate with the characteristics of the practice or the source within a practice, and with the likelihood and magnitude of exposures.**"

2.22. An important feature of the graded approach in planned exposure situations is the provision for exemption and clearance. Requirement 8 of GSR Part 3 [2] states that:

> "**The government or the regulatory body shall determine which practices or sources within practices are to be exempted from some or all of the requirements of these Standards. The regulatory body shall**

approve which sources, including materials and objects, within notified practices or authorized practices may be cleared from regulatory control."

MANAGEMENT SYSTEM

2.23. Requirement 5 of GSR Part 3 [2] states that "**The principal parties shall ensure that protection and safety are effectively integrated into the overall management system of the organizations for which they are responsible.**" For occupational exposure in planned exposure situations, the principal party is the employer. For emergency exposure situations or existing exposure situations, the principal parties are those persons or organizations designated to deal with the situation.

2.24. In terms of paras 2.47–2.52 of GSR Part 3 [2], the "principal parties shall demonstrate commitment to protection and safety at the highest levels within the organizations" and "shall ensure that the management system…is designed and applied to enhance protection and safety" while maintaining coherence between measures for protection and safety and other measures, such as those addressing operational performance and security.

2.25. Specific actions should be taken to provide the necessary degree of confidence in the measures taken for achieving protection and safety and to ensure their regular assessment and review. A safety culture should be promoted and maintained at all levels within the organization. The management system should also address human factors by supporting good performance and good practices to prevent human and organizational failures, with attention being given to the design of equipment, the development of operating procedures, limits and conditions, as appropriate, training and the use of safety systems to mitigate consequences of human error.

2.26. More detailed requirements and guidance on the management system for facilities and activities are given IAEA Safety Standards Series Nos GSR Part 2, Leadership and Management for Safety [5], and GS-G-3.1, Application of the Management System for Facilities and Activities [6], and also in International Labour Organization guidelines [7]. Recommendations and guidance on the management system for providers of technical services in relation to protection and safety is given in Section 8.

DOSIMETRIC QUANTITIES

2.27. The dosimetric quantities recommended for radiation protection purposes (the protection quantities), and in which the dose limits are expressed in GSR Part 3 [2], are the equivalent dose H_T in tissue or organ T and the effective dose E.

2.28. The basic physical quantities include the particle fluence Φ, the kerma K and the absorbed dose D.

2.29. The determination of equivalent dose H_T in tissue or organ T involves the use of a radiation weighting factor w_R as a multiplier of absorbed dose for radiation type R to reflect the relative biological effectiveness (RBE) of the radiation in inducing stochastic effects at low doses:

$$H_T = \sum_R w_R \cdot D_{T,R} \tag{1}$$

where $D_{T,R}$ is the average absorbed dose in tissue or organ T for radiation type R.

2.30. The determination of effective dose E involves the use of a tissue weighting factor w_T as a multiplier of the equivalent dose for tissue T to account for the different sensitivities of different tissues and organs to the induction of stochastic effects:

$$E = \sum_T w_T \cdot H_T \tag{2}$$

which, on substituting for H_T from Eq. (1), gives:

$$E = \sum_T w_T \cdot \sum_R w_R \cdot D_{T,R} \tag{3}$$

2.31. The recommended values of w_R and w_T are based on a review of published biological and epidemiological studies, and are given in the definitions of terms in GSR Part 3 [2].

2.32. The protection quantities E and H_T relate to the sum of the effective doses or equivalent doses, respectively, received from exposure due to external sources within a given time period, and the committed effective doses or committed equivalent doses, respectively, from exposure due to intakes of radionuclides

occurring within the same time period. The total effective dose E received or committed during a given time period can be estimated from the operational quantities by using the following equation:

$$E \cong H_{\mathrm{p}}(d) + \sum_{j} e(g)_{j,\mathrm{ing}} \cdot I_{j,\mathrm{ing}} + \sum_{j} e(g)_{j,\mathrm{inh}} \cdot I_{j,\mathrm{inh}} \quad (4)$$

where

$H_{\mathrm{p}}(d)$ is the personal dose equivalent in soft tissue at an appropriate depth d below a specified point on the body during a given time period;

$e(g)_{j,\mathrm{ing}}$ is the committed effective dose per unit intake by ingestion for radionuclide j by the group of age g during the same time period;

$e(g)_{j,\mathrm{inh}}$ is the committed effective dose per unit intake by inhalation for radionuclide j by the group of age g during the same time period;

$I_{j,\mathrm{ing}}$ is the intake via ingestion of radionuclide j during the same time period;

and $I_{j,\mathrm{inh}}$ is the intake via inhalation of radionuclide j during the same time period.

For occupational exposure, the appropriate values of $e(g)_{j,\mathrm{ing}}$ and $e(g)_{j,\mathrm{inh}}$ are those for workers.

2.33. The dose limits are such that deterministic effects will not occur. For situations that could lead to severe deterministic effects (e.g. emergency exposure situations), the RBE of different types of radiation in causing severe deterministic effects should be considered. The recommended dosimetric quantity is the RBE weighted absorbed dose D_{T} in tissue or organ T. The determination of RBE weighted absorbed dose involves the use of tissue specific and radiation specific factors $\mathrm{RBE}_{\mathrm{T,R}}$ as multipliers of absorbed dose in a tissue or organ to take account of the RBE in causing the development of severe deterministic health effects from a given absorbed dose that is delivered in a tissue or organ by a given type of radiation. Recommended values of $\mathrm{RBE}_{\mathrm{T,R}}$ for the development of selected severe deterministic effects are based on a review of published biological studies and are given in the definitions of terms in GSR Part 3 [2]. The use of effective dose is inappropriate for the assessment of tissue reactions. In such situations, it is necessary to estimate absorbed dose and to take into account the appropriate RBE as the basis for any assessment of radiation effects.

Operational quantities for individual monitoring in external dosimetry

2.34. Since radiation protection quantities cannot be measured directly, the ICRU introduced operational quantities for practical use in radiation protection where

exposure due to external sources is concerned. Definitions of these quantities can be found in GSR Part 3 [2] and Ref. [8]. The operational quantities provide an estimate of effective or equivalent dose in a way that avoids underestimation and overestimation in most radiation fields encountered in practice. Radiation quality factors $Q(L)$ are used in calculating the operational dose equivalent quantities used in monitoring [3]. The quality factor characterizes the biological effectiveness of the radiation type based on the ionization density along the tracks of charged particles in tissue. Q is defined as a function of the unrestricted linear energy transfer L_∞ (often denoted as L or linear energy transfer, LET) of charged particles in water. A detailed evaluation of the numerical relationship between the physical quantities, radiation protection quantities and operational quantities was conducted by a joint task group of the ICRP and ICRU [9]. The conceptual relationship between those quantities is illustrated in Fig. 1 [9].

2.35. Strongly penetrating radiation and weakly penetrating radiation are differentiated by the ICRU [10]. 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 less than ten times larger than the effective dose, the radiation is said to be strongly penetrating. If the equivalent dose is more than ten times larger than the effective dose, the radiation is said to be weakly penetrating.

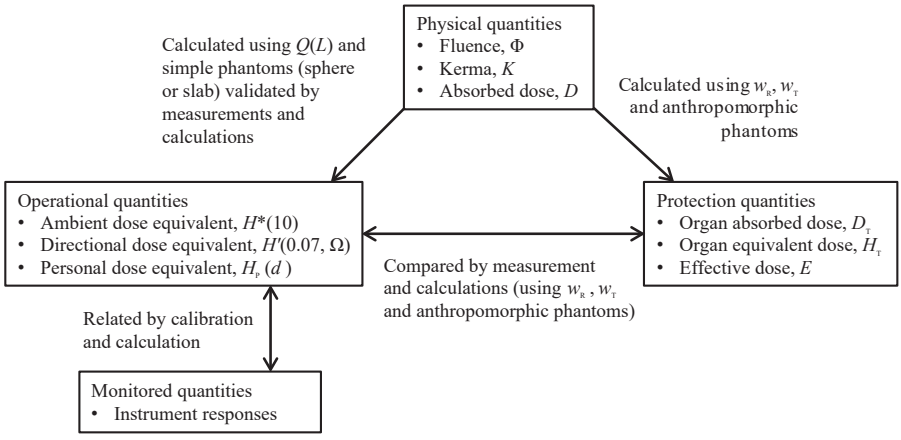


FIG. 1. Conceptual relationship between physical quantities, radiation protection quantities and operational quantities used for radiation protection purposes.

2.36. The operational quantity for individual monitoring is the personal dose equivalent $H_p(d)$. Any statement of personal dose equivalent should include a specification of the reference depth d. For strongly penetrating radiation, the reference depth is 10 mm. For weakly penetrating radiation, the reference depth is 0.07 mm. 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 are denoted by $H_p(10)$ and $H_p(0.07)$, respectively.

2.37. The sensitive cells of the skin for stochastic effects are considered to be between 0.02 mm and 0.1 mm below the skin surface, and $H_p(0.07)$ is therefore used to estimate the equivalent dose to small areas of the skin. A tissue thickness of 0.07 mm can be penetrated not only by photons but also by beta radiation with energy greater than 70 keV. For all types of radiation for which exposure of the extremities is of concern, the skin of the extremities is more likely to become the limiting tissue or organ, rather than the extremity itself. An estimation of the equivalent dose to the skin will be a conservative estimate of equivalent dose to the extremity. Thus, an extremity dosimeter essentially becomes a skin dosimeter and should be designed to measure $H_p(0.07)$.

2.38. For monitoring of the lens of the eye, a depth of 3 mm is recommended by the ICRU [8], so the operational quantity to be used is $H_p(3)$. In practice, however, the use of $H_p(3)$ has not yet been implemented for routine individual monitoring. In specific cases, when actual workplace radiation fields are known, monitoring of the lens of the eye using dosimeters calibrated for $H_p(0.07)$ or $H_p(10)$ could be acceptable. In Ref. [11], it is stated that $H_p(0.07)$ can be considered a good operational quantity for the lens of the eye for exposures to fields for which most of the dose is due to photons, including X rays. In such cases, it should be borne in mind that the uncertainty associated with the estimation of equivalent dose will be higher.

Quantities for workplace monitoring in external dosimetry

2.39. The operational quantities recommended for workplace monitoring are defined in a phantom known as the ICRU sphere [12]. This is a sphere, 30 cm in diameter, made of tissue equivalent material with 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.

2.40. The two quantities recommended by the ICRU for workplace monitoring [8] are the ambient dose equivalent $H^*(d)$ and the directional dose equivalent $H'(d, \Omega)$.

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

2.42. An 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.

2.43. Any statement of ambient dose equivalent should include a specification of the reference depth d. For strongly penetrating radiation, the recommended depth is 10 mm. The value of d should be expressed in millimetres, so the ambient dose equivalent for strongly penetrating radiation is $H^*(10)$. When measuring $H^*(10)$, the radiation field should be uniform over the sensitive volume of the instrument and the instrument should have an isotropic response.

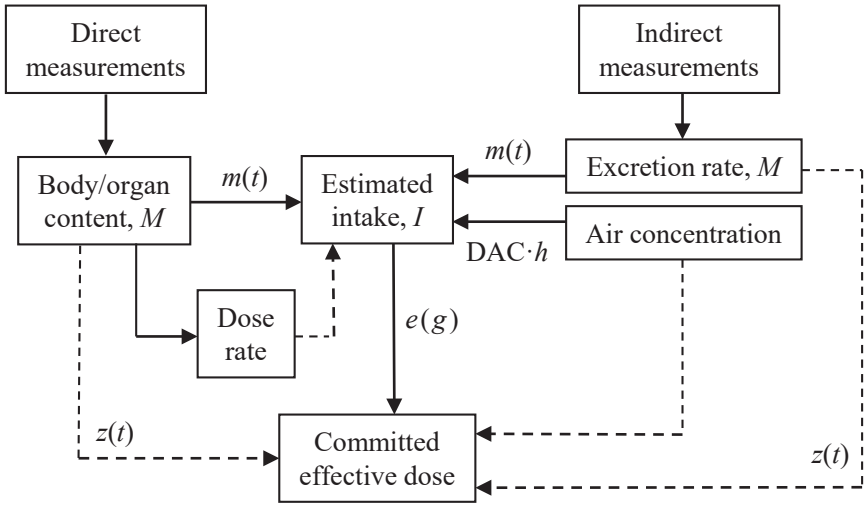
2.44. The directional dose equivalent $H'(d, \Omega)$ at a point in a radiation field is the dose equivalent that would be produced by the corresponding expanded field in the ICRU sphere at a depth d on a radius in a specified direction Ω. Any statement of directional dose equivalent should include a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

2.45. 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, $H'(d) = H^*(d)$. When measuring $H'(d, \Omega)$, the radiation field should be uniform over the dimensions of the instrument and the instrument should have the appropriate directional response.

2.46. For exposure of the lens of the eye, the recommended depth is 3 mm, but there are, at present, no published conversion coefficients for converting from the basic physical quantity kerma to the directional dose equivalent $H'(3, \Omega)$. However, if the monitoring device is not designed to measure $H'(3, \Omega)$, then $H'(0.07, \Omega)$ may be used as a surrogate [8].

Quantities for individual monitoring in internal dosimetry

2.47. Internal doses cannot be measured directly; they can only be inferred from individual measurements of other quantities, such as measurements of activity in the body or in excretion samples. In circumstances in which individual monitoring is inappropriate, inadequate or not feasible, the occupational exposure of workers may be assessed on the basis of workplace monitoring and other relevant information such as location and durations of exposure. Individual measurements include measurements made by both direct and indirect methods. Methods for the measurement of activity content in the body, such as whole body, thorax or thyroid counting, are examples of direct methods. In vitro measurements of activity in collected biological samples or measurements made using personal air sampling are examples of indirect methods. The conceptual framework for the assessment of doses from such measurements is illustrated in Fig. 2.



Note: Possible alternative approaches for calculation are indicated as dashed lines.

FIG. 2. General scheme for the assessment of internal doses from monitoring measurements [9].

2.48. As shown in Fig. 2, the quantity of primary interest for internal dose is the intake I (i.e. the activity of the radionuclide taken into the body). The value of the intake is obtained by dividing the measured body content or excretion rate M by the appropriate value of $m(t)$:

$$I = \frac{M}{m(t)} \tag{5}$$

where $m(t)$ is the fraction of an intake that remains in the body (for direct methods) or that is being excreted from the body (for indirect methods) at time t after the intake [13]. This fraction depends on the radionuclide, its chemical and physical form, the route of intake and the time t.

2.49. In the case of an intake of a mixture of radionuclides or of repeated intakes, the intake I_j of radionuclide j can be calculated by using the relevant measurement M_j and the derived fraction $m(t)_j$.

2.50. The doses expected to result from a given intake I are called the committed equivalent dose $H_T(\tau)$ to tissue or organ T and the committed effective dose $E(\tau)$, where τ is the time period after the intake over which the dose is integrated. The committed effective dose $E(\tau)$ is normally used for routine evaluation of occupational exposure. For occupational exposure of adults, τ is taken to be 50 years, irrespective of the age at intake. For occupational exposure of apprentices between the ages of 16 and 18 years, and for exposure of students between the ages of 16 and 18 years, τ is the time to the age of 70 years.

2.51. To derive the value of committed equivalent dose to a tissue or organ, the intake is multiplied by $h_T(g)$, the committed equivalent dose per unit intake for ingestion or inhalation, as appropriate, by the group of age g. For routine evaluation of occupational exposure, the adult group of age is considered, except for apprentices.

2.52. To derive the value of the committed effective dose, the intake is multiplied by $e(g)$, the committed effective dose per unit intake for ingestion or inhalation, as appropriate, by the group of age g.

2.53. In the case of an intake of a mixture of radionuclides, the intake of each radionuclide should be assessed separately and multiplied by the applicable dose coefficient (i.e. by the applicable committed effective dose per unit intake).

2.54. The committed dose can be seriously underestimated if the dose coefficient $h_T(g)$ or $e(g)$ is applied directly to the measured body content rather than to the inferred intake.

2.55. Various biokinetic models for calculating the values of $m(t)$ and $e(g)$ have been developed (see para. 7.141(a)). Values of $m(t)$ at selected times for a subset of

radionuclides have been reported by the ICRP in graphical and tabular form (see Ref. [13]). A compilation of dose coefficients $e(g)$ for intakes of radionuclides by workers is presented in Ref. [14] and can also be found in table III.2A of GSR Part 3 [2]. These dose coefficients are based on the calculation methods and parameters given in Ref. [15]. The current published values of $m(t)$ and $e(g)$ have been superseded by new values in Ref. [16] based on updated biokinetic models and on the methods of calculation and the parameters given in Ref. [3].

2.56. The ICRP has provided dose coefficients per unit body content $z(t)$ [16]. As illustrated in Fig. 2, these coefficients will enable the committed effective dose to be calculated directly from the results of monitoring measurements, without going through the process of calculating the corresponding intake, according to the equation:

$$E(\tau) = M \cdot z(t) \tag{6}$$

2.57. In situations of exposure due to a single radionuclide by inhalation or ingestion, with no external exposure, the limit on intake I_{L} corresponding to the limit L on effective dose is given by:

$$I_{\mathrm{L}} = \frac{\mathrm{L}}{e(g)} \tag{7}$$

where $e(g)$ is the applicable value of the committed effective dose per unit intake. When there is internal exposure due to a range of radionuclides or external exposure, the total effective dose should be calculated by summation of the individual contributions and should be compared with the relevant limit on effective dose.

2.58. The potential for inhalation of radionuclides may be assessed when necessary by measuring activity concentrations in air samples. The derived air concentration is defined as the concentration of airborne activity which would result in the intake $I_{\mathrm{inh,\,L}}$ by a worker exposed continuously for one year (taken to be 2000 h). The derived air concentration is usually expressed in units of becquerels per cubic metre. For a standard breathing rate of 1.2 m^3/h and for an intake expressed in becquerels, the derived air concentration is thus given by:

$$\mathrm{DAC} = \frac{I_{\mathrm{inh,\,L}}}{2000 \times 1.2} \tag{8}$$

2.59. The measured airborne activity concentration, expressed as a fraction of the derived air concentration, can be multiplied by the exposure time in hours to obtain an estimate of intake expressed in units of DAC hours.

Quantities for monitoring short lived progeny of radon (^{222}Rn)

2.60. The dose to the lung arises almost entirely from the short lived progeny of ^{222}Rn, rather than from ^{222}Rn itself (see para. 5.45). The short lived progeny are unlikely to be in equilibrium with the parent radionuclide. Therefore, for purposes of radiation protection, special quantities are used for expressing the concentration of ^{222}Rn progeny in air and the resulting exposure due to inhalation.

Potential alpha energy

2.61. The potential alpha energy ε_p of a single atom of a short lived ^{222}Rn progeny radionuclide is the total alpha energy emitted by that atom during complete decay from ^{222}Rn to ^{210}Pb.

2.62. The potential alpha energy emitted by 1 Bq of a radionuclide, rather than by a single atom, is given by:

$$\text{Potential alpha energy per unit activity (J/Bq)} = \frac{\varepsilon_p}{\text{activity per atom}} = \frac{\varepsilon_p}{\lambda} = \frac{\varepsilon_p t}{\ln 2} \quad (9)$$

where λ is the decay constant (in reciprocal seconds) and t is the half-life of the radionuclide (in seconds). The relevant values for the short lived progeny of ^{222}Rn are given in Table 1.

Potential alpha energy concentration

2.63. When considering exposure situations involving ^{222}Rn progeny, it is usual to express the total potential alpha energy as an energy concentration in air (in joules per cubic metre). This is referred to as the potential alpha energy concentration. For any mixture of short lived ^{222}Rn progeny in air, the contribution of each radionuclide to the potential alpha energy concentration is its potential alpha energy per unit activity (ε_p/λ) as given in Table 1 multiplied by its activity concentration c. The total potential alpha energy concentration is then the sum of these individual contributions:

$$\text{PAEC} = \sum_j c_j \frac{\varepsilon_{p,j}}{\lambda_j} \quad (10)$$

TABLE 1. POTENTIAL ALPHA ENERGIES OF SHORT LIVED ^{222}Rn PROGENY

Radionuclide	Half-life	Alpha energy (J)	Yield (%)	Potential alpha energy	
				Per atom ε_p (J)	Per unit activity ε_p/λ (J/Bq)
Po-218	3.098 min	0.962×10^{-12}	100	2.19×10^{-12}	0.588×10^{-9}
Pb-214	26.8 min	Nil (beta emitter)	—	1.23×10^{-12}	2.85×10^{-9}
Bi-214	19.9 min	Nil (beta emitter)	—	1.23×10^{-12}	2.12×10^{-9}
Po-214	164.3 μs	1.23×10^{-12}	100	1.23×10^{-12}	3×10^{-16}

Source: 2014 data from the NuDat Database (see www.nndc.bnl.gov/nudat2).

2.64. It can be deduced from Table 1 (simply by adding the values in the right hand column) that if all the progeny were to be in equilibrium with the parent ^{222}Rn at a concentration of 1 Bq/m^3, the potential alpha energy concentration of the mixture would be 5.56×10^{-9} J/m^3.

2.65. In practice, the progeny will rarely, if ever, be in equilibrium, and the potential alpha energy concentration will, therefore, be some fraction of the equilibrium value. This fraction is called the equilibrium factor F:

$$F = \frac{\text{PAEC}}{\text{PAEC (equilibrium)}} \tag{11}$$

2.66. By way of example, consider a non-equilibrium mixture of ^{222}Rn and its progeny, in which the individual radionuclide activity concentrations are 100 Bq/m^3 for ^{222}Rn, 75 Bq/m^3 for ^{218}Po, 50 Bq/m^3 for ^{214}Pb and 25 Bq/m^3 for each of ^{214}Po and ^{214}Bi. From Table 1, the potential alpha energy concentration of the mixture is:

$$\begin{aligned}\text{PAEC} &= \left(0.588\times10^{-9}\times75\right)+\left(2.85\times10^{-9}\times50\right)+\left(2.12\times10^{-9}\times25\right)+\left(3\times10^{-16}\times25\right)\\ &= 2.40\times10^{-7}\ \text{J/m}^3\end{aligned} \tag{12}$$

2.67. If the mixture were in equilibrium, all radioisotopes of the decay series would have an activity concentration of 100 Bq/m^3 and the potential alpha energy concentration, in accordance with para. 2.64, would be:

$$\text{PAEC (equilibrium)} = 5.56\times10^{-9}\times100 = 5.56\times10^{-7}\ \text{J/m}^3 \tag{13}$$

The equilibrium factor of the mixture is therefore:

$$F = \frac{2.40\times10^{-7}}{5.56\times10^{-7}} = 0.432 \tag{14}$$

Potential alpha energy exposure[7]

2.68. The exposure of an individual to ^{222}Rn progeny (P_{RnP}) is determined by multiplying the potential alpha energy concentration (in joules per cubic metre) by the exposure period (in hours). The exposure is therefore expressed in units of joule hours per cubic metre. Since the potential alpha energy concentration will generally vary during the exposure period, the exposure should be calculated as an integral over time:

$$P_{\text{RnP}} = \int_0^{\tau} \text{PAEC}(t)\,\text{d}t \tag{15}$$

where τ is the period of exposure. The exposure period is usually calculated over the course of one year. It is common to adopt a default annual exposure period of 2000 h for workplaces. It should be borne in mind that the adoption of this default value may lead to a conservative estimate of the annual exposure.

Equilibrium equivalent concentration and equilibrium equivalent exposure

2.69. There is an alternative way of referring to the concentration of ^{222}Rn progeny in air. If the ^{222}Rn gas concentration (in becquerels per cubic metre) is multiplied by the equilibrium factor F, the resulting quantity is called the equilibrium equivalent concentration (EEC) of the ^{222}Rn parent (also expressed in units of becquerels per cubic metre). The EEC can be regarded as the concentration of ^{222}Rn in equilibrium with its progeny that would give the same potential alpha energy concentration as the actual non-equilibrium mixture. It can be determined

[7] Potential alpha energy exposure is not a type of potential exposure.

from para. 2.64 that the numerical relationship between the potential alpha energy concentration and the EEC is as follows:

$$\text{PAEC (J/m}^3) = 5.56 \times 10^{-9} \times \text{EEC (Bq/m}^3) \quad (16)$$

In the same way, exposure due to ^{222}Rn progeny can be expressed as the equilibrium equivalent exposure, in units of becquerel hours per cubic metre:

$$\text{Equilibrium equivalent exposure} = \int_0^{\tau} \text{EEC}(t)\,\mathrm{d}t \quad (17)$$

The choice between potential alpha energy exposure and equilibrium equivalent exposure is not important, since these two quantities are simply related by a constant factor of 5.56×10^{-9} J/Bq.

^{222}Radon gas concentration as a surrogate for exposure due to ^{222}Rn progeny

2.70. In many situations involving exposure due to ^{222}Rn progeny, the measurement process can be simplified considerably by using the time weighted average ^{222}Rn gas concentration in air (in units of becquerels per cubic metre) as a surrogate for potential alpha energy. For instance, measurements in a large number of buildings over an extended time period are best made using passive track etch devices that detect ^{222}Rn. Such devices are small, simple, robust and inexpensive. When adopting this approach, an appropriate value for the equilibrium factor F should be assumed. The use of a default value of 0.4 is usually adequate for this purpose. It has been found that most values of F in indoor air are within 30% of this value. However, workplaces such as underground mines or water treatment facilities can have significantly lower F values. The potential alpha energy exposure is then given by:

$$\text{Potential alpha energy exposure (J} \cdot \text{h} \cdot \text{m}^{-3}) = {}^{222}\text{Rn concentration} \times 5.56 \times 10^{-9} \times 0.4 \times T \quad (18)$$

where T is the exposure period (h). By using a default annual exposure period of 2000 h for workplaces, this formula gives a potential alpha energy exposure of 4.45×10^{-6} J·h·m^{-3} for a ^{222}Rn concentration of 1 Bq/m^3.

Quantities for monitoring short lived progeny of ^{220}Rn

2.71. Radon-220 (commonly referred to as thoron) is not normally of concern in workplaces, except where material with a high thorium content is processed or stored, for example in the processing of monazite to extract rare earths and

thorium. In such instances, a similar approach to that for ^{222}Rn progeny can be followed. The short lived progeny of ^{220}Rn are likely to be out of equilibrium with the parent. In enclosed workplaces, the short half-life of ^{220}Rn (55.6 s) means that the spatial distribution of ^{220}Rn is very different from that of its progeny. The assessment of an equilibrium factor is difficult and, for dose assessment purposes, an approach based on the measurement of the concentration of ^{220}Rn progeny is easier and more appropriate than an approach based on measurement of the concentration of ^{220}Rn.

2.72. Of the various ^{220}Rn progeny radionuclides, only ^{212}Pb and ^{212}Bi make major contributions — 91% and 9%, respectively — to the total potential alpha energy. The potential alpha energy of ^{212}Pb is 6.91×10^{-8} J/Bq, while that of ^{212}Bi is 6.56×10^{-9} J/Bq. The contribution of the parent radionuclide ^{220}Rn is more than an order of magnitude lower than that of ^{212}Bi. Since ^{212}Pb contributes almost all of the total potential alpha energy, its activity concentration in air can be used as a surrogate for potential alpha energy concentration, in which case a ^{212}Pb concentration of 1 Bq/m^3 corresponds to a potential alpha energy concentration of 6.91×10^{-8} J/m^3.

3. EXPOSURE OF WORKERS IN PLANNED EXPOSURE SITUATIONS

3.1. Paragraphs 3.1–3.4 of GSR Part 3 [2] specify the scope of application of the requirements for planned exposure situations. The scope is defined in terms of the practices involved and the exposures to sources within practices. With regard to exposure to radiation from natural sources, para. 3.4 of GSR Part 3 [2] states that such exposure is normally subject to the requirements for existing exposure situations (see Section 5). Only in certain cases do the requirements for planned exposure situations apply (see paras 3.159 and 3.161).

3.2. GSR Part 3 [2] requires any person or organization intending to carry out any activity within the scope of application of the requirements to submit a notification to the regulatory body of such an intention. Notification alone is sufficient provided that the exposures expected to be associated with the practice or action are unlikely to exceed a small fraction of the relevant limits, as specified by the regulatory body, and that the likelihood and magnitude of potential exposures and any other potential detrimental consequences are negligible.

3.3. Where notification alone is not sufficient, the person or organization concerned is required to apply to the regulatory body for authorization, which takes the form of registration or licensing. Typical practices that are amenable to registration are those for which: (i) safety can largely be ensured by the design of the facilities and equipment; (ii) the operating procedures are simple to follow; (iii) the safety training requirements are minimal; and (iv) there is a history of few problems with safety in operations. Registration is best suited to those practices for which operations do not vary significantly.

3.4. One of the primary responsibilities of management with regard to occupational exposure is set out in Requirement 21 of GSR Part 3 [2]:

> "**Employers, registrants and licensees shall be responsible for the protection of workers against occupational exposure. Employers, registrants and licensees shall ensure that protection and safety is optimized and that the dose limits for occupational exposure are not exceeded.**"

3.5. In terms of para. 3.78 of GSR Part 3 [2], where a worker's exposure arises from sources that are not required by, or directly related to, the work, the management is required to provide that worker with the same level of protection against such exposure as members of the public.

3.6. In accordance with the graded approach to regulation (see paras 2.20–2.22), the government or the regulatory body is required to determine which practices or sources within practices are to be exempted from some or all of the requirements of GSR Part 3 [2], including the requirements for notification, registration or licensing (see para. 3.10 of GSR Part 3 [2]). Similarly, the regulatory body is required to approve which sources, including materials and objects, that are already within a notified practice or an authorized practice may be cleared from regulatory control (see para. 3.12 of GSR Part 3 [2]). Exemption or clearance is the appropriate regulatory option if the radiation risks are too low to warrant regulatory control or if the imposition (or retention) of regulatory control would yield no net benefit (see paras I.1–10 of GSR Part 3 [2]).

3.7. In terms of paras I.2 and I.11 of GSR Part 3 [2], the general criterion for exemption or clearance without further consideration is an effective dose of the order of 10 μSv or less in a year (or 1 mSv or less in a year in the case of low probability scenarios). However, for bulk material containing radionuclides of natural origin, the 10 μSv criterion is not appropriate, since it is one or two orders of magnitude below the normal variations in exposure to natural background

radiation. For such material, the criterion for exemption is an effective dose of the order of 1 mSv or less in a year (para. I.4 of GSR Part 3 [2]), while the criterion for clearance is an activity concentration of 1 Bq/g or less for each radionuclide in the ^{238}U decay series and the ^{232}Th decay series, and 10 Bq/g or less for ^{40}K (or, for certain residues, an effective dose of 1 mSv or less in a year) (see para. I.12 of GSR Part 3 [2]).

OPTIMIZATION

General

3.8. Paragraphs 3.76(b) and 3.77 of GSR Part 3 [2] state that:

> "3.76. Employers, registrants and licensees shall ensure, for all workers engaged in activities in which they are or could be subject to occupational exposure, that:
>
> …….
>
> (b) Protection and safety is optimized in accordance with the requirements of these Standards;
>
> …….
>
> "3.77. Employers, registrants and licensees:
>
> (a) Shall involve workers, through their representatives where appropriate, in optimization of protection and safety;
> (b) Shall establish and use, as appropriate, constraints as part of optimization of protection and safety."

3.9. For control of occupational exposure in planned exposure situations, guidance on meeting the relevant requirements of GSR Part 3 [2] for optimization of protection and safety is provided in paras 3.10–3.18. Further information of a more practical nature is provided in Ref. [17].

3.10. Optimization of protection and safety should be considered at all stages in the lifetime of equipment and installations, in relation to both exposures from normal operations and potential exposures. As a consequence, all situations — from design through operation to decommissioning and waste management — should be considered in the optimization procedure.

3.11. From a practical viewpoint, the requirements for optimization call for an approach that:

(a) Considers all possible actions involving the source(s) and the way workers operate with or near the source(s).
(b) Implies a 'management by objective' process with the following sequence: planning, setting objectives, monitoring, measuring performance, evaluating and analysing performance to define corrective actions, and setting new objectives.
(c) Can be adapted to take into account any significant change in the state of techniques, the resources available for the purposes of protection or the prevailing societal context.
(d) Encourages accountability, such that all parties adopt a responsible attitude to the process of eliminating unnecessary exposures.

3.12. The quantity collective effective dose can be used as an instrument for optimization, for comparing available radiological technologies and for protection procedures. This quantity takes account of the exposure of all individuals in a group over a given time period or during a given operation executed by this group in designated radiation areas. The collective effective dose is calculated as the sum of all individual effective doses over the time period or during the operation being considered and is expressed in man-sieverts (man Sv).

3.13. The process of optimization should take account of:

(a) The resources available for protection and safety;
(b) The distributions of individual exposure and collective exposure in different groups of workers;
(c) The probability and magnitude of potential exposure;
(d) The potential impact of actions for the purposes of radiation protection on the level of other (non-radiological) risks to workers or to members of the public;
(e) Good practices in relevant sectors;
(f) Societal and economic aspects.

3.14. Some of the options considered in the optimization of protection and safety for workers may lead to increased exposure of other persons or, in the medical field, a reduction in the efficacy of the clinical procedure. Such impacts should be taken into account in the optimization process, especially when considering the establishment of administrative controls and the use of personal protective equipment. In particular, the arrangements for the protection of medical staff

should not lead to a reduction in the protection of the patient or a deterioration in the clinical outcome.

3.15. In general, incremental benefits to be obtained in terms of dose reduction decrease progressively as the associated expenditure increases. Even the cost of considering the ways in which doses may be reduced can become significant compared with the benefit to be achieved. At some stage, the effort might not be worthwhile for low doses. In this context, it is noted that para. 3.10 of GSR Part 3 [2] provides for the exemption of practices from regulatory control when an assessment shows that exemption is the optimum option for protection. This provision is simply a recognition of the more general concept of diminishing returns.

3.16. The optimization of protection and safety should be considered at the design stage of equipment and installations, when some degree of flexibility is still available. The use of engineered controls should be examined carefully at this stage in defining the protection options. In image guided interventional procedures, for example, where there is a potential for workers to receive a significant dose to the lens of the eye, attention should be paid to the installation of fixed shielding and to the selection of equipment. Even if protection has been optimized at the design stage, however, the requirements for optimization in the operational phase still apply. At this stage, the content and the scale of the optimization process will depend on the situation. For example, when dealing with X ray machines, the optimization process can be quite straightforward, involving local rules and appropriate training of the operators. For nuclear facilities, situations are more complicated, and a structured approach should be followed as part of the radiation protection programme, including the use of decision aiding techniques (see paras 3.24–3.27), the establishment of dose constraints (see paras 3.28–3.33) and the establishment of investigation levels (see paras 3.122–3.128).

3.17. Optimization of protection and safety in operation is a process that begins at the planning stage and continues through the stages of scheduling, preparation, implementation and feedback. This process of optimization through work management is applied in order to keep exposure levels under review and to ensure that they are as low as reasonably achievable. The elaboration of a radiation protection programme, adapted to the specific situation, is an essential element of work management.

3.18. The management should record information on the way in which optimization of protection and safety is being applied and should disseminate the information, as appropriate. This information could cover the following:

(a) The rationale for proposed operating, maintenance and administrative procedures, together with other options that have been considered and the reasons for their rejection;
(b) Periodic review and trend analysis for doses due to occupational exposure of individuals in various work groups, and other performance indicators;
(c) Internal audits and peer reviews, and the resulting corrective actions;
(d) Incident reports and lessons to be learned.

Commitment to optimization of protection

3.19. The primary responsibility for optimization of protection and safety lies with the management. Commitment to an effective protection and safety policy is essential at all levels of the management, but in particular at the senior level. The commitment of the management should be demonstrated by means of written policy statements that make radiation protection criteria an integral part of the decision process, and by provision of adequate resources and clear and demonstrable support for those persons with direct responsibility for radiation protection in the workplace.

3.20. The senior management should translate its commitment to optimization of protection and safety into effective action by incorporating optimization into an appropriate radiation protection programme, commensurate with the level and nature of the radiation risks presented by the practice. The scope of such a programme is set out in para. 3.60.

3.21. Workers should also have a commitment to protection and safety. The employer should ensure that mechanisms are in place by which workers can be involved, as much as possible, in the development of methods to keep doses as low as reasonably achievable, and have the opportunity to provide their views on the effectiveness of radiation protection measures.

3.22. Optimization of protection and safety is a regulatory requirement. The regulatory body should be committed to optimization of protection and safety, and should encourage its application. Where necessary, the regulatory body should undertake all relevant actions to enforce regulatory requirements on the management to apply this principle.

3.23. The management should ensure that training programmes, with content and duration commensurate with, and adapted to, the functions and responsibilities of the staff concerned, are provided for staff at all levels, including the senior management. The staff of regulatory bodies should have the training necessary to ensure that optimization of protection and safety is appropriately applied and enforced.

Use of decision aiding techniques

3.24. The process of optimization of protection and safety using decision aiding techniques can range from intuitive qualitative analyses to quantitative analyses, but should be sufficient to take all relevant factors into account in a coherent way, so as to contribute to achieving the following objectives:

(a) To determine optimized measures for protection and safety for the prevailing circumstances, with account taken of the available options for protection and safety as well as the nature, magnitude and likelihood of exposures;
(b) To establish criteria, on the basis of the results of the optimization process, for restriction of the magnitudes of exposures and of their probabilities by means of measures for preventing accidents and mitigating their consequences.

3.25. In most situations, a qualitative approach based on professional judgement will be sufficient for deciding upon the most favourable level of protection that can be achieved. In more complex situations, particularly those having significant implications for expenditure (e.g. at the design stage of installations), the use of a more structured approach may be appropriate. Some complex situations may be quantifiable using cost–benefit analysis or other quantitative techniques. In other cases, however, it may not be possible to quantify all of the factors involved, or to express them in commensurate units. It may also be difficult to make a balance between collective doses and individual doses, and between doses to workers and doses to the public, and to take account of broader societal factors. For these situations, qualitative decision aiding techniques such as multicriteria analysis can be useful.

3.26. A structured approach to the selection of appropriate measures for protection and safety should include the following steps, with account taken of exposures from normal operations and of potential exposures:

(a) Identify all practicable protection options that might potentially reduce the occupational exposure;
(b) Identify all relevant economic, societal, radiological and, where appropriate, non-radiological factors for the particular situation under review that distinguish between the identified options (e.g. collective dose, distribution of individual dose, impact on public exposure, impact on future generations and investment costs);
(c) Quantify, where possible, the relevant factors for each protection option;
(d) Compare all options and select the optimum option(s);
(e) Where appropriate, perform a sensitivity analysis (i.e. evaluate the robustness of the solutions obtained by testing using different values for the key parameters for which recognized uncertainties exist).

3.27. Whatever the situation, decision makers should keep in mind that decision aiding techniques do not necessarily provide the definitive answer, nor do they provide the only possible solution. These techniques should be seen as tools to help structure problems in order to compare the relative effectiveness of various possible options for protection and safety, to facilitate the integration of all relevant factors and to assist in taking coherent decisions.

Dose constraints

3.28. Paragraph 1.22 of GSR Part 3 [2] states that:

> "Dose constraints…are used for optimization of protection and safety, the intended outcome of which is that all exposures are controlled to levels that are as low as reasonably achievable, economic, societal and environmental factors being taken into account. Dose constraints are applied to occupational exposure and to public exposure in planned exposure situations."

For occupational exposures, a dose constraint is a source related value of individual dose used to limit the range of options considered in the process of optimization, and it will always be a fraction of the dose limit. Paragraph 1.22 continues that:

> "Dose constraints are set separately for each source under control and they serve as boundary conditions in defining the range of options for the purposes of optimization of protection and safety. Dose constraints are not dose limits: exceeding a dose constraint does not represent non-compliance with regulatory requirements, but it could result in follow-up actions."

3.29. Paragraph 1.23 of GSR Part 3 [2] states that:

> "While the objectives of the use of dose constraints for controlling occupational exposure and public exposure are similar, the dose constraints are applied in different ways. For occupational exposure, the dose constraint is a tool to be established and used in the optimization of protection and safety by the person or organization responsible for a facility or an activity.... After exposures have occurred, the dose constraint may be used as a benchmark for assessing the suitability of the optimized strategy for protection and safety...that has been implemented and for making adjustments as necessary. The setting of the dose constraint needs to be considered in conjunction with other health and safety provisions and the technology available."

3.30. The objective of applying a dose constraint is to place a ceiling on values of individual dose — doses from a source, a set of sources in an installation, a practice, a task or a group of operations in a specific type of industry — that could be considered acceptable in the process of optimization of protection for those sources, practices or tasks. Depending on the situation, the dose constraint can be expressed as a single dose or as a dose over a given time period. The setting of any dose constraints should be such that dose limits for occupational exposure are complied with when workers incur exposures from multiple sources or tasks.

3.31. To apply the optimization principle, individual doses should be assessed at the design and planning stage, and it is these predicted individual doses for the various options that should be compared with the appropriate dose constraint. Options predicted to give doses below the dose constraint should be considered further; those predicted to give doses above the dose constraint should normally be rejected. Dose constraints should not be used retrospectively to check compliance with the requirements for protection and safety.

3.32. Dose constraints should be used prospectively in optimizing radiation protection in various situations encountered in planning and executing tasks, and in designing facilities or equipment. They should, therefore, be set on a case by case basis in accordance with the specific characteristics of the exposure situation. Since dose constraints are source related, the source to which they relate should be specified. Dose constraints should be set in consultation with those involved. Regulatory bodies may use them in a generic way — for categories of similar sources, practices or tasks — or specifically, in authorizing individual sources, practices or tasks. The establishment of constraints may be the result of interaction

between the regulatory body, the affected operators and, where appropriate, workers' representatives. As a general rule, it would be more appropriate for the regulatory body to encourage the development of constraints for occupational exposure within particular industries and organizational groupings, subject to regulatory oversight, than to stipulate specific values of constraints.

3.33. The process of deriving a dose constraint for any specific situation should include a review of operating experience and feedback from similar situations, if possible, and considerations of economic, societal and technical factors. For occupational exposure, experience with well managed operations is of particular importance and should be considered in setting constraints. National surveys or international databases that capture a large amount of experience with exposures relating to specific operations can be useful for such purposes.

DOSE LIMITATION

3.34. Paragraph 3.76(a) of GSR Part 3 [2] states that:

> "Employers, registrants and licensees shall ensure, for all workers engaged in activities in which they are or could be subject to occupational exposure, that:
>
> (a) Occupational exposure is controlled so that the relevant dose limits for occupational exposure specified in Schedule III are not exceeded".

3.35. In accordance with para. III.1 of GSR Part 3 [2]:

> "For occupational exposure of workers over the age of 18 years, the dose limits are:
>
> (a) An effective dose of 20 mSv per year averaged over five consecutive years[66] (100 mSv in 5 years) and of 50 mSv in any single year;
> (b) An equivalent dose to the lens of the eye of 20 mSv per year averaged over five consecutive years (100 mSv in 5 years) and of 50 mSv in any single year;
> (c) An equivalent dose to the extremities (hands and feet) or to the skin[67] of 500 mSv in a year.
>
> "[66] The start of the averaging period shall be coincident with the first day of the relevant annual period after the date of entry into force of these Standards, with no retrospective averaging.

> "[67] The equivalent dose limits for the skin apply to the average dose over 1 cm^2 of the most highly irradiated area of the skin. The dose to the skin also contributes to the effective dose, this contribution being the average dose to the entire skin multiplied by the tissue weighting factor for the skin."

3.36. The regulatory body or other relevant authority should clearly define the convention to be followed in determining the periods to be used for dose limitation. Calendar years or national fiscal years are simple examples that can be used for the single year periods. 'Rolling' five year periods, in which the current single year (calendar or fiscal) is considered the final year in the five year period, can be selected for averaging purposes. Alternative conventions may be adopted to accord with regulatory preferences.

3.37. As stated in para. III.1 of GSR Part 3 [2], the limits on equivalent dose to the skin apply to the average dose over 1 cm^2 of the most highly irradiated area of the skin. The dose to the skin also contributes to the effective dose, this contribution being the average dose to the entire skin multiplied by the tissue weighting factor for the skin.

3.38. As stated in paras III.1 and 3.114 of GSR Part 3 [2], additional restrictions apply to occupational exposure for a female worker who has notified pregnancy or who is breast-feeding (see paras 3.46 and 6.2–6.20).

3.39. In accordance with para. III.2 of GSR Part 3 [2]:

> "For occupational exposure of apprentices of 16 to 18 years of age who are being trained for employment involving radiation and for exposure of students of age 16 to 18 who use sources in the course of their studies, the dose limits are:
>
> (a) An effective dose of 6 mSv in a year;
> (b) An equivalent dose to the lens of the eye of 20 mSv in a year;
> (c) An equivalent dose to the extremities (hands and feet) or to the skin[67] of 150 mSv in a year.
>
> "[67] The equivalent dose limits for the skin apply to the average dose over 1 cm^2 of the most highly irradiated area of the skin. The dose to the skin also contributes to the effective dose, this contribution being the average dose to the entire skin multiplied by the tissue weighting factor for the skin."

For occupational exposure, the employer of the apprentice is required to be responsible for protection and safety for the apprentice.

3.40. Recommendations on the application of dose limits to itinerant workers are provided in paras 6.21–6.100.

3.41. Cases in which the flexibility provided by the averaging of doses over five years might be needed include planned maintenance operations in nuclear power plants and routine work in some uranium mining operations. In most situations, however, provided that the principle of optimization of protection has been appropriately applied, it would be unusual for workers to receive an annual effective dose that exceeds 20 mSv. Where the flexibility provided by averaging is not needed, the regulatory body could continue to operate with an annual limit; the dose limit would then be 20 mSv in any single year.

3.42. The general approach to the application of the dose limits where full flexibility is used (i.e. averaging of doses over five years) can be summarized as follows:

(a) In general, the exposure of an individual worker should be controlled so that the effective dose does not exceed 20 mSv in a year. This includes the external dose as well as the internal dose received by the worker during the period.
(b) Where the exposure of an individual worker results in an effective dose exceeding 20 mSv in a year but within the dose limit of 50 mSv, the management should do the following, as appropriate:
    (i) Carry out a review of exposure to determine whether exposures were as low as reasonably achievable and, where appropriate, to take the necessary corrective action;
    (ii) Consider ways to restrict further exposures of the individual worker to ensure that the effective dose over the chosen five year averaging period is less than 100 mSv;
    (iii) Notify the regulatory body of the magnitude of the dose and the circumstances leading to the exposure.

3.43. In para. 3.48 of GSR Part 3 [2], registrants and licensees are required to report immediately to the regulatory body any event in which a dose limit is exceeded. The management should therefore have a suitable reporting system in place. Such a system should also provide for the notification of those workers involved in an event in which the dose limit for occupational exposure is exceeded.

3.44. Incidents in which a worker was exposed such that the single year dose limit of 50 mSv was exceeded would be considered exceptional. In such exceptional circumstances, it would be appropriate for the worker to continue working with radiation provided that:

(a) The regulatory body, having due regard for the health of the worker, considers that there is no reason to prevent the worker's continuing work with radiation;
(b) The employer and the regulatory body, in consultation with the worker (through the worker's representatives, where appropriate), and with the occupational physician, where appropriate, agree on a temporary dose restriction and the period to which it applies.

3.45. A restriction based pro rata on the remaining period of time to which the dose limit relates might be appropriate, and further restrictions might have to be applied in order to keep within the dose limit of 100 mSv in five years.

3.46. In general, the dose limits for occupational exposure apply equally to male and female workers. However, because of the possible relevance of the greater sensitivity of the embryo or fetus or the breastfed infant to radiation, additional controls should be considered for pregnant and breast-feeding workers. Special requirements for the radiation protection of female workers during and after pregnancy are addressed in paras 6.2–6.20.

3.47. The regulatory body should ensure that systems are in place to prevent workers who have received a dose close to a relevant dose limit being deprived of their right to work. Situations might arise in which a worker has unintentionally received a dose that is close to the relevant dose limit, such that further exposures could result in that limit being exceeded. This situation should be treated in a similar manner to the situation in which a worker's dose exceeds a dose limit (see paras 3.44 and 3.45).

3.48. The management should plan work programmes so as to ensure, to the extent possible, that workers do not receive a dose corresponding to a significant proportion of the relevant dose limit in a short period of time, such that subsequent exposures might result in the annual dose limit being exceeded.

RADIATION PROTECTION PROGRAMME

Objectives

3.49. The general objective of the radiation protection programme is to fulfil the management's responsibility for protection and safety through the adoption of management structures, policies, procedures and organizational arrangements that are commensurate with the nature and extent of the risks. The radiation protection programme should cover all the main elements contributing to protection and safety. The radiation protection programme could relate to all phases of a practice or to the lifetime of a facility (i.e. from design through commissioning and operation or process control to decommissioning).

3.50. Radiation protection is only one element in ensuring the overall health and safety of workers. The radiation protection programme should be established and managed in close cooperation with those responsible for other areas of health and safety such as industrial hygiene, industrial safety and fire safety.

3.51. Paragraph 3.93 of GSR Part 3 [2] requires that:

> "Employers, registrants and licensees shall minimize the need to rely on administrative controls and personal protective equipment for protection and safety by providing well engineered controls and satisfactory working conditions, in accordance with the following hierarchy of preventive measures:
>
> (1) Engineered controls;
> (2) Administrative controls;
> (3) Personal protective equipment."

3.52. Although the radiation protection programme could include protection of both workers and the public, this Safety Guide focuses only on those aspects dealing with the protection of workers. In most practices, doses received by workers are well below the relevant dose limits in GSR Part 3 [2], and only a small fraction of the workforce will potentially be affected by the requirements for dose limitation. The requirements for optimization should be the principal impetus for the establishment and implementation of radiation protection programmes, including, in many cases, measures to prevent or reduce potential exposures and measures to mitigate the consequences of accidents.

Prior radiological evaluation and safety assessment

3.53. The characteristics of exposure situations may vary considerably depending on the type of facility concerned (ranging from 'simple' facilities, such as baggage inspection equipment in airports, to much more complex facilities, such as nuclear reprocessing plants), and on the stage of activity (e.g. construction, operation, maintenance and decommissioning). It should be ensured by the use of a graded approach that the radiation protection programme is well adapted to the situation (see paras 2.20–2.22). As the first step towards the definition of a radiation protection programme, a prior radiological evaluation of the facility or activity should be performed.

3.54. The prior radiological evaluation should describe, as precisely as necessary, the situation involving occupational exposures. In accordance with a graded approach, the level of effort, formality and detail of the evaluation, and the scrutiny to which it is subjected, should be linked to the magnitude of the exposures in normal operation, and to the magnitude and probability of potential exposures.

3.55. The prior radiological evaluation should identify the following for all aspects of operations:

(a) The sources of routine exposure and reasonably foreseeable potential exposure, such as surface contamination, airborne contamination and sources of external radiation.
(b) The nature and magnitude of exposures in normal operations.
(c) The nature, likelihood and magnitude of potential exposures. This should include the ways in which structures, systems and components, and procedures relating to radiation protection or safety, might fail, singly or in combination, or otherwise lead to potential exposures, and the consequences of such failures.
(d) The measures for protection and safety that are necessary to implement the optimization process.
(e) Appropriate monitoring systems.
(f) An assessment of potential public exposure due to radioactive effluents from the facility.

3.56. The assessment of exposures in the prior radiological evaluation may be made by one or more of the following methods:

(a) Use of workplace monitoring. This method can give a good assessment of the doses that workers will receive, provided that the radiological conditions in the workplace are reasonably predictable over a long period (at least for several months). Workplace monitoring should be repeated at appropriate intervals, and certainly when the working conditions change significantly.
(b) Use of data from the scientific literature and information from comparable facilities. Some dose values are given in the literature for various workplace situations. These can, in principle, be used to judge whether monitoring is needed.
(c) Use of simulations. Numerical simulations can be powerful and can provide information instantly on the parameters that influence doses that would be received in given exposure situations. The results of simulations should be verified by measurement.
(d) Use of confirmatory measurements. Performing confirmatory measurements with personal dosimeters can help to determine whether individual monitoring is needed.

3.57. The prior radiological evaluation will help to determine what can be achieved at the design stage to establish satisfactory working conditions through the use of engineered features. Examples would be the provision of shielding, containment, ventilation or interlocks. These considerations should be aimed at minimizing the need for relying on administrative controls and personal protective equipment for protection and safety during normal operations (see para. 3.51). Consideration may then be given subsequently to additional operational procedures and restrictions that might be implemented to further control workers' exposure. Only if these measures are not sufficient to adequately restrict the doses received by workers will the prior evaluation need to include consideration of the use of special tools, personal protective equipment and specific task related training.

3.58. With respect to the safety assessment process, Requirement 13 of GSR Part 3 [2] states that:

> "**The regulatory body shall establish and enforce requirements for safety assessment, and the person or organization responsible for a facility or activity that gives rise to radiation risks shall conduct an appropriate safety assessment of this facility or activity.**"

3.59. Paragraph 3.31 of GSR Part 3 [2] requires that:

> "Safety assessments shall be conducted at different stages, including the stages of siting, design, manufacture, construction, assembly, commissioning, operation, maintenance and decommissioning (or closure) of facilities or parts thereof, as appropriate,..."

More specific requirements on safety assessment for facilities and activities are established in IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), Safety Assessment for Facilities and Activities [18], and various IAEA Safety Guides on safety assessment are under development.

Scope of the radiation protection programme

3.60. The radiation protection programme should document the following, with an appropriate level of detail:

(a) The assignment of responsibilities for protection and safety for workers to different management levels, including corresponding organizational arrangements and, if applicable (e.g. in the case of itinerant workers), the allocation of the respective responsibilities between employers and the registrant or licensee;
(b) The designation and functions of qualified experts, as appropriate (see paras 3.65–3.71);
(c) The integration of occupational radiation protection with other areas of health and safety, such as industrial hygiene, industrial safety and fire safety;
(d) The system for the accountability for radiation generators and radioactive sources (see paras 3.72–3.74);
(e) The designation of controlled areas and supervised areas (see paras 3.75–3.86);
(f) The local rules for workers to follow and the supervision of work (see paras 3.87–3.92);
(g) The provision of personal protective equipment, if applicable (see paras 3.93 and 9.53–9.64);
(h) The arrangements for monitoring workers and the workplace, including the acquisition and maintenance of suitable instruments (see paras 3.97–3.128 and Section 7);
(i) The system for recording and reporting all of the relevant information relating to the control of exposures, the decisions regarding measures for occupational radiation protection and safety, and the monitoring of individuals (see paras 3.132–3.140 and Section 7);

(j) The education and training programme on the nature of the hazards and on measures for protection and safety (see paras 3.141–3.151);
(k) The methods for periodically reviewing and auditing the performance of the radiation protection programme (see paras 3.157 and 3.158);
(l) The emergency plan, where the need for such a plan is indicated by the safety assessment (see paras 4.5 and 4.6);
(m) The programme for workers' health surveillance (see Section 10);
(n) The requirements for the assurance of quality and process improvement.

3.61. Paragraph 3.13 of GSR Part 3 [2] states that:

> "Registrants and licensees shall bear the responsibility for setting up and implementing the technical and organizational measures that are necessary for protection and safety for the practices and sources for which they are authorized. Registrants and licensees may designate suitably qualified persons to carry out tasks relating to these responsibilities, but they shall retain the prime responsibility for protection and safety. Registrants and licensees shall document the names and responsibilities of persons designated to ensure compliance with the requirements of these Standards."

3.62. The responsibility for the implementation of the radiation protection programme within an organization should be allocated by the management to staff as appropriate. The responsibilities of each hierarchical level, from the senior management to workers involved in specific tasks regarding each aspect of the radiation protection programme should be clearly delineated and documented in written policy statements to ensure that all management and staff are aware of them.

3.63. The organizational structures should reflect the assignment of responsibilities and the commitment of the organization to protection and safety. The management structure should facilitate cooperation between the various individuals involved. The radiation protection programme should be designed in such a way that the relevant information is provided to the individuals in charge of the various aspects of the work.

3.64. In order to coordinate decision making concerning the choice of measures for protection and safety, it may be appropriate, depending on the size and complexity of the facility, to create a specific advisory committee with representatives of those departments concerned with occupational exposure. The main purpose of this committee would be to advise senior management on the radiation protection programme. Its members should therefore include

management staff from the relevant departments and workers with field experience. The functions of the committee should be to delineate the main objectives of the radiation protection programme in general, and operational radiation protection in particular, to validate the goals of radiation protection, to make proposals regarding the choice of measures for protection and safety, and to make recommendations to the management with regard to the resources, methods and tools to be assigned to the fulfilment of the radiation protection programme.

Qualified experts

3.65. The radiation protection programme should specify the need for, and designate, qualified experts in the relevant fields such as the following:

(a) Radiation protection;
(b) Internal and external dosimetry;
(c) Workplace monitoring;
(d) Ventilation (e.g. in underground mines);
(e) Occupational health;
(f) Radioactive waste management.

3.66. The management should ensure that the relevant services of qualified experts are provided and that the persons providing such services relating to radiation protection work in close cooperation and maintain close working contacts with persons responsible for the control of non-radiological hazards. A radiation protection officer should be appointed, when required by the regulatory body, to oversee the application of the relevant regulatory requirements and compliance.

3.67. The functions of the qualified experts in each field are interrelated in many ways and may be combined for the operation of some facilities. For instance, in a small underground mine, it might be appropriate to combine the functions of the radiation protection officer and the ventilation officer. Where the responsibilities are divided between two or more qualified experts, the qualified experts should maintain a close working relationship.

3.68. The qualified experts should report directly to the senior representative of the employer at the facility who has overall responsibility for safety.

3.69. The qualified experts should be provided with adequate equipment, resources and staff to fulfil their functions.

3.70. The effectiveness of the control measures implemented by the qualified experts should be assessed periodically.

3.71. Management should consult the appointed qualified experts, as appropriate, on aspects of the radiation protection programme, including the designation of controlled areas and supervised areas, the preparation of local rules, the provision of personal protective equipment and the arrangements for monitoring the workplace and workers, and on any subsequent changes having a significant impact on protection and safety.

Accountability for radiation generators and radioactive sources

3.72. The basic requirement is set out in Requirement 17 of GSR Part 3 [2], which states that "**Registrants and licensees shall ensure the safety of radiation generators and radioactive sources.**"

3.73. More detailed requirements on ensuring the safety of radiation generators and radioactive sources are given in paras 3.49–3.60 of GSR Part 3 [2]. Guidance on the safety of radiation generators and sealed radioactive sources is given in IAEA Safety Standards Series No. RS-G-1.10, Safety of Radiation Generators and Sealed Radioactive Sources [19].

3.74. The accountability system for radiation generators and radioactive sources should include an inventory that contains records of the location and description of each radiation generator or radioactive source, and the activity and physical and chemical form of each radioactive source. This inventory should be updated and verified periodically. In addition, consideration should be given to keeping records on any special instructions for each radioactive source held and details of the disposal of any such source.

Classification of areas

3.75. The management should consider classifying working areas whenever there is occupational exposure to radiation. These areas should be clearly defined in the radiation protection programme, and their classification should result from the prior radiological evaluation referred to in paras 3.53–3.56. Two types of area can be defined: controlled areas and supervised areas.

Controlled areas

3.76. Detailed requirements for controlled areas are set out in paras 3.88–3.90 of GSR Part 3 [2], which state that:

> "3.88. Registrants and licensees shall designate as a controlled area any area...in which specific measures for protection and safety are or could be required for:
>
> (a) Controlling exposures or preventing the spread of contamination in normal operation;
> (b) Preventing or limiting the likelihood and magnitude of exposures in anticipated operational occurrences and accident conditions.
>
> "3.89. In defining the boundaries of any controlled area, registrants and licensees shall take account of the magnitude of the exposures expected in normal operation, the likelihood and magnitude of exposures in anticipated operational occurrences and in accident conditions, and the type and extent of the procedures required for protection and safety.
>
> "3.90. Registrants and licensees:
>
> (a) Shall delineate controlled areas by physical means or, where this is not reasonably practicable, by some other suitable means.
> (b) Shall, where a source is only intermittently brought into operation or energized, or is moved from place to place, delineate an appropriate controlled area by means that are appropriate under the prevailing circumstances and shall specify exposure times.
> (c) Shall display the symbol recommended by the International Organization for Standardization...and shall display instructions at access points to and at appropriate locations within controlled areas.
> (d) Shall establish measures for protection and safety, including, as appropriate, physical measures to control the spread of contamination and local rules and procedures for controlled areas.
> (e) Shall restrict access to controlled areas by means of administrative procedures such as the use of work permits, and by physical barriers, which could include locks or interlocks, the degree of restriction being commensurate with the likelihood and magnitude of exposures.
> (f) Shall provide, as appropriate, at entrances to controlled areas:
> (i) Personal protective equipment;
> (ii) Equipment for individual monitoring and workplace monitoring;

(iii) Suitable storage for personal clothing.

(g) Shall provide, as appropriate, at exits from controlled areas:
  (i) Equipment for monitoring for contamination of skin and clothing;
  (ii) Equipment for monitoring for contamination of any objects or material being removed from the area;
  (iii) Washing or showering facilities and other personal decontamination facilities;
  (iv) Suitable storage for contaminated personal protective equipment.

(h) Shall periodically review conditions to assess whether there is any need to modify the measures for protection and safety or the boundaries of controlled areas.

(i) Shall provide appropriate information, instruction and training for persons working in controlled areas."

3.77. An area should be designated as a controlled area when the management considers that there is a need to adopt procedural controls to ensure an optimized level of protection and compliance with the relevant dose limits. The designations should be based on operational experience and judgement. In areas where there is no problem of contamination by unsealed radioactive substances, designated areas may sometimes be defined in terms of the dose rate at the boundary. Values of dose rate based on a fraction of the relevant dose limit have often been used in the past for defining the boundaries of controlled areas. Such an approach might still be appropriate, but it should not be used without careful radiological evaluation. For instance, account should be taken of the length of time for which the dose rate remains at, or above, the defined level and the risks of potential exposures.

3.78. Work with unsealed radioactive substances can result in contamination of the air and surfaces, and this, in turn, can lead to intakes of radionuclides by workers. Such contamination will generally be of an intermittent nature, and it will not normally be possible to control intakes by placing reliance solely on design features, particularly in the event of an incident. Operational procedures will, therefore, be necessary to prevent or reduce the possibility of intake, and controlled areas should, in general, be established.

3.79. Controlled areas may not need to be set up where only small quantities of unsealed radioactive substances are used (e.g. for tracer studies in a research laboratory). They may also be unnecessary when only materials with low activity concentrations are handled, such as materials in various industrial activities involving naturally occurring radioactive material.

3.80. The caution signs at the entrances to controlled areas should be used to indicate to employees, especially maintenance staff, that special procedures apply in the area and that radiation sources are likely to be present.

3.81. In setting up controlled areas, the management may find it useful to make use of existing physical boundaries, such as the walls of rooms or buildings. This might mean that the areas will be larger than would strictly be necessary on the basis of radiation protection considerations alone. For instance, for practical purposes, in some underground uranium mines, it may be appropriate to designate as a controlled area the entire underground area. Similarly, in some diagnostic medical facilities, it may be appropriate to designate the entire examination room as a controlled area.

3.82. In specifying access controls for controlled areas, practical considerations and the need for access controls for other (non-radiological) reasons should be taken into account. In many workplaces, especially those in purpose designed buildings involving relatively few workers, comprehensive controls, such as physical barriers involving locks and interlocks, might be practical to install and operate, and may be required already for security reasons. In other workplaces, such as underground mines, in which thousands of workers are employed, means of access controls such as cards and tags and supervision may be the more practical and appropriate alternative.

Supervised areas

3.83. Requirements for supervised areas are set out in paras 3.91 and 3.92 of GSR Part 3 [2], which state that:

> "3.91. Registrants and licensees shall designate as a supervised area any area not already designated as a controlled area but for which occupational exposure conditions need to be kept under review, even though specific measures for protection and safety are not normally needed.
>
> "3.92. Registrants and licensees, taking into account the nature, likelihood and magnitude of exposures or contamination in the supervised areas:
>
> (a) Shall delineate the supervised areas by appropriate means;
> (b) Shall display approved signs, as appropriate, at access points to supervised areas;

(c) Shall periodically review conditions to assess whether there is any need for further measures for protection and safety or any need for changes to the boundaries of supervised areas."

3.84. The essential purpose of a supervised area is to identify those parts of the workplace that should be subject to regular review of the radiological conditions to determine whether the status of the area should be changed — as a result, for example, of circumstances that were not foreseen in the prior radiological evaluation — or whether there has been some breakdown of control, either in the design features or in the procedures that apply in any adjacent controlled area. Usually, the review of the radiological conditions would comprise a programme of regular monitoring of the area and, in some cases, of the individuals who work in it. It should not automatically be necessary to set up a supervised area around every controlled area, as the requirements that apply within a designated controlled area may well be sufficient.

3.85. As with controlled areas, the definitions of supervised areas are best based on operational experience and judgement, but again, use may be made of a dose rate to define the boundary. A reasonable objective would be to ensure that those workers exposed outside designated areas receive the same level of protection as if they were members of the public. This would imply the use of a dose rate based on an effective dose of 1 mSv in a year as one possible means of defining the outer boundary of a supervised area. The conditions in supervised areas should be such that employees are able to enter the area with a minimum number of formalities for radiation protection. Furthermore, it may be appropriate to make use of existing physical boundaries when defining supervised areas (see para. 3.81).

3.86. Although it may be appropriate in many cases for the boundaries of supervised areas to be marked with caution signs, this may not always be necessary or productive. For example, it may be necessary to designate supervised areas in parts of hospitals to which members of the public might have access; signs at the entrances to such areas may cause unnecessary concern.

Local rules and supervision

3.87. As stated in para. 3.94(a) and (b) of GSR Part 3 [2], the management:

"in consultation with workers, or through their representatives where appropriate:

(a) Shall establish in writing local rules and procedures that are necessary for protection and safety for workers and other persons;
(b) Shall include in the local rules and procedures any relevant investigation level or authorized level, and the procedures to be followed in the event that any such level is exceeded".

The management should ensure that work involving occupational exposure is adequately supervised and that the rules, procedures and measures for protection and safety are made known to those workers to whom they apply. The management should also take all reasonable steps to ensure that the rules, procedures and measures for protection and safety are observed.

3.88. The local rules and procedures should correspond to the design and objectives of the facility concerned, and should be designed to aid in the optimization of protection and safety.

3.89. The local rules and procedures should describe the organizational structures and the procedures to be followed in controlled areas and may include some, or all, of the provisions for various components of the radiation protection programme, such as the following:

(a) Monitoring of exposures and contamination;
(b) Engineered controls such as ventilation systems;
(c) Use of personal protective equipment;
(d) Personal hygiene;
(e) Workers' health surveillance;
(f) Management of radioactive waste;
(g) Environmental monitoring;
(h) Management system;
(i) Training;
(j) Development of a safety culture;
(k) Keeping of records;
(l) Reporting;
(m) Emergency preparedness and response, where appropriate.

3.90. The local rules and procedures should be prominently displayed or should be readily available in the workplace.

3.91. Workers should be given adequate training to enable them to comply with the local rules and procedures.

3.92. The management should assign responsibility for the supervision of tasks. This supervision should be exercised to ensure that all the required measures for protection and safety have been followed during work time. In remote workplaces, such a responsibility should be assigned to the direct supervisor at the site of the work.

Personal protective equipment

3.93. When engineered and administrative controls are not sufficient to provide an optimized level of protection for the tasks to be performed, the management is required, in accordance with para. 3.95 of GSR Part 3 [2] to ensure that workers are provided with suitable and adequate personal protective equipment that has been maintained in proper condition and, if appropriate, tested at regular intervals. When measures for reduction of exposures by using personal protective equipment are being considered, account should be taken of any possible increased exposure due to delays or inconveniences caused by the use of the equipment. Workers should be trained in the use of such personal protective equipment prior to the start of the work. Further details on the use of personal protective equipment are given in paras 9.53–9.64.

Work planning and work permits

3.94. When work is to be conducted during which significant radiation levels or contamination levels might be encountered, or when the work is complex (involving several groups of workers and numerous activities), advance work planning is one of the most important means of achieving optimization of protection and safety. The radiation protection officer should take part in the planning of the work and should advise on the conditions under which work can be undertaken in controlled areas. Situations that warrant the use of detailed work plans and work permits are generally encountered in the nuclear industry, but may also be found in non-nuclear industries (e.g. in the maintenance or dismantling of accelerators). Additional guidance on the use of work planning for optimization at nuclear power plants has been published by the OECD Nuclear Energy Agency [20].

3.95. Written procedures should be used as part of the work planning process, as appropriate and depending upon the type of facility or activity. Elements to be considered include the following:

(a) Information from similar work completed previously;

(b) Time for starting the work, its estimated duration and the human resources involved;
(c) Maps of estimated dose rates;
(d) Operational state of the plant (e.g. for a nuclear power plant, cold or hot shutdown, and operation at full or reduced power);
(e) Other activities in the same area which could interfere with the work;
(f) Preparation and assistance in operations (e.g. isolation of the process, scaffolding and insulation work);
(g) Protective clothing and tools to be used;
(h) Communication necessary to ensure supervisory control and coordination;
(i) Management of any radioactive waste arising from the work;
(j) Coordination with protective measures for conventional safety.

3.96. For each task that needs special radiological precautions to be taken, a radiation work permit should normally be prepared. The radiation work permit is issued by the persons in charge of the planning of the operations, in collaboration with the radiation protection officer. A copy of the radiation work permit should be provided to the supervisor of the work and should remain with the working team during the performance of the work. In addition to a description of the work to be performed, the radiation work permit can include:

(a) A detailed dose rate map of the working area and possible hot spots, produced from a survey made prior to the work or otherwise estimated;
(b) An estimate of contamination levels and how they could change during the course of the work;
(c) Specification of any additional workplace monitoring of radiation levels to be carried out before or during the work;
(d) An estimate of individual exposure and collective exposure for each work step;
(e) Specification of any additional dosimeters to be used by workers;
(f) Specification of personal protective equipment to be used in different phases of the work;
(g) Details of any time restrictions or dose restrictions;
(h) Instructions on when to contact the radiation protection officer.

Monitoring and assessment of exposures

Objectives of monitoring

3.97. The general term ‘monitoring’ refers to a process that includes the making of measurements in relation to the assessment or control of exposure to radiation

and exposure due to radioactive materials. Although measurements play a major part in any monitoring programme, monitoring is more than simply measurement; it requires interpretation and assessment. The primary justification for making a measurement should therefore be expressed in terms of the way in which it helps to achieve and demonstrate adequate protection and safety, including in the optimization process.

3.98. A programme of monitoring may serve various purposes, depending on the nature and extent of the practice. These purposes can include the following:

(a) Assessing the exposure of workers and demonstrating compliance with regulatory requirements.
(b) Confirming the effectiveness of working practices (e.g. the adequacy of supervision and training) and engineering standards.
(c) Determining the radiological conditions in the workplace, whether these are under adequate control and whether operational changes have improved or worsened the situation.
(d) Evaluating and improving operating procedures from a review of the collected monitoring data for individuals and groups. Such data may be used to identify both good and bad features of operating procedures and design characteristics, and thereby contribute to the development of safer working practices in relation to radiation.
(e) Providing information that can be used to enable workers to understand how, when and where they are exposed, and to motivate them to take steps to reduce their exposure.
(f) Providing information for the evaluation of doses in the event of accidental exposures.

Furthermore, monitoring data may be used for the purpose of risk–benefit analysis and to supplement medical records.

3.99. Monitoring can provide important supplementary benefits in the fields of industrial relations or public relations (such as reassurance and motivation of the workforce) or of scientific investigation (such as data for epidemiological studies), or in providing information useful in the determination of liability in the event of the expression of adverse health effects in individual workers. These considerations may affect decisions about the nature and extent of monitoring programmes, but they do not in themselves provide the primary justification for a monitoring programme for protection and safety.

Monitoring programme

3.100. The principal responsibility for setting up a monitoring programme rests with the management. The monitoring programme should be designed in consultation with an appropriate qualified expert on the basis of the prior radiological evaluation discussed in paras 3.53–3.59, with due account being taken of regulatory requirements.

3.101. Monitoring programmes can be divided and subdivided into several different types. The first division relates to the objectives of the monitoring. At this level, four types of monitoring can be defined for the purposes of radiation protection:

(a) Routine monitoring is associated with continuing operations and is intended to meet regulatory requirements and to demonstrate that the working conditions, including the levels of individual dose, remain satisfactory.
(b) Special monitoring is investigative in nature and typically covers a situation in the workplace for which insufficient information is available to demonstrate adequate control. It is intended to provide detailed information to elucidate any problems and to define future procedures. It should normally be undertaken at the commissioning stage of new facilities, or following major modifications to facilities or procedures, or when operations are being carried out under abnormal circumstances, such as an accident.
(c) Confirmatory monitoring is performed where there is a need to check assumptions made about exposure conditions (e.g. to confirm the effectiveness of protective measures).
(d) Task related monitoring applies to a specific operation. It provides data to support the immediate decisions on the management of the operation. It may also support the optimization of protection.

3.102. Each of these types of monitoring programme can be subdivided on the basis of the location of the monitoring:

(a) Individual monitoring comprises measurements made using equipment worn by individual workers, or measurements of quantities of radioactive substances in or on their bodies, and the interpretation of such measurements.
(b) Workplace monitoring comprises measurements made in the working environment and the interpretation of such measurements.

3.103. Individual monitoring can be further subdivided into monitoring for external exposure, for internal exposure and for skin contamination. Workplace monitoring can be further subdivided into monitoring for external radiation, for air contamination and for surface contamination. The details of the programmes will be influenced by factors such as the type and energy of the radiation and the radionuclides involved (see Section 7).

3.104. The programme design should reflect the objectives of the monitoring programme, and these should be clearly specified and recorded. The design should include the basis for the interpretation of the monitoring results and how this relates to the objectives of the programme, and this basis should be recorded. A distinction should be made in the programme between monitoring for the purpose of controlling operations and monitoring for the formal assessment of exposure to meet regulatory requirements.

3.105. The equipment to be used in the monitoring programme should be suitable for the types of radiation and the forms of radioactive material encountered in the workplace. The equipment should be calibrated to meet appropriate standards. More detailed guidance, including guidance on the provision of approved dosimetry services, is presented in Section 7. Guidance on the management system for dosimetry service providers is given in Section 8.

3.106. The design and implementation of a monitoring programme should conform to the quality assurance requirements embodied in the management system to ensure that procedures are established and followed correctly and to ensure that records are promptly compiled and correctly maintained. The design of the monitoring programme should indicate the records that should be kept, and the associated procedures for keeping and discarding records. All of these aspects should be reviewed regularly, at predetermined intervals or following any major change in operations of the installation or in regulatory requirements. The purpose of such reviews should be to ensure that the monitoring effort (type, frequency and extent) is appropriately employed. The information should also be used to identify both good and bad features of operating procedures, and both good and bad design characteristics.

Individual monitoring

3.107. The need for, and appropriateness of, individual monitoring of workers will depend on factors such as the following:

(a) The amount of radioactive material present and the radionuclides involved;

(b) The physical and chemical form of the radioactive material;
(c) The type of containment used;
(d) The operations performed;
(e) The expected levels and likely variations in the doses or intakes;
(f) The complexity of the measurement procedures and interpretation procedures of the measurement programme;
(g) The general working conditions.

For example, workers involved in the handling of sealed sources (or dispersible sources that are in closed containers with or without shielding) may need to be monitored for external exposure but not for internal exposure. Conversely, workers handling radionuclides such as ^{3}H, ^{125}I or ^{239}Pu may need to be monitored for internal exposure but not for external exposure.

3.108. The need for individual monitoring is likely to be greater in the early stages of an operation. As experience in the workplace is accumulated, the need for routine individual monitoring can be kept under review to decide on the need for continuation of individual monitoring or whether workplace monitoring is sufficient for radiation protection purposes. In determining the necessity for individual monitoring, consideration should also be given to the potential for accidental exposures.

3.109. For work involving internal exposure, the decision to register a worker in an individual monitoring programme should be based on the likelihood of an intake of radionuclides in excess of a predetermined level. If operational procedures need to be set up to prevent or reduce the possibility of an intake, a controlled area should, in general, be established. Individual monitoring for intakes of radionuclides should be used routinely only for workers who are employed in areas that are designated as controlled areas specifically in relation to the control of contamination and in which there are grounds for expecting significant intakes. If experience has shown that it is unlikely that committed effective doses from occupational exposure due to annual intakes of radionuclides would exceed 1 mSv, then individual monitoring may be unnecessary, but workplace monitoring should be undertaken. The following activities are examples of those for which routine individual monitoring for internal exposure should be considered:

(a) The handling of large quantities of gaseous or volatile materials, for example of tritium and its compounds in large scale production processes, in heavy water reactors and in manufacturing of gaseous light sources;
(b) The processing of plutonium and other transuranic elements;

(c) The maintenance of reactor facilities, which can lead to exposure due to fission products and activation products;
(d) The bulk production of radioisotopes;
(e) The production and handling of large quantities of radiopharmaceuticals, such as ^{18}F for diagnostics by positron emission tomography or ^{131}I for therapy;
(f) The mining of high grade uranium ores, processing of uranium mineral concentrates and production of nuclear fuel;
(g) The processing of mineral concentrates, such as monazite that is rich in thorium, and the production of products containing thorium.

3.110. To obtain the necessary accuracy and precision, individual dosimetry should be performed, whenever possible, by an approved dosimetry service. The regulatory body should give consideration to the establishment of a national accreditation procedure as a basis for the approval of dosimetry services. The management system for dosimetry service providers is discussed in Section 8.

3.111. For visitors making short and infrequent visits to controlled areas, individual monitoring may be performed but is not necessarily required. However, a record of the radiological conditions in the controlled areas visited (e.g. data from workplace monitoring or from individual monitoring of the visitors' escort) and the length of time spent in these areas during the visits should be retained.

Workplace monitoring

3.112. The requirements for workplace monitoring are set out in paras 3.96–3.98 of GSR Part 3 [2], which state that:

> "3.96. Registrants and licensees, in cooperation with employers where appropriate, shall establish, maintain and keep under review a programme for workplace monitoring under the supervision of a radiation protection officer or qualified expert.
>
> "3.97. The type and frequency of workplace monitoring:
>
> (a) Shall be sufficient to enable:
> (i) Evaluation of the radiological conditions in all workplaces;
> (ii) Assessment of exposures in controlled areas and supervised areas;
> (iii) Review of the classification of controlled areas and supervised areas.

(b) Shall be based on dose rate, activity concentration in air and surface contamination, and their expected fluctuations, and on the likelihood and magnitude of exposures in anticipated operational occurrences and accident conditions.

"3.98. Registrants and licensees, in cooperation with employers where appropriate, shall maintain records of the findings of the workplace monitoring programme. The findings of the workplace monitoring programme shall be made available to workers, through their representatives where appropriate."

3.113. The programmes for monitoring the workplace should specify:

(a) The quantities to be measured;
(b) Where and when the measurements are to be made, and at what frequency;
(c) The most appropriate methods and procedures for measurement;
(d) Investigation levels and the actions to be taken if they are exceeded.

3.114. The results and findings of workplace monitoring should be recorded and should be made available to the management and to workers through their representatives, where appropriate. This information should be used in support of pre-job and post-job evaluations, work planning, control of contamination and management of radiological control operations. Significant changes in monitoring results should be identified and trends should be analysed periodically. Corrective actions should be taken as necessary. Data should be recorded that:

(a) Demonstrate compliance with regulations;
(b) Identify significant changes to the working environment;
(c) Give details of radiation surveys, for example date, time, location, dose rate, airborne activity concentration, instruments used, surveyor or other comments;
(d) Give details of any reports received about the workplace, whereby compliance with relevant requirements could be adversely affected;
(e) Give details of any appropriate actions taken.

3.115. Particular attention should be given in the selection and use of instruments to ensure that their performance characteristics are appropriate for the specific workplace monitoring situation. Guidance on considerations relating to the acquisition, use, maintenance and testing of workplace monitoring instruments is given in Section 7.

Assessment of exposure

3.116. Specific requirements for the assessment of occupational exposure are set out in paras 3.99–3.102 of GSR Part 3 [2], which state that:

> "3.99. Employers, as well as self-employed persons, and registrants and licensees shall be responsible for making arrangements for assessment of the occupational exposure of workers, on the basis of individual monitoring where appropriate, and shall ensure that arrangements are made with authorized or approved dosimetry service providers that operate under a quality management system.
>
> "3.100. For any worker who usually works in a controlled area, or who occasionally works in a controlled area and may receive a significant dose from occupational exposure, individual monitoring shall be undertaken where appropriate, adequate and feasible. In cases where individual monitoring of the worker is inappropriate, inadequate or not feasible, the occupational exposure shall be assessed on the basis of the results of workplace monitoring and information on the locations and durations of exposure of the worker[33].
>
> "3.101. For any worker who regularly works in a supervised area or who enters a controlled area only occasionally, the occupational exposure shall be assessed on the basis of the results of workplace monitoring or individual monitoring, as appropriate.
>
> "3.102. Employers shall ensure that workers who could be subject to exposure due to contamination are identified, including workers who use respiratory protective equipment. Employers shall arrange for appropriate monitoring to the extent necessary to demonstrate the effectiveness of the measures for protection and safety and to assess intakes of radionuclides and the committed effective doses.
>
> "[33] The distinction between types of worker in paras 3.100 and 3.101 for the purposes of monitoring has similarities to the distinction between category A and category B workers in European Union legislation [(see Ref. [21])]."

3.117. An assessment of the exposure of individual workers in normal and foreseeable abnormal conditions should be considered if, for any single component of the exposure (e.g. strongly penetrating photon irradiation, neutron irradiation or internal exposure), the corresponding annual effective dose is

expected to exceed 1 mSv. Consideration should also be given to the likelihood and possible magnitude of potential exposures.

3.118. In general, when the magnitude or variability of the exposure is likely to be significant, an individual worker's radiation exposure should be assessed from the results of individual monitoring. There are occasions, particularly in the assessment of internal exposure, when this may not be feasible or practicable and reliance should be placed on workplace monitoring. Where this is the case, the monitoring programme should provide detailed information on the worker's movements and on the temporal and spatial variations in air concentrations in the worker's immediate environment. Where possible, site specific data on characterization of the workplace should be used rather than default values.

3.119. For work involving risk of internal exposure, a level of activity concentration in air or intake of activity into the body may need to be established to be used as an indication of whether there is the potential for a significant individual exposure. In the derivation of such a level, the particular radioactive materials and exposure pathways of the relevant workplace should be taken into account to the extent possible. If the level is exceeded, additional direct measurements of the individual's internal exposure may be necessary. This may also be desirable if there is any doubt as to whether the assessed exposure for the specific workplace conditions is sufficiently accurate.

3.120. For any assessment of occupational exposure, the accuracy of the particular monitoring procedures or devices used to determine external and internal exposure should be evaluated. The objective should be to establish as comprehensive a record as is reasonable of credible, formally assessed exposures. Account should be taken of the factors affecting the accuracy of the assessment. The accuracy criteria for measurements and their interpretation should be defined, and reasonable and appropriate measures to quantify and minimize uncertainties should be taken.

3.121. More detailed guidance on assessment of exposure is provided in Section 7.

Investigation levels

3.122. Experience with a particular situation sometimes indicates a need to review procedures and performance. This experience can be qualitative (e.g. the observation that the frequency of occurrence of minor contamination may have increased) or quantitative (e.g. a trend in the results of monitoring programmes).

The use of quantitative experience can be helped by the application of investigation levels to the monitoring results for individuals and workplaces. An investigation level is defined as the "value of a quantity such as effective dose, intake or contamination per unit area or volume at or above which an investigation would be conducted" [2].

3.123. Investigation levels play an important role in monitoring programmes as tools for use by the management. Investigation levels should be defined at the planning stage of activities and may be revised on the basis of operational experience. The regulatory body may also wish to establish, for regulatory purposes, a generic investigation level in terms of individual exposure. Investigation levels can be set in terms of virtually any measurable quantity relating to the individual or the working environment. They should be defined by the management in the radiation protection programme, their purpose being to facilitate the control of operations and exposures.

3.124. Investigation levels should be used in a retrospective sense only and should not be confused with dose constraints. If an investigation level is exceeded, a review should be initiated to determine the causes, and to consider the arrangements for protection and safety, and the reasons for the value being exceeded. Such a review may lead to the introduction of additional measures for protection and safety. The review should have the objectives of learning lessons that may be appropriate for any future operations and determining whether additional measures are necessary to improve the current arrangements for protection and safety.

3.125. Investigation levels should be set by the management on the basis of a knowledge of the conditions in the workplace, the expected levels and variability of the quantities being determined (e.g. effective dose and intake), and the type and frequency of monitoring. The value of the investigation level should also be consistent with the objectives of the monitoring programme and with the type of investigation that will be initiated. The value of an investigation level may be based on a selected fraction of the relevant dose limit, and it should correspond to the period of time to which the individual monitoring result refers. For instance, an investigation level for a routine operation with routine monitoring may be set on the basis of a committed effective dose of 5 mSv from intakes over the course of a year. For N monitoring periods per year, the investigation level IL_j (in becquerels) for the intake of radionuclide j in a given monitoring period would be given by:

$$\mathrm{IL}_j = \frac{0.005}{N \cdot e(g)_j} \tag{19}$$

where $e(g)_j$ is the dose coefficient for inhalation or ingestion of radionuclide j, as appropriate (in sieverts per becquerel). The value of the investigation level should be established with other sources of exposure taken into account.

3.126. A level may be set for individuals involved in a particular operation, or may be derived specifically for individuals within a place of work without reference to a particular operation. The latter situation is particularly relevant when individuals are exposed to a number of different sources in a workplace or are involved in a number of different tasks at work.

3.127. The management should identify those persons responsible for initiating investigations when they are required. The purpose of, and the actions associated with, each investigation level should be clearly defined in advance. The investigation should address:

(a) The circumstances leading to the suspected exposure;
(b) Verification of the dosimetric results;
(c) The probability that dose limits or levels will be exceeded under current working conditions;
(d) Corrective actions to be taken.

3.128. Workplace monitoring may involve the measurement of dose rates, contamination levels, airborne activity concentrations or a combination thereof. Investigation levels for workplace monitoring should be set by the management on the basis of the expected levels and operational experience. A value of surface contamination (activity per unit area) derived from a fraction of the relevant dose limit may be useful in indicating the significance of particular measurements, and could, therefore, be used as an investigation level to indicate a deterioration in the radiological conditions in the workplace.

Recording levels

3.129. During the routine monitoring of the workplace or of individuals, a large amount of data will be generated that may have little quantitative significance in terms of converting them into effective (or equivalent) dose. A recording level is a level of dose, exposure or intake specified by the regulatory body at, or above, which values of dose to, exposure of, or intake by workers are to be entered

into their individual exposure records (see para. 3.105(b) of GSR Part 3 [2]). For instance, the recording level for an intake of a radionuclide could be set to correspond to a committed effective dose of 1 mSv from intakes over the course of a year. Thus, for N monitoring periods per year, the recording level RL_j (in becquerels) for intake of radionuclide j in a given monitoring period would be given by:

$$\mathrm{RL}_j = \frac{0.001}{N \cdot e(g)_j} \tag{20}$$

In cases of exposure of workers to radiation of more than one type or to multiple radionuclides, the contributions of each type of radiation or each radionuclide should be taken into account in selecting the recording level for each contribution to the dose, exposure or intake. In the case of individual monitoring for external exposure, the minimum level of detection is usually used as the recording level.

3.130. For the assessment of internal dose, if the dose or intake is below the recording level, the measurement result should always be maintained in the dose record for the workplace and/or the individual.

Derived investigation and recording levels

3.131. It may be convenient to express investigation levels and recording levels in terms of the quantities actually measured (i.e. radionuclide activities measured in the body or in excretion samples). These are termed derived investigation levels (DILs) and derived recording levels (DRLs), respectively. They are the measurement values that correspond to the investigation levels or recording levels for parameters such as committed effective dose or radionuclide intake. For intakes of radionuclides, DILs and DRLs are calculated separately for each radionuclide, are specific to the physical and chemical form of the radionuclide in the workplace, and are a function of the period of time between the time of intake and the time of measurement. For the examples given in Eqs (19) and (20):

$$\mathrm{DIL}_j = \frac{0.005}{N \cdot e(g)_j} m(t_0)_j \tag{21}$$

$$\mathrm{DRL}_j = \frac{0.001}{N \cdot e(g)_j} m(t_0)_j \tag{22}$$

where $m(t_0)_j$ is the fraction of the intake of radionuclide j remaining in the body or in the excretion sample after an elapsed time period t_0. The value of t_0 is usually

based on the assumption that the intake occurs at the midpoint of the monitoring period, in which case:

$$t_0 = \frac{365}{2N} \text{ days} \tag{23}$$

Records of occupational exposure

3.132. Record keeping is an essential part of the individual monitoring process, as indicated in paras 3.103 and 3.106 of GSR Part 3 [2], which state that:

> "3.103. Employers, registrants and licensees shall maintain records of occupational exposure[34] for every worker for whom assessment of occupational exposure is required in paras 3.99–3.102.
>
> …….
>
> "3.106. Employers, registrants and licensees:
>
> (a) Shall provide workers with access to records of their own occupational exposure;
> (b) Shall provide the supervisor of the programme for workers' health surveillance, the regulatory body and the relevant employer with access to workers' records of occupational exposure;
> (c) Shall facilitate the provision of copies of workers' exposure records to new employers when workers change employment;
> (d) Shall make arrangements for the retention of exposure records for former workers by the employer, registrant or licensee, as appropriate;
> (e) Shall, in complying with (a)–(d) above, give due care and attention to maintaining the confidentiality of records.
>
> "[34] Records of occupational exposure are also referred to as 'exposure records' or 'dose records'."

3.133. The management should establish a procedure that indicates how monitoring data and results are to be reported, which dose levels are to be recorded, and which documents and records of occupational exposure are to be maintained. In general, the dosimetry service provider has limited direct contact with workers and the facility management. Monitoring results are, however, often used by the management to advise personnel with responsibilities for operational radiation protection as to when intervention with workers, such as follow-up sampling or restriction of work, is necessary. Consequently, close cooperation is necessary

between those involved in different parts of the monitoring programmes and the protection programmes.

3.134. Records of individual occupational exposure should include any assessed equivalent doses or intakes, including the dose to the skin and to the lens of the eye, as appropriate. Details of any involvement in abnormal events should be included, even if no estimates of exposure could be made. Records referencing the objectives, monitoring methods and models used for data analysis and interpretation should be retained, because these may be needed for future interpretation of the records of occupational exposure. Traceability of the measurements and of assessments of exposure is essential.

3.135. The monitoring programme should specify the periods over which monitoring and assessment of exposure are carried out, these being related to the dosimeter processing or sampling programme. Records of occupational exposure for individual workers such that the exposures assessed for these periods are separately identifiable should be constructed.

3.136. Records of occupational exposure should be kept up to date, and procedures should be established to ensure that assessments of exposure from any monitoring period are incorporated promptly into the individual's exposure record.

3.137. Recording systems need to be capable of producing information on the assessment of occupational exposure for any reporting period defined in the radiation protection programme or required by the regulatory body. If a worker changes employment, records of occupational exposure should be promptly updated and completed.

3.138. The dose records should be easily retrievable and should be protected against loss. Such protection is usually obtained 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; should be complete and accurate; should be identified by site, purpose, date and originator; and should be legible and intelligible to a qualified person. Consideration should be given to any applicable national requirements or international agreements concerning the privacy of individual data records.

3.139. If employers, registrants and licensees cease to conduct activities in which workers are subject to occupational exposure, they should make arrangements for the retention of workers' records of occupational exposure by

the regulatory body or by a State registry, or by a relevant employer, registrant or licensee, as appropriate.

3.140. More detailed guidance on records of occupational exposure is given in Section 7.

Information, instruction and training

3.141. Paragraph 3.110 of GSR Part 3 [2] states that:

> "Employers, in cooperation with registrants and licensees:
>
> (a) Shall provide all workers with adequate information on health risks due to their occupational exposure in normal operation, anticipated operational occurrences and accident conditions, adequate instruction and training and periodic retraining in protection and safety, and adequate information on the significance of their actions for protection and safety;
> (b) Shall provide those workers who could be involved in or affected by the response to an emergency with appropriate information, and adequate instruction and training and periodic retraining, for protection and safety;
> (c) Shall maintain records of the training provided to individual workers."

3.142. It is the management's responsibility to ensure that workers who may be occupationally exposed to radiation and persons with assigned responsibilities in the radiation protection programme receive general information on, and training in, radiation protection. This should include training of workers' representatives and members of relevant safety committees, where appropriate.

3.143. Senior management should be trained in the risks associated with radiation, the basic principles of protection and safety, their main responsibilities regarding the management of radiation risks and the principal elements of the radiation protection programme.

3.144. Training for those workers directly involved in work with radiation sources should include relevant information, presented in the form of documents, lectures, applied training and on the job training that emphasizes procedures specific to the worker's job assignment. Training for workers considered occupationally exposed should address topics at a level of detail commensurate

with the workers' job assignments and the potential hazard. The training should cover topics such as the following:

(a) The main risks associated with ionizing radiation;
(b) Basic quantities and units used in radiation protection;
(c) Requirements for radiation protection (including optimization of protection and limitation of doses);
(d) The fundamentals of practical radiation protection (e.g. use of personal protective equipment, shielding and behaviour in designated areas);
(e) Specific task related issues;
(f) Responsibility to advise a designated person immediately if any unforeseen occurrence involving increased radiation risk arises;
(g) Where appropriate, actions that may need to be taken in the event of an accident.

3.145. Where work involving significant exposure to radiation is to be undertaken, consideration should be given to the use of training on mock-ups or simulators to ensure that the work will proceed as smoothly as possible, that all unnecessary hazards will be avoided and that exposure periods will be minimized.

3.146. Workers who might not be occupationally exposed but whose work may have an impact on the level of exposure of other workers or of members of the public (e.g. designers, engineers and planners) should be provided with appropriate information on the principles of protection and safety. They should also be trained in how to take account of requirements for protection and safety in their activities, so as to optimize the protection of other people.

3.147. Individuals whose job assignments are incidental to the use of radiation, such as caretakers or security staff, and others who might spend brief periods in areas where exposure is possible should be given basic information on the hazards and on any preventive actions to be taken. For such individuals, there is a need only to include a brief discussion of items such as the use of time and distance to limit exposure, a qualitative discussion of the risks from the exposure that they may undergo, and specific directives regarding prohibited, required or recommended actions.

3.148. The specific requirements of GSR Part 3 [2] in relation to female workers who might enter controlled areas or supervised areas are addressed in paras 6.2–6.20. The management should consider the possible need for further information and training in relation to any change of working conditions, so as

to restrict exposure of the embryo or fetus or the breastfed infant following a notification of pregnancy.

3.149. Particular attention should be paid to contractors, including subcontractors and itinerant workers. Employers should cooperate to ensure that contractors, including subcontractors and itinerant workers, are provided with the necessary information and with appropriate training (see paras 6.73–6.76).

3.150. Workers' knowledge of the fundamentals of protection and safety, their level of training and their competence to perform their specified tasks safely should be evaluated, and should be determined to be adequate, prior to any unsupervised assignment. A process for the evaluation of workers' knowledge, level of training and competence should be established by the management.

3.151. Information and training programmes on protection and safety should be documented and approved at an appropriate level within the organization. Such programmes should be reviewed periodically to ensure that they remain up to date. Formal records of each worker's training and testing should be maintained and retained for an appropriate period after cessation of employment. Periodic retraining should be provided to ensure that workers have the most up to date knowledge relevant to their work, and that they do not become complacent about workplace hazards. Retraining should also be undertaken when there are significant changes in policy or procedures. Training should be updated at regular intervals.

3.152. Further guidance on education and training of workers is given in IAEA Safety Standards Series No. RS-G-1.4, Building Competence in Radiation Protection and the Safe Use of Radiation Sources [22].

Workers' qualification and certification

3.153. Workers who require a significant level of expertise in a specific work area involving sealed sources, unsealed sources or radiation generators should be suitably qualified and, where appropriate, should be in possession of the relevant certification. Examples of such workers are diagnostic radiographers, operators of industrial radiography equipment and operators of master–slave manipulators in hot cells for radiation sources.

3.154. The regulatory body should provide guidance on requirements for qualification for each category of job. This guidance should address the minimum educational level, minimum training and retraining requirements, and minimum

level of experience for each job category. In addition, the regulatory body should enforce requirements concerning the recognition of qualifications relating to certain duties and responsibilities, such as those of radiation protection officers. Alternatively, the regulatory body should review and approve, if appropriate, proposals made by the management with regard to training requirements.

3.155. Following the successful completion of the required training and the necessary period of work experience, the worker may be formally recognized as qualified. The recognition of such a qualification may be accorded by the employer, by the regulatory body or by a designated board, society, or professional or academic body.

3.156. It may be appropriate and convenient for the regulatory body to recognize certain training centres and courses for their quality and suitability. Such recognition can be formally conferred by the process of accreditation.

Audits and reviews

3.157. The radiation protection programme should be assessed on a regular basis. Audits and reviews of activities within the radiation protection programme should be scheduled on the basis of the status and importance of the activity. The management system (see paras 2.23–2.26) should include a process for such assessments to identify and correct administrative and management problems that could prevent the achievement of programme objectives. Audits and reviews should be conducted by persons who are technically competent to evaluate the processes and procedures being assessed, but who do not have any direct responsibility for those activities. These may be staff from other work areas within the organization, or there may be advantages in independent assessment by other organizations. The objective of such assessments is to enhance the effectiveness and efficiency of the radiation protection programme.

3.158. Audits and reviews should be performed in accordance with written procedures and checklists. They should be conducted when one or more of the following conditions apply:

(a) When required by the regulatory body;
(b) When a systematic independent assessment of the programme is considered necessary by the management;
(c) Following the implementation of a new radiation protection programme or substantive elements of the radiation protection programme;

(d) When significant changes are made to functional areas of the radiation protection programme, such as significant reorganization or procedural revision;
(e) When necessary to verify implementation of previously identified corrective actions.

EXPOSURE OF WORKERS DUE TO NATURAL SOURCES

Applicability of the requirements for planned exposure situations

3.159. According to para. 3.4 of GSR Part 3 [2], occupational exposure due to natural sources is, in general, subject to the requirements for existing exposure situations (see Section 5). This is always the case when the exposure is due to "Radionuclides of natural origin, regardless of activity concentration, in commodities, including food, feed, drinking water, agricultural fertilizer and soil amendments, and construction materials, and residual radioactive material in the environment" (para. 5.1(c)(ii) of GSR Part 3 [2]). In the case of occupational exposure due to radionuclides of natural origin in materials other than these everyday commodities and due to radionuclides in residues in the environment (these 'other' materials being essentially materials from industrial processes), the applicable requirements depend on the radionuclide activity concentrations, as follows:

(a) If, in any process material, the activity concentration of any radionuclide in the ^{238}U decay series or the ^{232}Th decay series exceeds 1 Bq/g, or if the activity concentration of ^{40}K exceeds 10 Bq/g, the industrial activity is regarded as a practice and the requirements for planned exposure situations apply.
(b) If, in every process material, the activity concentrations of all radionuclides in the ^{238}U decay series and the ^{232}Th decay series are 1 Bq/g or less and the activity concentration of ^{40}K is 10 Bq/g or less, the material is not regarded as naturally occurring radioactive material, the industrial activity is not regarded as a practice and the requirements for existing exposure situations apply.

3.160. The criteria in para. 3.159 represent (in order of magnitude terms) the upper bounds of the activity concentrations in normal soil [23], as illustrated in Fig. 3 for radionuclides in the ^{238}U decay series and the ^{232}Th decay series. It is evident from Fig. 3 that many commercially exploited minerals contain activity

concentrations of ^{238}U and ^{232}Th below 1 Bq/g and may not need to be regulated as naturally occurring radioactive material.

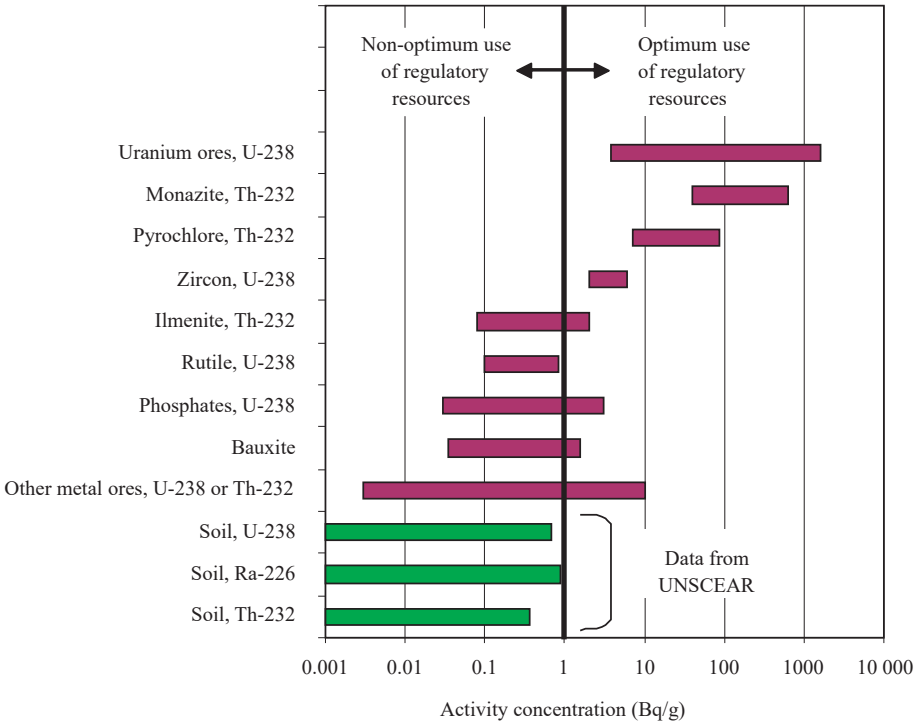


FIG. 3. Radionuclide activity concentrations in natural materials.

3.161. Exposure due to radon in the workplace is normally subject to the requirements for existing exposure situations. However, in terms of para. 3.4 of GSR Part 3 [2], the requirements for planned exposure situations apply to:

(a) Exposure due to ^{222}Rn, ^{220}Rn and their progeny in workplaces in which occupational exposure due to other radionuclides in the ^{238}U decay series or the ^{232}Th decay series is controlled as a planned exposure situation;
(b) Exposure due to ^{222}Rn and its progeny in workplaces in which the annual average activity concentration of ^{222}Rn in the air remains above the reference level (see paras 5.19–5.23).

The ^{222}Rn progeny referred to in para. 3.161(a) and (b) are ^{218}Po, ^{214}Pb, ^{214}Bi and ^{214}Po. The ^{220}Rn progeny referred to in para. 3.161(a) are ^{216}Po, ^{212}Pb, ^{212}Bi, ^{212}Po and ^{208}Tl. Further information on ^{222}Rn, ^{220}Rn and their progeny is given in paras 5.45–5.51.

3.162. As a result of the criteria given in paras 3.159 and 3.161, and with account taken of current published measurements of occupational exposure, the following industrial activities are, or may be, subject to the requirements for planned exposure situations [24]:

(a) Mining and processing of uranium ore;
(b) Extraction of rare earth elements [25];
(c) Production and use of thorium and its compounds;
(d) Production of niobium and ferro-niobium;
(e) Mining of ores other than uranium ore;
(f) Production of oil and gas [26];
(g) Manufacture of titanium dioxide pigments [27];
(h) Activities in the phosphate industry [28];
(i) Activities in the zircon and zirconia industries [29];
(j) Production of tin, copper, aluminium, zinc, lead, and iron and steel;
(k) Combustion of coal;
(l) Water treatment.

Graded approach

3.163. The graded approach to regulation should be adopted for industrial activities involving naturally occurring radioactive material because of the following:

(a) The economic importance of many industries involving naturally occurring radioactive material.
(b) The large volumes of residues and process wastes that can be generated, and thus the limited options for their management.
(c) The potentially high cost of regulation in relation to the reductions in exposure that can realistically be achieved when exposure levels and the associated radiation risks are already rather low.
(d) The recognition that doses are always expected to be well below the threshold for deterministic health effects; in addition, there is never any real prospect of a radiological emergency.

3.164. In order to determine the optimum regulatory approach, the regulatory body should go beyond just establishing that the criteria in paras 3.159 and 3.161 are exceeded. It should consider, in addition, particular types of operation, process and material in more detail, including a prior radiological evaluation of possible exposures and consideration of the costs of regulation in relation to the benefits achievable.

3.165. In terms of the graded approach, the regulatory body should first determine whether exemption of the practice is the optimum regulatory option; experience has shown that this could well be the case for many industrial activities involving naturally occurring radioactive material. For exposure due to naturally occurring radioactive material, the criterion for exemption without further consideration, as given in para. I.4 of GSR Part 3 [2], is a dose of the order of 1 mSv or less in a year. In deciding upon the optimum regulatory option (exemption, notification, registration or licensing), due account should be taken of the effect (and effectiveness) of existing controls that could reduce doses and that could already be in place as a result of other forms of regulation, such as occupational health and safety regulation, otherwise the dose may be significantly overestimated. The need for the highest level of the graded approach (licensing) for practices involving exposure due to naturally occurring radioactive material is likely to be limited to only those operations involving substantial quantities of material with very high radionuclide activity concentrations.

3.166. According to para. I.12(b) of GSR Part 3 [2], material (e.g. naturally occurring radioactive material residues) containing radionuclides of natural origin within an authorized practice can be cleared without further consideration from regulatory control provided that the activity concentration of each radionuclide in the ^{238}U decay series or the ^{232}Th decay series is 1 Bq/g or less and the activity concentration of ^{40}K is 10 Bq/g or less.

3.167. Material that has been cleared from an authorized facility on account of its low radionuclide content could still give rise to non-radiological risks to humans and to the environment as a result of other constituents, such as heavy metals. Such material may, therefore, require ongoing control under the relevant regulations.

Prior radiological evaluation

Exposure pathways

3.168. When conducting a prior radiological evaluation of industrial activities involving naturally occurring radioactive material, the exposure pathways to workers that are most likely to necessitate consideration are those involving external exposure to gamma radiation emitted from process material and internal exposure via the inhalation of radionuclides in dust, as follows:

(a) The main radionuclides of natural origin contributing to gamma exposure are ^{214}Pb and ^{214}Bi from the ^{238}U decay series, and ^{228}Ac, ^{212}Pb and ^{208}Tl from the ^{232}Th decay series. The highest gamma energy (2614 keV) is associated with ^{208}Tl. Exposure to gamma radiation arises mainly from accumulations of mineral concentrates or residues. Dose rates are generally highest near process tanks, piping, filters and large stockpiles of material.
(b) Airborne dust particles arise from the resuspension of contamination on floors and other surfaces, from releases from processing operations and from the conveying of minerals. For inhalation of such particles by workers in industrial activities involving naturally occurring radioactive material, exposure due to radionuclides in the ^{238}U decay series and the ^{232}Th decay series may be of concern for the purposes of radiation protection.

3.169. Consideration of internal exposure via the inhalation of ^{222}Rn emitted from process material — leading to exposure due to its short lived progeny — may be necessary in some activities involving minerals and raw materials (in terms of para. 3.161, such an exposure situation would not necessarily be considered a planned exposure situation). Exposure due to ^{220}Rn and its progeny is not normally of concern because the half-life of ^{220}Rn is much shorter than that of ^{222}Rn. Attention should be given to ^{220}Rn in certain workplaces involving minerals with a high ^{232}Th content, such as monazite. In such workplaces, it is likely that the exposure would, in any case, be controlled as a planned exposure situation rather than as an existing exposure situation because of the need to control exposure due to other radionuclides in the ^{232}Th decay series (see para. 3.161(a)).

3.170. Internal exposure of workers via ingestion is unlikely to require consideration under normal operational circumstances.

Expected exposure levels

3.171. Experience has shown that the annual effective doses received by workers in industrial activities involving naturally occurring radioactive material are often low, even when the concentrations of radionuclides in the ^{238}U decay series and the ^{232}Th decay series are significantly higher than 1 Bq/g. The prior radiological evaluation should therefore be conducted in such a way as to identify quickly which exposure pathways are of significant concern for protection and safety, as opposed to those of minimal concern. For exposure to gamma radiation and exposure due to airborne dust, it is possible to establish an indication in broad terms of the dose to be expected if there is knowledge of the activity concentrations in the various process materials. A methodology for this, which makes use of the underlying linear relationship between dose and activity concentration, is described in Appendix I.

3.172. In the vast majority of workplaces, ^{222}Rn concentrations are similar to normal indoor levels or can be reduced to such levels by means of improved ventilation, in accordance with the requirements for existing exposure situations (see Section 5). In terms of para. 3.161(a), exposure due to ^{222}Rn in workplaces involving naturally occurring radioactive material could become subject to the requirements for planned exposure situations because of the need to control (as a planned exposure situation) exposure due to other radionuclides in the ^{238}U decay series and the ^{232}Th decay series. Even in these workplaces, ^{222}Rn concentrations are still generally close to normal indoor levels because any ^{222}Rn released from minerals with elevated ^{226}Ra concentrations can be readily diluted by means of ventilation. Nevertheless, there are some workplaces with a potential for high ^{222}Rn concentrations. Concentrations of ^{222}Rn are high enough in some cases that, despite all reasonable efforts to reduce them, they remain above the reference level for ^{222}Rn (see para. 5.60), thus becoming subject to control as a planned exposure situation in terms of para. 3.161(b). In all likelihood, such workplaces will be underground workplaces for which there may be limitations on the amount of ventilation possible and/or where there may be a significant release of ^{222}Rn into the air from radium rich minerals (such as in underground uranium mines) or from radium rich water (such as in underground mines and groundwater treatment plants). Concentrations of ^{222}Rn in the workplace tend to be highly variable and exposures are very difficult to predict by modelling. Where the possibility of significantly elevated concentrations of ^{222}Rn is suspected, a ^{222}Rn survey should be conducted in the workplace as part of the prior radiological evaluation in order to determine the extent to which measures for the control of exposure due to ^{222}Rn might be necessary. This is irrespective of whether the

exposure situation is eventually to be treated as a planned exposure situation or as an existing exposure situation.

3.173. Since natural potassium contains 0.012% ^{40}K, this radionuclide is widely present in minerals and raw materials. With a half-life of 1.25 billion years, ^{40}K decays by beta emission to ^{40}Ca (89%) and by electron capture to ^{40}Ar (11%), with the emission of 1.46 MeV gamma radiation. In the body, ^{40}K is homeostatically controlled and any excess is excreted. In the body of an adult, the potassium content is about 160 g. As it is generally accepted that it is not feasible to control ^{40}K in the body, ^{40}K is deemed to be not amenable to control and is excluded from the standards (see para. 1.42 of GSR Part 3 [2]). For purposes of protection and safety, the only possible concern is gamma emission from bulk quantities of material rich in potassium, such as some types of fertilizer. According to data presented in Ref. [24], the annual effective dose per unit activity concentration due to gamma radiation from ^{40}K in potassium rich minerals is expected to be 0.02–0.03 mSv per Bq/g. The activity concentration is always less than 30.6 Bq/g, this being the activity concentration of ^{40}K in pure potassium. The effective dose received by a worker exposed due to potassium rich minerals is therefore always expected to be less than 1 mSv in a year. In view of this, occupational exposure due to ^{40}K in potassium rich minerals can generally be disregarded in any prior radiological evaluation.

Control of exposures of workers

Exposure to gamma radiation

3.174. To minimize external exposure due to naturally occurring radioactive material, specific protection measures in the workplace, such as control of the occupancy period or even shielding, may sometimes be appropriate. Materials with relatively low activity concentrations give rise to modest gamma dose rates (typically no more than a few microsieverts per hour), even on contact. In such cases, discouraging and reducing access, for example by storing materials in mostly unoccupied areas, may be sufficient. In areas containing materials with relatively high activity concentrations, physical barriers and warning signs may be necessary.

Exposure due to dust and other airborne contaminants

3.175. Exposure due to airborne dust is likely to be controlled in many workplaces through general occupational health and safety regulations. Control of the air quality for the purpose of minimizing levels of dust may also help to

reduce concentrations of decay products of ^{222}Rn and ^{220}Rn. Therefore, the extent to which existing control measures for the purposes of occupational health and safety are effective in minimizing workers' radiation exposure is something that the regulatory body should first establish before deciding to impose additional control measures for purely radiological reasons. In some workplaces, existing control measures for the purposes of occupational health and safety alone may provide sufficient protection against internal exposure. In other workplaces, additional control measures specifically for radiation protection purposes may become necessary for achieving compliance with the requirements that apply for planned exposure situations.

3.176. Many workplaces involving exposure due to naturally occurring radioactive material are inherently dusty. Such workplaces include mining areas, ore crushing areas, and product handling and packaging areas. In such workplaces, in particular those that are not open to the atmosphere, ventilation systems are generally crucial for the control of airborne dust. Ventilation systems may also be crucial for the control of ^{222}Rn and its progeny, as well as non-radiological airborne contaminants; in underground mines, these non-radiological contaminants can include methane gas and blasting fumes. The design of ventilation systems for underground mines should be an integral part of overall planning and development of the mine. Where possible, the buildup of ^{222}Rn in underground workplaces should be minimized by avoiding the passage of fresh air through mined out areas and by achieving a 'one pass' system. Air velocities should be high enough to dilute the airborne contaminants but not so high as to cause settled dust to be resuspended. Areas from where the supply of air is drawn should be well separated from the areas where the exhaust air is discharged to avoid mixing of the two air streams. It is preferable to operate the primary ventilation system continuously to avoid the buildup of activity in work areas. Access by workers to any non-ventilated areas should be prevented unless such workers are specially authorized and adequately protected. Placing fixed workstations in return airways should be avoided. Where this is not possible, operator booths with a filtered air supply should be provided.

3.177. In facilities that have a high potential for exposure due to airborne dust, ^{222}Rn or other airborne contaminants, the employer should ensure that the services of a suitably qualified ventilation officer are employed. The ventilation officer should have the following responsibilities:

(a) Advising management on all matters relating to ventilation and air purification systems;

(b) Ensuring the proper operation of the ventilation systems (including auxiliary ventilation systems, which may be prone to rapid deterioration in underground mines), initiating any necessary modifications and ensuring that any deficiencies are promptly addressed;
(c) Ensuring that air flows and velocities are measured in accordance with good ventilation practice;
(d) Ensuring that properly calibrated instruments are used;
(e) Conducting programmes for sampling and control of dust, in conjunction with the radiation protection officer;
(f) Participating in training programmes and developing or approving all training material on ventilation and control of dust;
(g) Being familiar with the properties of ^{222}Rn, ^{220}Rn and their progeny, where applicable.

3.178. Complete containment of material is often impractical, especially where large quantities of materials of low activity concentration are involved. However, spills and the spread of materials outside the area are often of no radiological significance unless significant and persistent levels of airborne dust result. Prevention of resuspension of dust is therefore likely to be the most effective approach. The control of surface contamination may be difficult and impractical, and specific measures to control surface contamination only become meaningful where materials with higher activity concentrations are present. Nevertheless, even where the materials being handled have a low activity concentration, good industrial practice should always be followed, including the establishment of appropriate rules and working procedures (i.e. the use of vacuum cleaning) to ensure that resuspension of dust is adequately controlled. Measures to encourage good general housekeeping, spillage control and personal hygiene should be established and should be kept under review.

3.179. In situations where the radionuclide activity concentrations in the materials being handled are moderate, it should be recognized that the silica content of the airborne dust is likely to be of greater concern on grounds of occupational health than its radionuclide content.

Awareness and training of workers

3.180. Many industrial activities involving naturally occurring radioactive material are not automatically associated with exposure to radiation. The introduction of local rules, and an understanding of the precautions embodied in such rules, should be supported by the awareness and training of workers. Work practices of individual employees may exacerbate dust generation and, in some

cases, may completely negate the effects of any engineered controls installed. There may be deficiencies in the way in which tasks for equipment maintenance are undertaken, implying the need for periodic review to determine whether improvements are possible.

3.181. Programmes of education and training of workers should include topics specific to industrial activities involving naturally occurring radioactive material. Such topics should include, as appropriate:

(a) The properties and hazards associated with radionuclides in the ^{238}U decay series and the ^{232}Th decay series (including ^{222}Rn and ^{220}Rn, where relevant);
(b) The application of the principles of time, distance and shielding to minimize exposure to gamma radiation near large accumulations of naturally occurring radioactive material, especially where activity concentrations are high;
(c) The measurement of airborne activity in the form of dust, and ^{222}Rn and its progeny;
(d) The need for controlling and suppressing airborne dust, and the methods employed;
(e) The functioning and purpose of the ventilation system, and its importance for protection and safety.

4. EXPOSURE OF WORKERS IN A NUCLEAR OR RADIOLOGICAL EMERGENCY

4.1. The requirements for protection and safety for workers in emergency exposure situations are set out in GSR Part 3 [2] and in IAEA Safety Standards Series No. GSR Part 7, Preparedness and Response for a Nuclear or Radiological Emergency [30].

4.2. There are four groups of workers who may be exposed in a nuclear or radiological emergency, owing either to their involvement in the emergency response or to the nuclear or radiological emergency at a facility or an activity itself:

(a) Emergency workers who have specified duties;
(b) Workers performing their duties in workplaces and not being involved in the response to a nuclear or radiological emergency;

(c) Workers who are requested to stop performing their duties in workplaces and to leave the site;
(d) Workers who are accidentally exposed as a result of an accident or other incident at a facility or during the conduct of an activity and whose exposure is not related to the emergency response.

4.3. These four groups are derived from considerations of a wide range of scenarios as well as different duties and responsibilities of workers in a facility or activity (such as designated emergency workers, administrative staff at the site and employees of nearby operational units). The duties of different workers in a nuclear or radiological emergency will differ and appropriate protection strategies should be applied to ensure adequate protection of all workers. Protection of emergency workers as specified in para. 4.2(a) should be provided in line with the requirements set out in GSR Part 3 [2] for emergency exposure situations and in GSR Part 7 [30]. Protection of workers grouped in para. 4.2(b) should be provided in the same way as for workers in planned exposure situations in line with the requirements set out in GSR Part 3 [2]. Protection of workers grouped in para. 4.2(c) should be provided in the same way as for members of the public in emergency exposure situations in line with the requirements set out in GSR Part 7 [30]. Protection of workers who are accidentally exposed (para. 4.2(d)) in connection with medical follow-up and treatment and dose assessment should be in line with GSR Part 3 [2] and GSR Part 7 [30].

4.4. Protection of helpers in an emergency (i.e. members of the public who willingly and voluntarily help in response to a nuclear or radiological emergency) is not specifically addressed in this Safety Guide. However, helpers in an emergency should be registered, should be integrated into the emergency response operations and should be provided with the same level of protection as for emergency workers not designated as such at the preparedness stage, in accordance with GSR Part 7 [30].

EMERGENCY PLANNING AND RESPONSIBILITIES

4.5. Arrangements for the protection of workers in a nuclear or radiological emergency should be included in the emergency plan that is prepared on the basis of the hazard assessment in accordance with GSR Part 7 [30]. The degree of planning should be commensurate with the nature and magnitude of the risks, and the feasibility of mitigating the consequences if an emergency were to occur.

4.6. With regard to the protection of emergency workers, the emergency plan should include the following:

(a) The persons or organizations responsible for ensuring compliance with requirements for protection and safety for workers in a nuclear or radiological emergency, including those for controlling the exposure of emergency workers;
(b) Specified roles and responsibilities of all workers involved in the response to a nuclear or radiological emergency;
(c) Details of adequate protective actions to be taken, personal protective equipment and monitoring equipment to be used, and dosimetry arrangements;
(d) Consideration of access control for workers in a nuclear or radiological emergency on the site.

PROTECTION OF EMERGENCY WORKERS

4.7. The fundamental difference between members of the public and emergency workers in an emergency exposure situation is that members of the public could receive doses unless some action is taken to prevent it, whereas emergency workers will receive doses owing to specified duties assigned to them. Thus, to the extent possible, it is reasonable to continue to treat emergency workers' exposures according to the requirements for planned exposure situations, in accordance with the graded approach, in particular in the later stages of the emergency. The exposure of emergency workers starts with the assignment to undertake a particular action and finishes with completion of the assigned task or declaration of termination of the emergency.

4.8. Protection of emergency workers should include, as a minimum:

(a) Training of emergency workers designated as such in advance;
(b) Providing instructions immediately before their use to those emergency workers not designated as such in advance[8] on how to perform their specified duties under emergency conditions and on how to protect themselves ('just in time training');
(c) Managing, controlling and recording the doses received;

[8] Emergency workers who are not designated as such at the preparedness stage are required to be registered and integrated into the emergency response operations in line with GSR Part 7 [30].

(d) Provision of appropriate, specialized personal protective equipment and monitoring equipment;
(e) Provision of iodine thyroid blocking, where appropriate;
(f) Medical follow-up and psychological counselling, as appropriate;
(g) Obtaining the informed consent of emergency workers to perform specified duties, where appropriate.

Justification

4.9. At the preparedness stage, the protective actions and other response actions to be taken in a nuclear or radiological emergency should be justified. Due consideration should be given to the detriment associated with doses received by the emergency workers who take protective actions and other response actions. There should be a commitment to the justification process by all interested parties (the regulatory body, response organizations and others).

Optimization

4.10. At the preparedness stage, the process of optimization, including the use of reference levels, should be applied to the protection of workers. There should be a commitment to the optimization process by all interested parties (the regulatory body, response organizations and others).

4.11. As part of the process of optimization, reference levels should be established. A reference level should represent the level of dose above which it is judged to be inappropriate to plan to allow exposures to occur and for which protective actions should therefore be planned and optimized. The doses to be compared with the reference levels are usually prospective doses (i.e. doses that might be received in the future, as it is only those future doses that can be influenced by decisions on protective actions). The reference levels are not intended as a form of retrospective dose limit.

4.12. The initial phase of the response to a nuclear or radiological emergency is characterized by a lack of information about the event, a scarcity of materials for protective measures and the need for urgency in implementing protective actions. Therefore, there is little or no scope for applying the optimization process when managing the protection of emergency workers during this initial phase. Efforts should be aimed at reducing any exposures as far as practicable, with account taken of the difficult conditions of the evolving emergency.

4.13. In taking protective actions during the late phase of a nuclear or radiological emergency and at the transition from an emergency exposure situation to an existing exposure situation, the optimization process should be applied to the protection of emergency workers in the same way as for the protection of workers in planned exposure situations.

Restricting exposure of emergency workers

4.14. As the exposure of emergency workers is intentional and controlled, the dose limits for workers should be assumed to apply unless there are overriding reasons not to apply them. As stated in para. 4.15 of GSR Part 3 [2] (see also para. 5.55 of GSR Part 7 [30]):

> "Response organizations and employers shall ensure that no emergency worker is subject to an exposure in an emergency in excess of 50 mSv other than:
>
> (a) For the purposes of saving life or preventing serious injury;
> (b) When undertaking actions to prevent severe deterministic effects and actions to prevent the development of catastrophic conditions that could significantly affect people and the environment; or
> (c) When undertaking actions to avert a large collective dose."

4.15. Guidance values for restricting the exposure of emergency workers should be defined in accordance with the assigned task as provided in appendix I of GSR Part 7 [30] and reproduced in Table 2 of this Safety Guide. Where actions for the purposes of saving life are concerned, every effort should be made to keep individual doses of emergency workers below 500 mSv for exposure to external penetrating radiation, while other types of exposure should be prevented by all possible means. However, in estimating doses to emergency workers, the exposures via all pathways, external and internal, should be assessed and should be included in the total. The value of 500 mSv should be exceeded only under circumstances in which the expected benefits to others clearly outweigh the emergency worker's own health risks, and in which the emergency worker volunteers to take the action and understands and accepts this health risk.

4.16. Regardless of the circumstances, response organizations and employers should make all reasonable efforts to keep the doses received by emergency workers below the thresholds for severe deterministic effects given in GSR Part 3 [2], GSR Part 7 [30] and GSG-2 [31].

TABLE 2. GUIDANCE VALUES FOR RESTRICTING EXPOSURE OF EMERGENCY WORKERS [30]

Task	Guidance value[a]
Lifesaving actions	$H_p(10)$[b] < 500 mSv E[c] < 500 mSv AD_T[d] < ½AD_T[e] This value may be exceeded — with due consideration of the generic criteria in table II.1 of GSR Part 7 [30] — under circumstances in which the expected benefits to others clearly outweigh the emergency worker's own health risks, and the emergency worker volunteers to take the action and understands and accepts these health risks.
Actions to prevent severe deterministic effects and actions to prevent the development of catastrophic conditions that could significantly affect people and the environment	$H_p(10)$[b] < 500 mSv E[c] < 500 mSv AD_T[d] < ½AD_T[e]
Actions to avert a large collective dose	$H_p(10)$[b] < 100 mSv E[c] < 100 mSv AD_T[d] < 0.1AD_T[e]

[a] These values are set to be two to ten times lower than the generic criteria in table II.1 of GSR Part 7 [30] and they apply for:

(a) The dose from external exposure to strongly penetrating radiation for $H_p(10)$. Doses from external exposure to weakly penetrating radiation and from intake or skin contamination need to be prevented by all possible means. If this is not feasible, the effective dose and the RBE weighted absorbed dose to a tissue or organ have to be limited to minimize the health risk to the individual in line with the risk associated with the guidance values given here.

(b) The total effective dose E and the RBE weighted absorbed dose to a tissue or organ AD_T via all exposure pathways (i.e. both dose from external exposure and committed dose from intakes) which are to be estimated as early as possible in order to enable any further exposure to be restricted as appropriate.

[b] Personal dose equivalent $H_p(d)$ where d = 10 mm.

[c] Effective dose.

[d] RBE weighted absorbed dose to a tissue or organ.

[e] Values of RBE weighted absorbed dose to a tissue or organ given in table II.1 of GSR Part 7 [30].

4.17. When military personnel are designated as emergency workers, every effort should be made to ensure that they are protected in the same way as other emergency workers.

MANAGING THE EXPOSURE OF EMERGENCY WORKERS

4.18. In terms of para. 4.12 of GSR Part 3 [2], the government is required to establish a programme for managing, controlling and recording the doses received by emergency workers in a nuclear or radiological emergency. Response organizations and employers should implement this programme.

4.19. The group of emergency workers specified in para. 4.2(a) can be further divided into three categories of emergency worker:

(a) Category 1. Emergency workers undertaking mitigatory actions and urgent protective actions on the site, including lifesaving actions, actions to prevent serious injury, actions to prevent the development of catastrophic conditions that could significantly affect people and the environment, actions to prevent serious deterministic effects and actions to avert a large collective dose. Emergency workers in Category 1 are required to be designated as such at the preparedness stage. They are likely to be operating personnel at the facility or undertaking the activity, but they may be personnel from the emergency services. They are employed either by a registrant or licensee (operating organization) or by a response organization, and they should receive training in occupational radiation protection.

(b) Category 2. Emergency workers undertaking urgent protective actions off the site (e.g. evacuation, sheltering and radiation monitoring) to avert a large collective dose. They are most likely to be police, firefighters, medical personnel, and drivers and crews of evacuation vehicles. Every effort should be made to designate emergency workers in Category 2 as such at preparedness stage. They are to have pre-specified duties in an emergency response and should receive training in occupational radiation protection on a regular basis as first responders. They are not normally considered to be occupationally exposed to radiation, and their employers are response organizations.

(c) Category 3. Emergency workers undertaking early protective actions and other response actions off the site (e.g. relocation, decontamination and environmental monitoring) as well as other actions aimed at enabling the termination of the emergency. Emergency workers in Category 3 may or may not be designated as such at the preparedness stage. They may or may not normally be considered to be occupationally exposed to radiation, and they may or may not have received any relevant training, including training in radiation protection.

4.20. Any limit in the duration of work undertaken by emergency workers and any conditions on which they will conduct the work should be applied by planning the emergency work on the basis of guidance values of dose.

4.21. Tasks should be assigned, depending on the category of emergency worker, as follows:

(a) Category 1 emergency workers should carry out actions to save lives or prevent serious injury, actions to prevent severe deterministic effects and actions to prevent the development of catastrophic conditions that could significantly affect people and the environment.
(b) Category 2 emergency workers should not be the first choice for taking lifesaving actions.
(c) Category 1 and Category 2 emergency workers should carry out actions to avert a large collective dose.
(d) Category 3 emergency workers should carry out those actions in which they will not receive a dose of more than 50 mSv.[9]

4.22. In almost all emergencies, at best only the dose from external penetrating radiation will be measured continuously. Consequently, the operational guidance provided to emergency workers should be based on measurements of penetrating radiation (e.g. as displayed on an active or self-reading dosimeter). Doses from intakes, skin contamination and exposure of the lens of the eye should be prevented by all possible means, for instance by the use of personal protective equipment, iodine thyroid blocking (where exposure due to radioactive iodine might be involved) and by the provision of instructions concerning operations in potentially hazardous radiological conditions. Such instructions should cover the application of time, distance and shielding principles, the prevention of ingestion of radionuclides and the use of respiratory protection. Available information about radiological conditions on the site should be used to aid decisions on the appropriate protection of emergency workers.

4.23. Female workers who are aware or who suspect that they are pregnant or who are breast-feeding should be encouraged to notify their employer and they should typically be excluded from tasks in an emergency unless such tasks can be carried out within the requirements for occupational exposure set out in paras 3.114 and 4.15 of GSR Part 3 [2]. Female workers designated as emergency workers prior to an emergency and who are aware or who suspect that they are

[9] Helpers in an emergency should not be allowed to take actions that might result in their exceeding an effective dose of 50 mSv.

pregnant or who are breast-feeding during the emergency may volunteer to take emergency duties as long as para. I.4 of GSR Part 7 [30] is applied.

4.24. Response organizations and employers are required to ensure that emergency workers who undertake actions in which the doses received might exceed 50 mSv (see para. 4.14 of this Safety Guide, para. 4.17 of GSR Part 3 [2] and para. 5.57 of GSR Part 7 [30]) do so voluntarily. Such emergency workers should be clearly and comprehensively informed in advance of the associated health risks, as well as of available protective measures, and they should be trained, to the extent possible, in the actions that they are required to take. The voluntary basis for response actions by emergency workers is usually covered in the emergency arrangements.

4.25. Workers should not normally be precluded from incurring further occupational exposure because of doses received in an emergency.

ASSESSMENT OF EXPOSURE

4.26. Response organizations and employers should take all reasonable steps to assess and record the exposures received by workers in an emergency. Once the total dose received by emergency workers via all exposure pathways (including the committed doses from intakes) has been estimated, the guidance provided in Table 2 for the effective dose and the RBE weighted absorbed dose to a tissue or organ should be used for restricting further exposure in response to a nuclear or radiological emergency. The doses from exposures of emergency workers in an emergency response and from exposures of workers who are accidentally exposed (see para. 4.2(d)) should, if possible, be recorded separately from those doses incurred during routine work, but should be noted in the workers' records of occupational exposure.

4.27. The degree of accuracy required for any assessment of exposure should increase with the level of exposure likely to have been received by the worker. Some pre-established guidance can help in the management of exposures of emergency workers in Category 1, expressed in terms of dose and directly measurable quantities such as dose rate or air concentration. The exposures of emergency workers should be monitored on an individual basis, by using means appropriate to the situation, such as direct reading dosimeters or alarm dosimeters.

4.28. Records of occupational exposure should be generated and maintained in a simplified standard format by all response organizations and employers to avoid

confusion. Information on the doses received and on the associated health risks should be communicated to the emergency workers involved.

4.29. The guidance given in paras 7.222 and 7.223 may also be relevant for emergencies.

MEDICAL ATTENTION

4.30. Emergency workers and accidentally exposed employees should receive medical attention that is appropriate for the doses that they may have received (see paras 10.29–10.34). Screening based on equivalent doses to specific radiosensitive organs as a basis for medical follow-up and counselling should be provided if an emergency worker or an accidentally exposed employee has received an effective dose exceeding 100 mSv over a period of a month or if the worker so requests. Although an emergency worker or an accidentally exposed employee who receives doses in a nuclear or radiological emergency should normally not be precluded from incurring further occupational exposure, qualified medical advice should be obtained before allowing further occupational exposure where a person has received an effective dose exceeding 200 mSv. Such advice should also be made available at the request of the worker. Such qualified medical advice is intended to assess the continued health and fitness of the worker in line with GSR Part 3 [2] and GSR Part 7 [30].

4.31. A particular concern should be whether a worker has received a dose sufficient to cause severe deterministic effects. If the dose received by the worker exceeds the thresholds for severe deterministic effects specified in table IV.1 of GSR Part 3 [2] and table II.1 of GSR Part 7 [30], protective actions and other response actions should be taken in accordance with GSR Part 7 [30]. Such actions can include:

(a) Performing an immediate medical examination, consultation and indicated treatment;
(b) Carrying out control of contamination;
(c) Carrying out immediate decorporation[10] (if applicable);
(d) Carrying out registration for longer term medical follow-up;
(e) Providing comprehensive psychological counselling.

[10] Decorporation means the biological processes, facilitated by a chemical or biological agent, by which incorporated radionuclides are removed from the human body.

4.32. Additional information relating to the medical response in emergencies can be found in GSG-2 [31] and Refs [32, 33].

5. EXPOSURE OF WORKERS IN EXISTING EXPOSURE SITUATIONS

5.1. As stated in para. 5.1(a) of GSR Part 3 [2]:

> "The requirements for existing exposure situations…apply to:
>
> (a) Exposure due to contamination of areas by residual radioactive material deriving from:
> (i) Past activities that were never subject to regulatory control or that were subject to regulatory control but not in accordance with the requirements of these Standards;
> (ii) A nuclear or radiological emergency, after an emergency has been declared to be ended…."

5.2. The exposure referred to in para. 5.1 may be incurred directly from the residual radioactive material itself, or may be incurred indirectly from commodities that incorporate radionuclides arising from the residual radioactive material. Such commodities include food, feed, drinking water and construction materials. The radionuclides in the residual radioactive material may be radionuclides of artificial origin or radionuclides of natural origin.

5.3. Contamination of areas can also arise from facilities and activities that are subject to regulatory control in terms of the requirements for planned exposure situations, as a result of authorized activities such as discharges, the management of radioactive waste and decommissioning. An exposure situation resulting from such contamination is controlled as part of the overall practice and is, therefore, a planned exposure situation and not an existing exposure situation.

5.4. In terms of para. 5.1(c) of GSR Part 3 [2], the requirements for existing exposure situations also apply, in general, to exposure due to natural sources, where such exposure is not excluded from the scope of GSR Part 3 [2] (see para. 2.4 of this Safety Guide). Natural sources include:

(a) Materials (in a natural or processed state) in which the radionuclides are essentially all of natural origin;
(b) ^{222}Rn and ^{220}Rn, together with their progeny, as specified in para. 3.161;
(c) Cosmic radiation.

5.5. Measures for preventing or reducing doses that might otherwise occur in an existing exposure situation may take the form of remedial actions or protective actions:

(a) Remedial actions in an existing exposure situation involve removal of the source or reduction of its activity or amount. An example of a remedial action is the removal of residual radioactive material from a contaminated site.
(b) Protective actions in an existing exposure situation involve measures that act on the exposure pathways rather than on the source itself. Examples of protective actions are the control of access to a contaminated site and restrictions on the use of contaminated water for drinking purposes.

5.6. Categories of exposure in existing exposure situations are occupational exposure and public exposure. In considering occupational exposure, two groups of exposed workers can be identified:

(a) Workers who are exposed while carrying out remedial actions. The exposures of these workers may be increased as a direct result of their work (i.e. when such action involves the handling, transport or disposal of residual radioactive material).
(b) Workers who are exposed in the existing exposure situation but who do not carry out any remedial action. The exposures of these workers might eventually be reduced as a result of remedial or protective actions.

5.7. The doses received in existing exposure situations are expected to be well below the threshold for deterministic health effects. Therefore, stochastic health effects are the only health effects of concern.

PROTECTION STRATEGIES

5.8. According to paras 5.2 and 5.3 of GSR Part 3 [2], the government has certain responsibilities with regard to existing exposure situations. It is required to ensure that existing exposure situations are identified and evaluated to determine which exposures (including occupational exposures) are of concern from the point of

view of radiation protection. It is also required to make provision in the legal and regulatory framework for the management of exposures of concern, including the assignment of responsibilities for protection and safety, the establishment of appropriate criteria for protection and safety in the form of reference levels (see paras 5.19–5.23) and the making of decisions on the reduction of exposures by means of remedial and/or protective actions.

5.9. Where it is decided that exposures need to be reduced, appropriate protection strategies need to be established. Formal provision for the development and implementation of protection strategies is required to be made by the government in the legal and regulatory framework, including the following:

(a) Specification of the general principles underlying the protection strategies;
(b) Assignment of responsibilities for the development and implementation of the protection strategies to the regulatory body or other relevant authority (e.g. a health authority or an environmental protection agency)[11] and to the parties involved in the implementation process;
(c) Provision for the involvement of interested parties in the decision making process, as appropriate.

5.10. In terms of the graded approach (see para. 2.20), the government, in conjunction with the regulatory body or other relevant authority identified in para. 5.9(b), should ensure that protection strategies for existing exposure situations are commensurate with the associated radiation risks.

5.11. In terms of para. 5.4 of GSR Part 3 [2], the regulatory body or other relevant authority is required to ensure that the protection strategy for a particular existing exposure situation defines the objectives to be achieved and includes appropriate reference levels (see paras 5.19–5.23).

5.12. Various remedial and protective actions will generally be available for achieving the objectives of the protection strategy for a particular existing exposure situation. In terms of para. 5.5 of GSR Part 3 [2], the regulatory body or other relevant authority, in implementing the protection strategy, is required to make arrangements for these remedial actions and protective actions to be evaluated. This evaluation will include an evaluation of the effectiveness of those actions eventually planned and implemented.

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[11] More than one authority may be involved, in which case the term ‘authority’ refers to the system of authorities.

5.13. The regulatory body or other relevant authority, in implementing the protection strategy, is required to ensure that information is available to exposed individuals on the potential health risks associated with the exposure and on the means available for reducing their exposures and associated risks.

JUSTIFICATION

5.14. The regulatory body or other relevant authority should establish the protection strategy for a particular existing exposure situation in accordance with the principle of justification. This means that only those remedial actions and/or protective actions that are expected to yield sufficient benefits to outweigh the detriments associated with taking them, including detriments in the form of radiation risks, the costs of such actions and any harm or damage caused by the actions, should be considered for inclusion in the protection strategy.

5.15. The detriments in the form of radiation risks to be considered in the justification process should include exposures of workers engaged in taking any remedial actions.

OPTIMIZATION

General approach

5.16. The regulatory body or other relevant authority and other parties responsible for the establishment of a protection strategy should ensure that the form, scale and duration of remedial and protective actions are optimized (i.e. they will provide the maximum net benefit, in that all exposures are controlled to levels that are as low as reasonably achievable, economic, societal and environmental factors being taken into account). The implementation of the optimized protection strategy will not necessarily result in the greatest reduction in dose, since dose reduction is only one of several attributes considered in the optimization process.

5.17. As in the case of the justification process (see paras 5.14 and 5.15), the detriments in the form of radiation risks to be considered in the optimization process should include the exposures of workers engaged in any remedial actions.

5.18. Optimization of protection in an existing exposure situation is achieved by:

(a) An evaluation of the exposure situation, including any potential exposure;

(b) Identification of the possible protection options expressed in terms of justified remedial actions and/or protective actions;
(c) Selection of the best option under the prevailing circumstances;
(d) Implementation of the selected option.

Reference levels

5.19. A reference level should be used in the optimization process. It represents a level of dose above which it is judged to be inappropriate to plan to allow exposures to occur. In considering the various possible remedial actions and protective actions, a reference level serves as an upper bound on the range of options considered; this will ensure that the optimized protection strategy will be aimed at reducing doses to some value below the reference level.

5.20. A reference level also serves as a tool for prioritizing the implementation of remedial actions and protective actions. When an existing exposure situation has been found, actual exposures could be above or below the reference level. While the process of optimization is intended to provide optimized protection for all exposed individuals, priority should be given to those groups receiving doses above the reference level by taking all reasonable steps to reduce those doses to below the reference level.

5.21. Reference levels are generally expressed in terms of annual effective dose to the representative person in the range of 1–20 mSv. However, reference levels for exposure due to radon are expressed in terms of annual average radon concentration in air.

5.22. A reference level for a particular existing exposure situation should be established by the government or by a regulatory body or other relevant authority acting on behalf of the government. The value should be chosen by taking into account all relevant factors, including:

(a) The nature of the exposure and the practicability of reducing the exposure;
(b) Societal implications;
(c) National or regional factors;
(d) Past experience in the management of similar situations;
(e) International guidance and good practice elsewhere.

5.23. The regulatory body or other relevant authority should review reference levels periodically to ensure that they remain appropriate in the light of the prevailing circumstances.

EXPOSURE ARISING FROM REMEDIAL ACTIONS IN AREAS WITH CONTAMINATION BY RESIDUAL RADIOACTIVE MATERIAL

Application of the requirements for protection and safety

5.24. As stated in para. 5.6(a), workers carrying out remedial actions in connection with areas with contamination by residual radioactive material may be subjected to increased exposure as a result of activities such as the handling, transport or disposal of residual radioactive material. According to para. 5.26 of GSR Part 3 [2], the employers of such workers are required to ensure that exposures of the workers are controlled in accordance with the relevant requirements for planned exposure situations as established in section 3 of GSR Part 3 [2]. The guidance given in Section 3 of this Safety Guide is therefore applicable to such workers. The guidance given in paras 5.38, 5.39 and 5.42–5.44 is also relevant.

5.25. As stated in para. 5.6(b), workers who are not carrying out remedial actions may nevertheless be exposed in an existing exposure situation as a result of exposure levels in their workplaces being affected by the residual radioactive material. Exposures of such workers are also subject to control, in the sense that such exposures may be reduced as a result of remedial actions. The requirements for protection and safety under which this control is exercised are the same as those for controlling exposures of members of the public in existing exposure situations. In essence, therefore, the exposures of such workers are controlled as though they were members of the public. Guidance on the reduction of exposures by remedial actions, together with any necessary post-remediation activities, is given in paras 5.28–5.44. More detailed guidance is given in IAEA Safety Standards Series No. WS-G-3.1, Remediation Process for Areas Affected by Past Activities and Accidents [34].

Protection strategies

5.26. In formulating protection strategies for areas with contamination by residual radioactive material, all areas with, or potentially with, contamination should be monitored or surveyed by the regulatory body or other relevant authority so that those areas requiring remedial actions and/or protective actions can be identified and appropriate reference levels can be specified. It will be necessary to involve a number of governmental and private organizations, and provision should be made for liaison between them and for their input to the process. Account should be taken of any possible effects on neighbouring States.

5.27. The regulatory body or other relevant authority should establish safety criteria for the development and implementation of protection strategies, including criteria and methods for assessing the effectiveness of any remedial measures and criteria specifying conditions on the end points of the remediation.

Organizational arrangements for remedial actions

5.28. The organizational arrangements for remedial actions, funding mechanisms and roles and responsibilities, including the legal and regulatory framework, should be in accordance with the guidance provided in WS-G-3.1 [34].

Roles and responsibilities

5.29. Since the remediation of an area with contamination can involve several entities that include individuals who may be unfamiliar with the requirements for radiation protection and safety, the roles and responsibilities of the different parties involved in the remediation process should be clearly defined in the legal and regulatory framework. In particular, responsibilities should be specified for the protection of workers in the planning and implementation of the remediation programme.

5.30. Those persons or organizations responsible for providing adequate human resources, equipment and supporting infrastructure for occupational radiation protection in accomplishing the remediation should be clearly specified.

Regulatory considerations

5.31. The legal and regulatory framework, supported where necessary by guidance material, should provide for adequate protection of individuals (including workers) and the environment when remediation is undertaken.

5.32. Protective actions in the form of restrictions on the use of, or access to, the area should be considered before, during and, if necessary, after remediation. The basis for establishing such restrictions should be provided in the legal and regulatory framework.

5.33. The regulatory process for situations of remediation involves more than just radiation protection. Other laws and regulations covering such matters as occupational health and safety, environmental protection, land management, and food and drinking water standards are likely to be administered by different

governmental bodies. These other laws and regulations should be applied, as appropriate, to create a coherent regulatory approach.

Remediation programme

5.34. Remediation of an area with contamination involves the prior radiological evaluation of the situation, the preparation and approval of a remediation plan, the remediation work itself, and the management of radioactive waste arising from the remediation activities. In the prior radiological evaluation, the nature of the problem and the associated concerns in relation to radiation protection of workers should be appropriately characterized.

5.35. As part of developing a remediation plan, the following aspects, among others, relevant to protection of workers should be considered:

(a) Determining the nature and extent of the contamination;
(b) Identifying exposure pathways for workers;
(c) Assessing individual doses via all routes of exposure;
(d) Evaluating health and safety issues during remediation work, including the use of appropriate personal protective equipment.

5.36. The design of the site characterization survey is determined by the conditions in the area, the type and extent of on-site contamination and the available resources. It should be ensured that the most suitable instruments and sampling and measurement techniques are selected, and that proper attention is given to the calibration of instruments and the recording of data (see Section 7). Collection of data will most probably necessitate measurements of ambient gamma radiation as well as collection and measurement of samples of surface and subsurface soil, airborne radioactive material, water and biota.

5.37. The remedial actions and protective actions that are to be implemented should be justified and optimized (see paras 5.14–5.23). Priority should be given to situations in which the applicable reference level is exceeded (see para. 5.20). Decisions on remedial actions and protective actions should be made with the involvement of the relevant parties concerned with the situation of contamination. Protection and safety considerations should take account of future generations as well as of the present generation, including workers.

5.38. In the justification process, the positive attributes of remediation that should be taken into account include not only the eventual reductions in individual doses and collective doses, but also the expected reductions in anxiety among

individuals, including workers. The negative attributes that should be taken into account include not only the direct financial costs of the remediation, but also the societal and economic costs, the health and environmental impacts of the remediation work (including the radiation risks to workers undertaking such work), and the disruptive effects of the remediation work on society. While the overall objective is to reduce the doses received by individuals, the remediation work itself might temporarily give rise to additional doses. Such additional doses should be justified on the basis of the resulting net benefit, including consideration of the consequent reduction of the annual dose.

5.39. In the optimization process, remedial actions and protective actions should be optimized in accordance with the same general approach as that used for the optimization of protection in planned exposure situations (see paras 3.8–3.33), with the role of the reference level being, in some respects, equivalent to the role of the dose constraint in planned exposure situations. The optimum nature, scale and duration of remedial actions and protective actions should be selected from a set of justified options for remediation. In choosing the optimized remediation option, radiological impacts on individuals and on the environment should be considered together with non-radiological impacts, as well as technical, societal and economic factors. Factors relating to radioactive waste management should also be taken into account. These include the costs (including transport costs) of waste management, the radiation exposure of, and the health risks to, the workers managing the waste, and any subsequent exposure associated with disposal of the waste. In some cases, the outcome of the optimization process for remediation may be one in which the use of human habitats is subject to certain restrictions, in which case institutional controls to enforce those restrictions should be continued.

5.40. The remediation plan should include a monitoring programme to ensure that all necessary information on radiological conditions is gathered before, during and after the remediation process. To ensure that the remediation programme is adequately documented, a system of record keeping should also form part of the remediation plan, and should include the following:

(a) Descriptions of activities performed;
(b) Data from monitoring and surveillance programmes;
(c) Records of occupational health and safety for remediation workers;
(d) Records of the types and quantities of radioactive waste generated and of its management;
(e) Data from environmental monitoring;
(f) Records of financial expenditures;
(g) Records of the involvement of interested parties;

(h) Records of any continuing responsibilities for the site;
(i) Identification of locations that were remediated and those with residual contamination;
(j) Specifications of any areas to which access remains restricted and the restrictions that apply;
(k) Statements of any zoning or covenant restrictions or conditions;
(l) Statements of lessons identified.

5.41. Procedures should be established to ensure that any abnormal conditions relevant to protection and safety are reported to the regulatory body or other relevant authority. Individuals, including workers, should be kept informed and parties affected by the situation should be involved in the planning, implementation and verification of the remedial actions and of any post-remediation monitoring and surveillance. The remediation plan, supported by the prior radiological evaluation, should be submitted to the regulatory body or other relevant authority for approval. Its approval, depending on the circumstances, might involve the issue of an authorization in the form of a registration or licence, as might be required for a planned exposure situation (see para. 3.3).

Taking remedial actions

5.42. Throughout the remedial actions, the responsible person or organization should take overall responsibility for protection and safety, even when contractors are used to perform specific tasks or functions. This overall responsibility includes responsibility for protection and safety in the transport, processing, storage and disposal of the radioactive waste arising from the remediation work. Carrying out (and submission to the regulatory body or other relevant authority for approval) a safety assessment and, where appropriate, an environmental assessment, as well as any follow-up assessments, forms part of this responsibility. As explained in para. 5.24, although the remedial actions are undertaken in an existing exposure situation, the exposure of workers undertaking the remediation work should be controlled in accordance with the relevant requirements for occupational exposure in planned exposure situations. This places various obligations on the employer of the workers, such as the following:

(a) Preparing and applying appropriate protection and safety procedures;
(b) Applying good engineering practice;
(c) Ensuring that the staff are adequately trained, qualified and competent;
(d) Ensuring that protection and safety is integrated into the overall management system.

5.43. If the employer of the workers engaged in the remediation work is an outside contractor, the person or organization responsible for the remedial actions should cooperate with that employer, to the extent necessary, for compliance by both parties with the applicable requirements for protection and safety (see paras 6.21–6.100).

5.44. During the remedial actions, the regulatory body or other relevant authority is responsible for verifying day to day compliance with regulatory requirements, including requirements for protection against occupational exposure. This verification of compliance involves carrying out regular inspections and a review of work procedures, monitoring programmes and monitoring results. There are also responsibilities associated with non-routine matters, including enforcement actions in the event of non-compliance; responses (where necessary) to reports of abnormal occurrences; and the review and approval of any changes to procedures, to equipment or to the remediation plan itself, when such changes may have significant implications in terms of radiological considerations for workers, the public or the environment.

EXPOSURE DUE TO RADON

Exposure pathways

5.45. Uranium occurs naturally in rocks and soil. The decay of ^{226}Ra in the ^{238}U decay series results in the production of the radon isotope ^{222}Rn, an inert, naturally occurring radioactive gas with a half-life of 3.8 d. Some of this radon gas escapes to the air, while some dissolves in groundwater. The highest ^{222}Rn concentrations in air are found in enclosed spaces, the levels of which depend on the rate of ingress and the level of ventilation. Exposure of individuals to ^{222}Rn and its short lived progeny (^{218}Po, ^{214}Pb, ^{214}Bi and ^{214}Po) occurs mainly by breathing air, resulting in a dose to the lung. Only about 1% of the dose to the lung arises from ^{222}Rn itself, because most of the inhaled gas is exhaled. The dose arises almost entirely from the short lived progeny, atoms of which attach themselves to condensation nuclei and dust particles present in the air. These particles, as well as unattached particles, are deposited along the various airways of the bronchial tree. Exposure of the lung is caused mainly by the alpha particles emitted by the short lived progeny, even though there are also some emissions of beta particles and gamma radiation. Exposure from the ingestion of ^{222}Rn via the groundwater pathway is unlikely to be of significant concern for occupational radiation protection.

5.46. A similar situation pertains with respect to thorium in rocks and soil, with the decay of ^{232}Th resulting in the production of the gaseous isotope ^{220}Rn (commonly referred to as thoron). However, exposure due to ^{220}Rn and its short lived progeny is unlikely to be of concern in existing exposure situations because the half-life of ^{220}Rn (56 s) is much shorter than that of ^{222}Rn. The inhalation of ^{220}Rn and of its progeny by workers in the mining and processing of minerals with high thorium contents could give rise to exposures of concern. However, in such situations, these exposures would be controlled as a planned exposure situation, together with exposures due to other radionuclides in the ^{232}Th decay series (see para. 3.169). Consequently, the use of the term 'radon' hereinafter refers only to the isotope ^{222}Rn.

Radon concentrations

Buildings

5.47. In buildings, the accumulation of radon in the air occurs mainly as a result of the entry of radon directly from the underlying soil into the basement through cracks in the floor. In temperate zones, the air inside buildings is normally at a slightly lower pressure than the air outdoors as a consequence of the air inside the building being warmer than the air outside. This causes a convective flow which, together with the effect of wind blowing across chimneys and other openings, draws soil gas and, hence, radon into the building. In addition to pressure differences, other factors, such as relative humidity and soil moisture, can also influence radon concentrations in buildings.

5.48. The accumulation of radon in buildings also occurs, usually to a lesser extent, through the escape of radon from building materials into the air inside the building, particularly if such materials are porous and have elevated concentrations of ^{226}Ra. The water supply can also provide a route for the entry of radon into the air inside buildings, although radon concentrations in domestic water supplies are generally quite low except, possibly, when the supply comes from groundwater. The accumulation of radon in buildings that are workplaces may also be influenced by the presence of minerals and raw materials containing elevated concentrations of radionuclides in the ^{238}U decay series, although this influence is generally quite small if there is adequate ventilation (see para. 3.172).

5.49. Indoor radon concentrations differ between countries because of differences in geology, climate, construction materials, construction techniques, types of ventilation provided (whether natural or otherwise) and domestic habits. Within countries, there may be marked regional variations. Data on indoor

radon concentrations around the world are given in Ref. [35]. The arithmetic mean values reported for various countries vary from 7 Bq/m^3 to 200 Bq/m^3. Reported arithmetic mean values in high background areas are in the range of 112–2745 Bq/m^3. In some parts of northern Europe, maximum values of up to 84 000 Bq/m^3 are reported. The population weighted worldwide arithmetic mean is reported as 39 Bq/m^3.

Underground workplaces

5.50. The highest concentrations of radon tend to occur in underground workplaces. Such workplaces include underground mines, tunnels, basement storage and parking facilities, underground facilities for water treatment and distribution, caves, former mines open to the public and spas. In such workplaces, there are many interfaces through which there may be substantial entry of radon into the air, and there may be practical limitations on the amount of ventilation that can be provided. In some underground mines, including some in which the ^{226}Ra concentrations in the rock are not significantly elevated, high concentrations of radon arise from the entry of radon via groundwater and its subsequent release into the mine's atmosphere. A similar situation can be encountered in underground facilities for water treatment and distribution.

5.51. Concentrations of radon are reported to vary from 20 Bq/m^3 to more than 20 000 Bq/m^3 in workplaces in caves and underground mines open to the public and from about 200 Bq/m^3 to 7000 Bq/m^3 in workplaces in tunnels [36]. Much higher values have been found in some operating underground mines, particularly uranium mines.

Application of the requirements for protection and safety

5.52. As with any other exposure due to natural sources, occupational exposure due to radon is subject to the requirements for existing exposure situations (para. 5.1(c)(i) of GSR Part 3 [2]). However, the requirements for planned exposure situations will apply in certain situations, as specified in para. 3.161.

5.53. Occupational exposure due to radon is generally of concern only in enclosed workplaces such as buildings and underground mines. Occupational exposure due to radon outdoors is not usually of concern except, possibly, in open pit mines in certain atmospheric conditions.

Identifying workplaces in which exposure due to radon is of concern

5.54. The government should ensure that information is gathered on indoor concentrations of radon, including concentrations in workplaces. Since it is not feasible to measure radon concentrations in every workplace, surveys should be designed and carried out such that the information gathered is reasonably representative of the country as a whole, in a similar manner to surveys of radon in homes. This requires that the surveys be systematic and unbiased to the extent possible. Geographical considerations will often be a good general guide to identifying areas in which radon concentrations are likely to be above average. However, such an approach on its own has limitations because the relationships between indoor radon concentration and geological parameters, such as soil porosity and concentrations of uranium and radium, are complex. Geological considerations can nevertheless be used for interpolating between the survey results and may be useful in refining the identification of the relevant areas.

5.55. Radon concentrations measured in above ground workplaces could provide important input to the identification of radon prone areas for dwellings, or vice versa, since it is likely that radon prone areas for above ground workplaces will coincide with those for dwellings.[12]

5.56. Once the data from measurements have been gathered, the government should ensure that analysis of the data leads to the identification of any workplaces in which exposure due to radon is of concern. If there are no such situations, no further action is required. If, on the other hand, exposures of concern are identified, the government should ensure that action on exposures in workplaces is incorporated into an overall national action plan for indoor radon. The action plan should be appropriate for the exposure situation and should be adapted to national conditions.

Action plan

5.57. A national action plan for exposure due to indoor radon, including exposure in workplaces, should provide the means for defining remedial actions to address exposures of concern. It should also provide the means for ensuring that, by way of suitable campaigns, relevant information on exposure due to radon is provided

[12] A radon prone area is one in which, because of the characteristics of the ground or of the building design and usage, the percentage of buildings with ^{222}Rn concentrations above a certain predetermined level (most probably the applicable reference level) exceeds a threshold percentage level established by the regulatory body or other relevant authority.

to employers, workers and members of the public, and to other interested parties such as professional bodies. The objective of these information campaigns should be to share the key findings of the national surveys and to increase the understanding of radon, the potential health risks and the simple measures that can be taken to reduce the risks. Since smoking is such a prevalent cause of lung cancer, the increased risks relating to smoking and radon should be highlighted.

5.58. For exposures in workplaces that are identified as being of concern, the action plan should specify a series of coordinated actions to address radon concentrations in existing workplaces and future workplaces.

5.59. It is possible to focus the efforts to control radon by identifying 'radon prone buildings'. Such buildings can be identified on the basis of certain characteristics of the design, construction material or construction method that are likely to give rise to elevated radon concentrations.

Reference levels

5.60. In formulating the action plan, appropriate reference levels for radon in workplaces should be established, account being taken of the prevailing societal and economic circumstances. In general, the reference level for workplaces is required to be set at a value that does not exceed an annual average radon concentration of 1000 Bq/m^3 (para. 5.27 of GSR Part 3 [2]). This value corresponds to an annual effective dose of the order of 10 mSv, on the assumption of an equilibrium factor of 0.4 and an annual occupancy period of 2000 h. There is a practical advantage in adopting a single value for the reference level that applies to all workplaces irrespective of the equilibrium factor. Nevertheless, other reference levels may be appropriate if the equilibrium factor is significantly different from this, which may be the case in some underground mines, for instance. The choice of an appropriate reference level is complex — the value should be determined with considerable circumspection, with account taken not only of the level of exposure but also of the likely scale of remedial action involved, which has economic implications for industry and for the State as a whole. In buildings with high occupancy factors for members of the public, such as kindergartens, schools and hospitals, exposure of all occupants is controlled using the reference level for dwellings (para. 5.20 of GSR Part 3 [2]).

Remedial actions in workplaces

5.61. In workplaces that have been identified in the action plan, additional, more detailed measurements of radon concentrations may be necessary. Arrangements

for making these measurements, and for carrying out any subsequent remedial action, are required to be the responsibility of the employer concerned. The employer should have access to expert advice on remedial measures. It may be appropriate for the regulatory body or other relevant authority to provide written guidance in accordance with national building practices.

5.62. The employer should ensure that radon activity concentrations in workplaces are as low as reasonably achievable, with priority being given to those workplaces in which the reference level is exceeded. In some workplaces, particularly underground mines, there can be large variations in radon concentration in space and time. This should be taken into account when determining whether the reference level is exceeded.

5.63. If, despite all reasonable efforts by the employer to reduce radon concentrations in the workplace, such concentrations remain above the reference level, the relevant requirements for occupational exposure in planned exposure situations will apply (see para. 3.161(b)). This outcome is unlikely except for some underground mines where there might be practical limitations on restricting the entry of radon into the air and on the amount of ventilation that can be provided (see para. 3.176).

Methods for reducing radon concentrations in buildings

Subfloor depressurization

5.64. For foundations and basements in contact with soil, the most effective course of action for reducing radon concentrations is to reduce the pressure of the soil gas in the vicinity of the foundation relative to the pressure in the structure. This can be accomplished by installing a system of pipes leading from the soil under the foundation that maintains a negative pressure gradient between the soil and the foundation. The soil gas containing radon can then be vented harmlessly to the atmosphere. Where possible, it is desirable to install a small and simple cavity or sump within the foundations to which the system of pipes can be attached. For buildings with extensive and complex foundations, a number of such depressurization systems may be needed.

Subfloor ventilation

5.65. If the ground floor is not in contact with the soil, the amount of radon entering the structure can be reduced by ventilating the space beneath the floor.

This can be accomplished by increasing the natural ventilation or by installing a fan that removes air from under the floor and replaces it with outdoor air.

Floor sealing and membranes

5.66. Since most of the radon that enters from the soil enters through cracks and other openings in the floor, it is possible to reduce indoor radon concentrations by sealing such entry routes. However, this approach is generally less effective than depressurization and ventilation because it is difficult to seal all entry routes adequately and seals deteriorate over time. It can be used as a supplementary measure to increase the effectiveness of subfloor depressurization or ventilation. Heavy duty plastic membranes incorporated into the foundations can act as effective radon barriers provided that all joints are properly sealed and the membranes are not punctured during installation. They cannot be retrofitted to existing buildings, however.

Increased ventilation

5.67. Indoor radon can be diluted by means of increased ventilation with outside air. This approach can be costly in terms of energy loss, particularly in hot or cold climates. Energy loss can be reduced by means of heat exchangers, but these involve significant capital, operating and maintenance costs. In some structures, increased ventilation can actually increase indoor radon concentrations by increasing the negative pressure differential between the indoor air and the soil gas.

Removal of subsoil

5.68. Elevated indoor radon concentrations are sometimes caused by high ^{226}Ra concentrations in the soil underneath or surrounding the building. In such situations, indoor radon concentrations can be reduced by removing the subsoil and replacing it with uncontaminated soil. This is a major undertaking and it should be carried out only when there are no straightforward options.

Water treatment

5.69. In the few situations in which the water used in the building is a significant source of indoor radon, prior treatment of the water by aeration can be effective. Filtration with activated charcoal can also be used, but it is likely to be less effective. Although aeration of the water can reduce radon concentrations in the buildings to which it is supplied, it can aggravate the problem in the municipal

water treatment plants where aeration is carried out. In any water treatment plant, the air spaces of frequently accessed areas should be well ventilated to prevent the buildup of radon in high concentrations. In treatment plants processing groundwater with high radon concentrations, such measures alone may not be sufficient, and it may be necessary to restrict the periods of occupancy of plant workers in areas of high radon concentrations. This is not usually a problem, because workers usually make only brief periodic inspections in such areas.

Preventive measures in new buildings

5.70. In addition to any remedial action to be taken in existing workplaces, which is the responsibility of the employer, consideration should also be given by the regulatory body or other relevant authority to preventive measures that can be applied to new buildings, including workplaces, in radon prone areas. In the case of dwellings, it has been found that, in areas in which more than 5% of buildings have radon concentrations exceeding 200 Bq/m^3, preventive measures in all new buildings are likely to be cost effective [37]. The difficulty with new buildings is that radon concentrations cannot be predicted with accuracy; they can only be determined after the completion of the construction. The regulatory body or other relevant authority should establish a basis for identifying, in advance, those buildings for which preventive measures should be included in the design and construction and, after construction, should apply checks on the effectiveness of the preventive measures. Appropriate construction codes and guidance on construction practices should be developed. Particularly careful consideration should be given to building development on made up ground if there are indications that the fill material might have elevated concentrations of ^{226}Ra. A thorough quantitative assessment may be necessary and, where necessary, restrictions applied by the regulatory body or other relevant authority.

5.71. The foundations of new buildings constructed in radon prone areas should be designed and constructed so that the ingress of radon from the soil is minimized. Some preventive measures may necessitate major changes to the design and construction of the foundations. Other measures can be very simple and can be incorporated at relatively low cost. These include the provision of a porous fill layer under the floor slab so that radon in the soil gas can be extracted. Space can also be left for an interior exhaust duct for the extracted air. Consideration should also be given to design features that allow the easy introduction of further remedial measures after the construction has been completed, if these are found to be necessary.

5.72. The approach favoured by the regulatory body or other relevant authority will depend on local building styles and on the extent and severity of radon proneness. A combination of approaches may prove to be the best option. In the initial phase of the national action plan, the regulatory body or other relevant authority should closely monitor the outcome of preventive measures and remedial measures to ensure that they are reliable and durable.

EXPOSURE TO COSMIC RADIATION

Sources of exposure

5.73. There are three main sources of cosmic radiation that should be considered for occupational exposure:

(a) Galactic cosmic radiation from sources outside the solar system. Galactic cosmic radiation incident on the upper atmosphere consists of a 98% nucleonic component (mainly protons and helium ions) and 2% electrons. With increasing solar activity, the fluence rate decreases but the maximum of the energy spectrum is shifted to higher energies.
(b) Solar cosmic radiation generated near the surface of the sun by magnetic disturbances. This radiation originates from solar flares and coronal mass ejections when the particles produced are directed towards the Earth. These solar particles comprise mostly protons. Only the most energetic particles contribute to doses at ground level.
(c) Radiation from the Earth's radiation belts (the Van Allen belts). The Van Allen radiation belts are formed by the capture of protons and electrons by the Earth's magnetic field. There are two Van Allen belts: an inner belt centred at about 3000 km and an outer belt centred at about 22 000 km from the Earth's surface. In a region east of Brazil known as the South Atlantic Anomaly, the inner Van Allen belt descends to within a few hundred kilometres to the Earth's surface.

5.74. The intensity of cosmic radiation reaching the upper atmosphere is reduced by the Earth's magnetic field and therefore varies with latitude. The intensity is greatest near the geomagnetic poles and lowest near the equator. The intensity of the total cosmic radiation also varies with time. The variation follows the 11 year solar activity cycle, with radiation intensity at its lowest when solar activity is at its highest.

5.75. High energy particles incident on the atmosphere interact with atoms and molecules in the air and generate a complex set of secondary charged and uncharged particles, including protons, neutrons, pions and relatively light nuclei. Uncharged pions decay into high energy photons, which in turn produce a cascade of high energy electrons and photons. Charged pions decay into muons, which travel large distances in the atmosphere. Thus, at ground level, the muon component of cosmic radiation is the major contributor to dose, contributing about 80% of the dose rate for absorbed dose.

Application of the requirements for protection and safety

5.76. Exposure to cosmic radiation at ground level is regarded as unamenable to control and is therefore excluded from the scope of GSR Part 3 [2].[13]

5.77. Control of occupational exposure to cosmic radiation above ground level is required to be considered for aircrew in terms of the requirements for existing exposure situations, and exposure is required to be controlled for individuals in space based activities (see paras 5.30–5.33 of GSR Part 3 [2]).

Exposure of aircrew

5.78. At altitudes for commercial aircraft, typically 6100–12 200 m, the most significant components of cosmic radiation are neutrons, electrons, positrons, photons and protons, with neutrons contributing 40–80% of the dose rate for effective dose, depending on altitude, latitude and the stage in the solar cycle. The dose rate doubles for every 1830 m of increased altitude. At higher altitudes, the component for heavy nuclei becomes important [23].

5.79. Dose rates in commercial aircraft depend on altitude, latitude and the stage in the solar cycle. For an altitude of 9000–12 000 m at a latitude of 50° (corresponding to a flight between northern Europe and North America), the dose rate is generally in the range of 4–8 μSv/h. Dose rates at lower latitudes are generally lower and, with allowances for climbing and descent, an average dose rate of 4 μSv/h can be used for all long haul flights. For short haul flights, the altitude is generally lower (7500–10 000 m) and the corresponding average dose rate is about 3 μSv/h. Average annual flying times for aircrew are typically 600–900 h [23].

[13] The average annual effective dose to populations from cosmic radiation is estimated to be in the range of 0.3–2 mSv, with a population weighted average of about 0.38 mSv [23].

5.80. In recent years, there have been developments in the monitoring technology for estimating the radiation field on board aircraft (see para. 7.36). In addition, various computer codes have been developed for estimating the doses received by aircrew for specific flight route parameters. Good agreement has been observed between the measured values and the calculated values [38]. Computer codes are now used routinely to assess doses received by aircrew, rather than relying on measurements.

5.81. The average annual doses received by aircrew are typically in the range of 1–3 mSv, with maximum values of 3.5–6.5 mSv being reported from certain States [23].

5.82. Activities in civil aviation vary considerably between States, and in some parts of the world the possibility for aircrew to receive significant doses from cosmic radiation may be very limited. The regulatory body or other relevant authority (which could be a civil aviation authority) is required to determine whether assessment of the exposure of aircrew is warranted (para. 5.30 of GSR Part 3 [2]). If such assessment is not warranted, then no further action need be taken.

5.83. Where assessment of the exposure of aircrew due to cosmic radiation is deemed to be warranted, the following requirements apply (see paras 5.31 and 5.32 of GSR Part 3 [2]):

(a) The regulatory body or other relevant authority is required to establish a framework that includes a reference level of dose and a methodology for the assessment and recording of doses received by aircrew from occupational exposure to cosmic radiation. A reference level of about 5 mSv might be considered reasonable.
(b) Where the doses of aircrew are likely to exceed the reference level, employers of aircrew are required: (i) to assess and keep records of doses; and (ii) to make records of doses available to aircrew.
(c) Employers are required: (i) to inform female aircrew of the risk to the embryo or fetus due to exposure to cosmic radiation and of the need for early notification of pregnancy and (ii) to apply the requirements of para. 3.114 of GSR Part 3 [2] in respect of notification of pregnancy[14], which states that:

[14] Notification of an employer of a pregnancy or a suspected pregnancy or of breast-feeding cannot be made a requirement on a female worker in IAEA safety standards. However, it is necessary that all female workers understand the importance of making such notifications so that their working conditions may be modified accordingly.

> "Notification of the employer by a female worker if she suspects that she is pregnant or if she is breast-feeding shall not be considered a reason to exclude the female worker from work. The employer of a female worker, who has been notified of her suspected pregnancy or that she is breast-feeding shall adapt the working conditions in respect of occupational exposure so as to ensure that the embryo or fetus or the breastfed infant is afforded the same broad level of protection as is required for members of the public."

For female aircrew during pregnancy, the employer should implement the same radiation protection measures as those that would apply in planned exposure situations (see paras 6.2–6.20). This includes a limit of 1 mSv on the annual effective dose.

5.84. According to GSR Part 3 [2], the requirement for dose assessment and record keeping (see para. 5.83(b) above) applies only where the doses of aircrew are likely to exceed the reference level. This implies that the doses of only a small proportion of the workforce would need to be assessed. In practice, however, States with significant activities in civil aviation tend to include all aircrew in the dose assessment process. Given the availability of suitable computer codes for assessing dose directly from the flight parameters (see para. 5.80), this appears to be a practicable option (and appears to be more acceptable to the workforce).

5.85. In terms of current aviation practice, flying altitudes are firmly established and flying times of aircrew are controlled for non-radiological reasons — such controls may provide sufficient control of exposures. Some airlines, again for non-radiological reasons, may already have special working arrangements in place for female aircrew after notification of pregnancy, which will limit the doses received. These limitations ensure that the average annual doses remain at a small percentage, typically about 10%, of the annual dose limit for workers in planned exposure situations. While there are thus no apparent scenarios in which doses could increase above current levels, there are, at the same time, few reasonable opportunities for reducing doses. For instance, any further restriction on the flying times of aircrew could have unacceptable repercussions. Any attempt to reduce the doses received by individual crew members by reassigning them to other flights would do nothing to reduce the collective dose. All of these factors should be taken into account when considering whether there is anything to be gained by imposing further control measures to reduce doses. At present, it would seem that there is little justification for such additional measures.

Exposure of space crew

5.86. At altitudes of 200–600 km and at low inclinations, the main contribution to the exposure of space crew is delivered by protons and electrons trapped geomagnetically by the inner Van Allen belt in the South Atlantic Anomaly where it is closest to the Earth's surface [23]. For low Earth orbit missions of limited duration, the results of some dose assessments show values of personal dose equivalent for the mission varying from 1.9 mSv to about 27 mSv. For a broader range of space activities, mission doses can reach values of the order of 100 mSv.

5.87. Only a few States are involved in space travel. The approach to the control of exposures of space crew has been developed by national and regional space agencies. The requirements of GSR Part 3 [2] for controlling exposures in these exceptional conditions are, by necessity, rather general and essentially reflect current good practice in the States concerned:

(a) The regulatory body or other relevant authority should establish, where appropriate, a framework for radiation protection that applies to individuals in space based activities.
(b) All reasonable efforts should be made to optimize protection by restricting the doses received by space crew while not unduly limiting the extent of the activities that they undertake.

5.88. The framework for protection of space crew should make provision for the setting of appropriate reference levels (e.g. reference levels for mission dose and career dose). The protection framework should also make provision for identifying, at the pre-flight design stage, ways to minimize doses by means such as shielding and by the timing and duration of certain activities. Area monitoring and individual monitoring should be carried out, as appropriate, for dose assessment purposes and for providing warning of changing exposure conditions. Monitoring and dose assessment are essential inputs to the optimization process. Further guidance is provided in Refs [39–41].

6. PROTECTION OF WORKERS IN SPECIAL CASES

6.1. Section 6 provides guidance on occupational radiation protection for two groups of workers for whom there are specific management issues associated with the control of their radiation exposure:

(a) Female workers during and after pregnancy, with implications of exposure not only for themselves but also for the embryo or fetus or the breastfed infant.
(b) Workers who regularly carry out their work on the premises or site of another employer and who may be exposed due to the site operator's use of radiation; and workers who may take onto a site their own source of radiation, with implications for exposure both for themselves and for the employees of the site operator. Such workers, referred to as itinerant workers, are often employed by contractors.

FEMALE WORKERS DURING AND AFTER PREGNANCY

6.2. For the purposes of occupational radiation protection, there is no reason to make any general distinction between workers on the basis of gender. However, additional protection measures are required to be considered for a female worker during and after pregnancy in order to protect the embryo or fetus or the breastfed infant.

Exposure pathways to the embryo or fetus or the breastfed infant

6.3. The following exposure pathways to the embryo or fetus or the breastfed infant are of potential concern:

(a) In utero:
  (i) External exposures due to sources of radiation external to the body of the female worker that irradiate not only maternal tissues but also the embryo or fetus.
  (ii) Internal exposures due to the incorporation of radionuclides by the female worker (or are present in maternal hollow organs, such as the urinary bladder or bowel) that transfer to the fetus through the placenta; or exposure of the fetus to penetrating radiation from radionuclides deposited in maternal tissues (or that are present in maternal hollow organs).
(b) Breastfed infant:
  (i) External exposures due to penetrating radiation from radionuclides in maternal tissues or present in maternal hollow organs such as the urinary bladder or bowel.
  (ii) Internal exposures from the intake of radionuclides by the breastfed infant via transfer from maternal tissues to breast milk and subsequent ingestion during breast-feeding.

Responsibilities of the management

6.4. In terms of para. 3.113 of GSR Part 3 [2], the management is required to provide female workers who are liable to enter controlled areas or supervised areas, or who may undertake emergency duties, with appropriate information on the risk to the embryo or fetus or the breastfed infant, during and after pregnancy. Such a female worker cannot be compelled to notify her employer if she is aware or suspects that she is pregnant or if she is breast-feeding. However, the management is required to inform female workers of the importance of notifying their employer as soon as possible, so that the working conditions in respect of occupational exposure can be modified accordingly to protect the embryo or fetus or the breastfed infant.

6.5. Once a female worker who is liable to enter controlled areas or supervised areas, or who may undertake emergency duties, notifies her employer that she is aware or she suspects that she is pregnant or that she is breast-feeding, the employer is required to adapt the working conditions in respect of occupational exposure so as to ensure that the embryo or fetus or the breastfed infant is afforded the same broad level of protection as is required for members of the public (see para. 3.114 of GSR Part 3 [2]). It is required that such notification not be considered a reason to exclude the female worker from work (see para. 5.83), but it will entail the imposition of more stringent restrictions on the occupational exposures to which the female worker is subject. The employer should inform the female worker of the decision to apply more stringent restrictions.

6.6. The more stringent restrictions should not necessarily prevent the female worker from working with radiation or radioactive material, or from entering or working in designated radiation areas. However, the restrictions should be such as to ensure that under normal operational conditions the requirements of GSR Part 3 [2] with regard to dose limitation for members of the public are met for the embryo or fetus during pregnancy and for the breastfed infant thereafter. Furthermore, the adapted working conditions should be such as to avoid any significant potential exposure due to accidents or other unforeseen events that could result in high radiation doses from external exposure or internal exposure.

6.7. In determining more stringent dose restrictions, account should be taken of any doses that were received by the embryo or fetus as a result of the female worker's occupational exposure to external radiation in the period between conception and notification of pregnancy. Account should also be taken of any doses that were, or that will be, received by the embryo or fetus or by the

breastfed infant as a result of any intakes of radionuclides by the female worker prior to the notification of pregnancy, including intakes prior to conception.

6.8. The employer should consider whether the female worker needs further information and training as a result of any change of working conditions to restrict exposure of the embryo or fetus or of the breastfed infant.

Monitoring

6.9. Because of the more stringent restrictions on dose, female workers should be monitored during and after pregnancy. Doses should be assessed with account taken of all relevant pathways of external exposure and internal exposure.

6.10. Once pregnancy has been notified, the monitoring programme should be redefined so as to be able to determine that the estimated dose to the embryo or fetus or the breastfed infant (including the dose from intakes by the female worker prior to conception) that could be due to occupational exposure will not exceed 1 mSv. Modification of the monitoring programme for internal exposure might be necessary because some radionuclides might be of more relevance for fetal doses than for maternal doses. The biokinetics of some elements might change during pregnancy, although the available information is generally not sufficiently detailed to allow alternative modelling that relates excretion values or organ retention values to intake amounts. Some changes in biokinetics that have been considered by the ICRP [42] could be used for special dose assessments.

6.11. If there are indications that the dose to the embryo or fetus or the breastfed infant might approach 1 mSv/a, individual monitoring of the female worker and individual assessment of the committed dose to the embryo or fetus or the breastfed infant should be performed. Dose reports should be available quickly to allow for prompt action to be taken if it is found that the dose to the embryo or fetus or the breastfed infant might exceed 1 mSv/a.

6.12. A shorter period (i.e. greater frequency) of monitoring may be advisable to keep a closer control over possible inadvertent exposures. However, this frequency should be chosen in consideration of the recording level of the passive dosimeter or other techniques used. For dosimeters with a recording level of 0.1 mSv, a monitoring period of less than one month might not be long enough to evaluate adequately the dose to the fetus during the whole period after the notification of pregnancy. An active dosimeter might serve the purpose of maintaining alertness to any possible accidental exposures. In all cases, the dose recorded for the pregnant female worker should be that of her regular dosimeter.

6.13. The calibration of dosimeters should be considered in assessing doses to the embryo or fetus. For radiation fields of penetrating radiation, dosimeters that have been calibrated for the personal dose equivalent $H_p(10)$ will give an overestimation of the dose. However, this may not be the case for radiation fields of high energy neutrons or of particles in accelerator facilities, for which dosimeters calibrated for doses at different depths below the surface are required.

6.14. Although it is not required to use an additional dosimeter on the abdomen, it can provide reassurance that attention is being given to any exposure during pregnancy. The management should consider the use of an appropriate dosimeter to monitor the dose to the fetus. If the external radiation is homogeneous, there is no preferred position on the abdomen for the dosimeter; but if the radiation field is inhomogeneous, the dosimeter should be positioned on that part of the abdomen that might be irradiated more significantly.

6.15. In the case of a suspected accidental intake, special monitoring should be carried out to ensure that the dose limit for the embryo or fetus or for the breastfed infant will not be exceeded. Monitoring may be carried out by using whole body counting, individual organ counting (such as thyroid counting or lung counting) or in vitro analysis of the female worker's excretions.

Dose assessment

6.16. Information on the dose to the embryo or fetus from maternal intakes of radionuclides has been published by the ICRP [42]. This includes dose coefficients based on biokinetic and dosimetric models that take into account the transfer of radionuclides from the pregnant female through the placenta, and photon exposure due to radionuclides in the placenta and maternal tissues. The dose coefficients, expressed in units of sieverts per becquerel, represent the committed effective dose to the embryo or fetus per unit intake of activity by the female worker. Organ dose coefficients for the fetus are also provided.

6.17. When there is an acute intake by a female worker as a result of an accident or other incident during or before pregnancy, the dose coefficients of the ICRP can be used to calculate the committed organ doses and effective doses to the embryo or fetus. For chronic intakes, the dose coefficients of the ICRP cover three scenarios: (i) chronic intake during pregnancy; (ii) chronic intake one year before pregnancy; and (iii) chronic intake five years before pregnancy.

6.18. In the assessment of external dose to the fetus, only penetrating radiation should be considered. In the case of homogeneous radiation fields, for photons

and beta radiation, the dose recorded by the female worker's dosimeter will be a conservative estimate of the dose to the fetus because, by the time that pregnancy is notified, the dose at the depth of the fetus will generally be lower. In the case of inhomogeneous fields, a careful assessment of the dosimeter results and the corresponding dose to the fetus is necessary.

6.19. Information on the dose to the breastfed infant from the ingestion of radionuclides in the milk, including dose coefficients, has been published by the ICRP [43]. Intakes before and during pregnancy, as well as during lactation, are considered.

6.20. The evaluation of dose to the infant from external exposure due to radionuclides in maternal tissues is based on estimations of the position of the mother and infant, and of the time period during which the mother is holding or is close to the infant. Mathematical models of mother and infant are then used to perform Monte Carlo simulations of the mother's tissues as sources for exposure of the infant.

ITINERANT WORKERS

6.21. For the purposes of this Safety Guide, itinerant workers are occupationally exposed persons who work in supervised areas or controlled areas at various locations and who are not employees of the management of the facility where they are working [44]. Itinerant workers can be self-employed or can be employed by a contractor (or similar legal entity) that provides services at the facilities of other employers. (Such a facility may or may not be a registrant or licensee or be otherwise under regulatory control.)

6.22. The management of a facility and the contractor are both employers. The management of a facility has primary control of the facility, while the contractor provides services under contract. The employees of a contractor, when working in supervised areas or controlled areas at a facility that is not managed by, or under the primary control of, the contractor, will fall within the definition of itinerant workers. In more complex situations, a contractor might itself contract work to a subcontractor, whereupon the employees of both contractor and subcontractor could be itinerant workers. When the contractor is a self-employed person, that person is regarded as being, and as having the duties of, both an employer and a worker.

6.23. Itinerant workers may themselves work with sources of radiation or they may potentially be exposed to radiation sources controlled by the management of the facility at which they are working.

6.24. Itinerant workers may be apprentices who are being trained for employment involving radiation. Exposure of students is not occupational exposure, but students can also be considered 'itinerant students' in this sense when their courses of study or their work experience (overseen by their mentors in the contractor's organization) necessitate their presence in supervised areas or controlled areas established at the facility.

6.25. Examples of itinerant workers and the types of work they perform include:

(a) Maintenance workers in the nuclear power industry — employed by a contractor providing services during normal operations, shutdown or maintenance outages;
(b) Personnel for quality assurance, in-service inspection and non-destructive examination or testing in the nuclear power industry and other industries;
(c) Maintenance staff and cleaning staff in general industry who could be exposed to radiation from a wide range of applications;
(d) Contractors providing specialized services — for example removal of scale and sediment from within pipes and vessels (for the decontamination of equipment), the transport of radioactive waste, or the loading or changing of radioactive sources at irradiation facilities;
(e) Contracted workers in mining and minerals processing facilities who could be exposed to naturally occurring radioactive material;
(f) Industrial radiography companies contracted to work at a facility operated by a management other than their own;
(g) Workers performing contracted security screening using X ray machines or radioactive sources;
(h) Contracted workers involved in the decommissioning of facilities of various types, and in the decontamination of associated buildings and the remediation of outside areas;
(i) Contracted workers of companies installing and servicing medical equipment;
(j) Medical staff who work in supervised areas or controlled areas in several hospitals or clinics (whether fixed or mobile) not operated by their employer.

Issues associated with the use of itinerant workers

6.26. The effective management of itinerant workers is essential for ensuring their protection and safety but can be complicated by issues such as overlapping responsibilities, differences in local work procedures and protection standards, communication difficulties and remote supervision.

6.27. The issues associated with the use of itinerant workers primarily relate to managerial control. Uncertainties over the allocation of responsibilities for arrangements for protection of workers may give rise to difficulties with regard to the control of the exposure of individual itinerant workers over time, for example over a calendar year. As itinerant workers move from facility to facility, workers may accumulate doses that approach or even exceed the annual individual dose limit; this may be true even though none of the prospectively established dose constraints or administrative dose targets at the different facilities had been exceeded.

6.28. The range of work carried out by itinerant workers makes it difficult to assign responsibilities explicitly without first considering specific situations. These can range from situations in which the management of a facility will be required contractually to provide most of the necessary services for protection and safety for itinerant workers to situations in which most of the duties and responsibilities will usually fall on the contractor. Within this range, three types of exposure scenario can occur:

(a) The operation of a facility has the potential to cause exposure of the contractor's employees, who themselves do not possess a radiation source. In such cases, the management of the facility is the registrant or licensee and the contractor is merely an employer.
(b) The contractor's employees bring their own source of radiation to a facility and, hence, have the potential to cause exposure of the employees of that facility. In such cases, the contractor is the registrant or licensee and the management of the facility is merely an employer.
(c) A combination of (a) and (b), whereby the operation of the facility and the activities of the contractor on-site both have the potential to cause exposure to each other's employees. In such cases, both the management of the facility and the contractor are registrants or licensees.

Cooperation between employers

6.29. The main responsibility for protection and safety for workers at a facility lies with the management of that facility. At the same time, a contractor providing services to the management of the facility is responsible for protection and safety for its own employees. It follows that there will be overlapping responsibilities for the management of itinerant workers, and cooperation between the two employers (the management of the facility and the contractor) is required (Requirement 23 of GSR Part 3 [2]). The specific content of these joint responsibilities will depend on the type of work carried out, but it will require consultation and cooperation to the extent necessary for compliance with the requirements for protection and safety for all workers at the facility. This requirement is reflected in para. 3.85 of GSR Part 3 [2], which states that:

> "If workers are engaged in work that involves or that could involve a source that is not under the control of their employer, the registrant or licensee responsible for the source and the employer shall cooperate to the extent necessary for compliance by both parties with the requirements of these Standards."

6.30. Further requirements relating to cooperation between employers are given in paras 3.86 and 3.87 of GSR Part 3 [2], which state that:

> "3.86. Cooperation between the employer and the registrant or licensee shall include, where appropriate:
>
> (a) The development and use of specific restrictions on exposure and other means of ensuring that the measures for protection and safety for workers who are engaged in work that involves or could involve a source that is not under the control of their employer are at least as good as those for employees of the registrant or licensee;
> (b) Specific assessments of the doses received by workers as specified in (a) above;
> (c) A clear allocation and documentation of the responsibilities of the employer and those of the registrant or licensee for protection and safety.
>
> "3.87. As part of the cooperation between parties, the registrant or licensee responsible for the source or for the exposure as appropriate:

(a) Shall obtain from the employers, including self-employed persons, the previous occupational exposure history of workers...and any other necessary information;
(b) Shall provide appropriate information to the employer, including any available information relevant for compliance with the requirements of these Standards that the employer requests;
(c) Shall provide both the worker and the employer with the relevant exposure records."

6.31. Where the need for cooperation leads to agreement on procedures to be followed, this should ideally be documented. It is likely to be appropriate for such an agreement to form part of the formal contractual arrangement, particularly in large or complex contracting (and subcontracting) situations (e.g. when the management of a facility specifically delineates a part of its site to be handed over to a main contractor to carry out some work such as decommissioning). It should be ensured by means of this documentation that each party knows which of the legal demands on the employer it is specifically responsible for meeting. The detailed arrangements and identification of responsibilities will vary with the nature of the work and the relevant experience of the parties involved.

6.32. Information sheets and checklists are useful aids to the exchange of information between employers and for assessing the adequacy of arrangements for protection and safety. Information sheets and checklists should be used for summarizing the requirements for protection and safety to be fulfilled and for listing the various points that should be discussed and agreed between the management of the facility and the contractor before the start of the contract work.

Sources under the control of a facility

6.33. In many types of work, a contractor's employees whose work does not directly involve the use of radiation sources have to enter an area of a facility where they may be exposed to radiation arising from the normal operation of the facility. Examples of such itinerant workers include maintenance staff and cleaning staff. In many cases, the contractor and its employees will have little or no experience of working in radiation areas, and will have a limited knowledge of the regulatory requirements for protection and safety.

6.34. In such circumstances, it is the responsibility of the management of the facility to apply the same level of protection and safety to the itinerant workers as to its own employees. Having the necessary arrangements in place for achieving this should be a precondition for the engagement of the itinerant workers.

Consideration should be given to formalizing this by referring, in the contractual agreement, to the relevant measures for protection and safety, which are those specified in Section 3 and would include, as appropriate:

(a) Optimization of protection and safety (including any associated dose constraints);
(b) Dose limitation;
(c) Establishment of classified areas;
(d) Use of personal protective equipment;
(e) Local rules and procedures;
(f) Monitoring and dose assessment;
(g) Dose records;
(h) Information and training;
(i) Workers' health surveillance.

6.35. If the contractor's work includes non-standard operations, a prior radiological evaluation of those operations is necessary. The evaluation should consider the various protection options and the amount of detail in the evaluation should be commensurate with the radiation risks. Responsibility for preparation of the assessment should be assigned to the management of the facility because of its detailed knowledge of the work, but the contractor should be involved, possibly with the assistance of a qualified expert. It should thereby be ensured that all relevant issues for protection and safety are considered at an early stage.

6.36. The management of the facility will have arrangements in place for the assessment of doses for its own employees, and appropriate arrangements should also be made for the assessment of doses for the contractor's employees. This may necessitate the management of the facility providing the contractor with dosimeters and then assessing them at the completion of the work, or it may necessitate the contractor arranging for its own individual dosimetry. The arrangement to be followed should be specified in the contractual agreement. If the work is being carried out under dosimetry arrangements made by the management of the facility, the relevant dose records should be made available to the itinerant workers and their employer (the contractor). In all cases, each itinerant worker should comply with any requirements of local rules or procedures to wear an individual dosimeter in a particular area.

6.37. On completion of the work, and possibly at stages during the contract, the doses received by the contractor's employees should be compared with those predicted in the prior radiological evaluation.

6.38. In deciding which of its employees are suited to work under a particular contract, a contractor will require the following information from the management of the facility:

(a) Details of any radiological hazards and associated controls, and an estimate of the maximum radiation doses likely to be received by the contractor's employees in the work under the contract;
(b) Details of any additional training that will be needed and that, therefore, should be provided either by the contractor or by the management of the facility;
(c) Information on whether the contractor's employees need to wear individual dosimeters and, if so, what arrangements are in place;
(d) Details of non-radiological hazards such as chemicals, dust and heat;
(e) Provision of personal protective equipment, if required.

6.39. Before a contractor's employee is accepted into a facility to work in a controlled or supervised area, the management of the facility should obtain from the contractor specific information concerning the employee. If this information is immediately available, it will permit rapid entry to the facility. This information should include:

(a) Details of appropriate qualifications of the employee (training, experience and certification);
(b) Details of the employee's dose history;
(c) Any relevant information on the employee's fitness for work.

6.40. It will also be appropriate for the management of the facility to carry out an assessment of the competence of the contractor's employees. This is discussed further in paras 6.56–6.65.

6.41. The contractor should consider whether it needs to consult one or more qualified experts for the work it is to undertake, depending on the nature of the work and any contractual conditions. If the contractor wishes to consult a qualified expert, it may seek guidance from the management of the facility or from an independent source for suggestions of suitable experts. The following subjects are examples of those for which guidance could be required from a qualified expert:

(a) The review of engineered controls relating to protection and safety;
(b) The formulation of suitable local rules and procedures;
(c) Appropriate dosimetry arrangements;

(d) The requirement for personal protective equipment;
(e) The use of radiation monitoring equipment;
(f) Record keeping;
(g) Emergency procedures.

6.42. The management of the facility should discuss with the contractor the arrangements for radiological supervision at an early stage and may arrange for an existing radiation protection officer (see para. 3.66) to act as the radiation protection officer for the contractor and its employees. Alternatively, the contractor may be required to appoint one of its own employees as a radiation protection officer, and should then ensure that this person is adequately trained. The radiation protection officer appointed should be acceptable to the management of the facility and the contractual agreement should require this radiation protection officer to work closely with (and to take guidance from) a nominated member of the supervisory staff of the facility. The radiation protection officers of the facility and of the contractor should maintain the necessary level of communication.

Sources under the control of a contractor

6.43. A source under the control of a contractor may have to be taken by an employee of the contractor into a facility. Even though radiation sources (e.g. nuclear gauges) might be used within the facility as part of its normal operation, it is often the case that the area in which the contractor works is outside any classified areas associated with such sources. In such a situation, there is no potential for the itinerant worker to be exposed due to sources under the control of the facility. However, the source brought in by the itinerant worker could cause exposure of employees of the facility.

6.44. Such a situation arises most commonly when industrial radiography is carried out by a contractor on-site. Consequently, the guidance given in paras 6.45–6.50 refers specifically to such work. Similar principles and actions will apply to other work activities such as source loading operations in irradiation facilities. If unsealed sources are involved, precautions should be taken to avoid surface contamination and airborne contamination (see paras 9.24–9.46).

6.45. Industrial radiography involves the inspection of components (e.g. pipes, welds and pressure vessels) to determine whether cracks or other defects are present. The radiation source used is a sealed radioactive source or an X ray machine. Both types of source necessitate strictly controlled procedures to protect the radiographers using them and other persons on the site. An essential

part of these procedures is the maintenance of a barrier at a suitable distance from the source, intended to prevent unauthorized entry to the controlled area within the barrier (i.e. for cordoning of the controlled area). This type of work may be carried out at night or at a height, for which additional protective measures such as stronger lighting and stricter supervision should be considered. Cooperation between the management of the facility and the contractor is essential for ensuring adequate protection and safety for the employees of the facility.

6.46. Where the management of the facility has no direct in-house expertise in the work to be carried out by the contractor, it should restrict its involvement essentially to non-technical information gathering. The management of the facility should place the primary resposibility on the contractor for cooperation on the more technical aspects of the work, but should nevertheless be able to satisfy itself that the contractor has made adequate provision for achieving safe working conditions. In doing this, the management of the facility may need the assistance of a qualified expert.

6.47. Prior to the commencement of work, the management of the facility should obtain from the contractor:

(a) A telephone number that the contractor can be contacted in the event of an emergency.
(b) The name(s) of the radiation protection officer(s) who will be present during the work.
(c) The type of radiation generator or radiation source to be used.
(d) A copy of the contractor's local rules and procedures, which should provide sufficient information about the proposed work; if adequate local rules are not available, the contractor should not be allowed to undertake the work.

6.48. The management of the facility should ensure that the contractor implements the following measures for protection and safety:

(a) Placement of barriers to prevent access to controlled areas in which dose rates exceed predetermined levels;
(b) Posting of sufficient warning notices;
(c) Provision of warning signals (that do not have any other local meaning or significance) prior to and during the exposure;
(d) Display of explanatory notices at access points;
(e) Inspections of equipment and radiation monitors prior to and after use;
(f) Replacement or repair of equipment identified as inoperable prior to use;
(g) Searching of the controlled area before starting and periodically thereafter;

(h) Patrolling of the barrier to prevent unauthorized access;
(i) Use of a suitable, calibrated radiation monitor in setting or verifying placement of the barrier and confirming expected dose rates after exposures (this is especially important where pulsed X ray fields might be present);
(j) Provision of adequate storage facilities;
(k) Formulation of emergency plans;
(l) Use of appropriate personal monitoring devices.

6.49. The management of the facility should ensure that any of its employees who may be affected by the contractor's work are given sufficient information about the proposed work. This should include people whose duties may place them in the vicinity of the work, security staff, the management and people who would become involved in an emergency.

6.50. While work is in progress, the management of the facility should arrange occasional, unannounced safety audits to ensure that the contractor's employees are observing the agreed, safe working practices. Such audits could be undertaken by employees of the facility or by an independent third party. When carrying out an audit to assess the standard of protection and safety, the management of the facility may find it useful to refer to a checklist of the items to be audited (see para. 6.32).

6.51. A source under the control of a contractor might have to be taken by an employee of the contractor into an area of a facility where, during normal operation of the facility, there is also the potential for exposure due to a source under the control of the facility. While the guidance given in paras 6.33–6.50 remains relevant, the additional guidance given in paras 6.52–6.55 should also be followed.

6.52. Industrial radiography or other work may involve using a source under the control of the contractor in areas where there is a significant ambient dose rate arising from the operation of the facility. Before undertaking such work, the choice of an appropriate dose rate for which to erect barriers and signs should be discussed and agreed between the contractor and the management of the facility. Consideration should also be given to the timing of the proposed work.

6.53. Work should be carried out not only in accordance with the contractor's local rules and procedures but also in accordance with the local rules and procedures for those sources associated with the facility. The contractor may therefore need to modify its local rules and procedures so as to incorporate certain aspects of

the local rules and procedures of the facility, and so as to ensure that there are no conflicting requirements. This should be clearly included in the contract.

6.54. Special training of the contractor's employees may be required because of the potential for exposure due to sources under the control of the facility, even though such employees might already be trained in the use of their own radiation sources. In such circumstances, many facilities require contract radiographers and their radiation protection officers to be trained to a specified level.

6.55. Consideration should be given to the possible impacts of the contractor's radiation source on any radiation related instrumentation installed at the facility (e.g. the impact on area gamma monitors and detection systems for criticality incidents, and the risk of unnecessary false alarms). In the event of such incidents being identified, appropriate corrective actions should be taken. These could include the use of smaller sources or collimated radiation beams to minimize dose rates, or the deactivation of some instrumentation for a limited period.

Competence of itinerant workers

6.56. The management of facilities should ensure that contractors carrying out work at the facility are using personnel who are competent to carry out the work. Accordingly, the competence of contractor personnel may need to be formally assessed and documented. This approach will be appropriate where the contractors' employees are potentially exposed due to the sources under the control of the facility. The approach will also be appropriate where the contractors are themselves bringing a source into the facility and where there is a potential for the facility's employees to be exposed due to this source.

6.57. The assessment process should include formal procedures to determine the necessary competences (through education, experience, and initial and continued training programmes) and the qualification requirements for any job carried out by contractors that could have implications for protection and safety. Established guides or quality management procedures can be useful in the assessment process.

6.58. The level and detail of the assessment process will depend on the type of facility and the work being carried out. Some itinerant workers will work in professions that have qualification or certification schemes to demonstrate competence. Examples of such professions include radiological medical practitioners, medical physicists, medical radiation technologists and industrial radiographers. Management of facilities intending to employ itinerant workers in such professions should be aware of the certification and qualification

requirements for this work, and should incorporate these requirements into the assessment process. It may also be appropriate to specify these qualifications in the contractual arrangements. Other professions and skills might not have qualification requirements. In these circumstances, the assessment of competence could be restricted to a review of curricula vitae, certificates, training records, references and reports of similar work carried out at other facilities.

6.59. Under certain circumstances, the management of the facility may wish to specify site specific competence requirements to be fulfilled before the contractor is permitted to work on-site. These requirements could include the competence to use appropriate respiratory protective equipment. In these circumstances, the management of the facility should, as necessary, provide appropriate training to cover these competences, or should alternatively be able to recommend where such training could be obtained. The satisfactory completion of such training would be an input into the competence assessment process.

6.60. Contractors should ensure that their employees are suitably qualified for the work to be carried out and should submit details of each employee's qualifications to the management of the facility prior to commencing work at the facility. Itinerant workers should not be allowed to work without the required training and certification in the work and in radiation protection. The equipment and machines to be operated by them might, for instance, have very high intensity gamma sources with the potential for causing high level exposures in a short interval of time if not operated properly.

6.61. The assessment of the competence of contractor personnel will conclude either that the contractor's employees are competent to carry out the job or that there are deficiencies in qualifications or experience. If there are deficiencies, compensatory actions should be taken before the contractor's employees are allowed to work on the site. The main characteristics of each particular situation should be taken into consideration in order to define the most appropriate compensatory action.

6.62. For training related compensatory actions, consideration should be given to delivering any required training before the contractor's employees commence work on-site, and to initiating liaison between the site operator and the contractor to fill any gaps identified. The site operator may be able to provide any site specific training required.

6.63. The following additional management initiatives could also be implemented as compensatory measures:

(a) Provision of direct supervision by the site operator;
(b) Replacement of certain contractor personnel;
(c) Documentation of additional experience, training or education.

6.64. The contractor should periodically review the competence of its employees, with particular regard to the following:

(a) Any changes in the professional qualifications required;
(b) Any changes in legislation;
(c) Lessons to be learned from experience at the facility and other facilities;
(d) The worker's dose record;
(e) The ongoing adequacy and effectiveness of the level of training acquired;
(f) The need for refresher training;
(g) Any change in fitness for work;
(h) Any changes in facilities, operations or work practices.

6.65. The performance of the individual worker should also be assessed. Lessons learned from problems encountered, and actions taken to resolve difficulties, may lead to the identification of further competence training for one or more workers.

Radiation protection programme

6.66. The complexities associated with the management responsibilities and radiation protection arrangements for itinerant workers highlight the need for the work to be conducted in accordance with an effective radiation protection programme (see paras 3.49–3.158). The radiation protection programme should, among other things, assign responsibilities for protection and safety for itinerant workers to the management of the facility and to the contractor in accordance with the terms of the contractual agreement.

6.67. For most situations, the prior radiological evaluation on which the radiation protection programme will be based should be a collaborative effort by the management of the facility and the contractor, with the more qualified of the two employers taking the leading role. Use should be made of the results of previous assessments. For a facility that uses radiation sources as part of its normal operation, the management should carry out a prior radiological evaluation for its own operations, followed up by a more detailed safety assessment. Similarly, where the contractor has its own sources of radiation, it should carry out a prior radiological evaluation and a safety assessment that are appropriate for most of the facilities at which those sources are likely to be used.

6.68. The management of the facility and the contractor share joint responsibility for developing the radiation protection programme but, as with the prior radiological evaluation and the safety assessment, the levels of knowledge and expertise of those two parties may be expected to contribute to the mutually agreed allocation of responsibilities to ensure the development of an effective radiation protection programme. In many cases, the existing radiation protection programme of the facility or of the contractor may need limited modifications to reflect the proposed work by the contractor at the facility.

6.69. The use of an existing radiation protection programme as the basis for ensuring radiation protection for itinerant workers is illustrated by the following two examples:

(a) At a nuclear power plant, the management will have acquired extensive knowledge of the radiation risks associated with the operation and maintenance of the facility, will have already carried out a detailed safety assessment for its own employees (and likely for those employees of contractors that are foreseen to be used for assessed tasks) and will have established a comprehensive radiation protection programme. In this instance, therefore, it would be appropriate for the management of the facility: to communicate the relevant information on safety assessment to the contractor; to discuss work related circumstances and any identified concerns with the contractor; and to draw up a simplified radiation protection programme that covers the work of the contractor.
(b) An industrial radiography company working at a chemical plant will already have developed its own radiation protection programme for work on-site, but it will need to liaise with the safety officers at the facility and to provide them with appropriate information from the radiation protection programme. That information will include the arrangements for management and supervision, and the procedures to be used to ensure radiation protection of the employees at the facility.

Records of occupational exposure

6.70. Some itinerant workers might work at a facility for much less than a year before moving on to the next facility. In this way, they might accrue doses at multiple facilities within a period of one year. At each facility, the accrued dose may or may not be substantive; however, the accrued dose for several facilities in one year could result in a total accrued dose that approaches the applicable dose limits. These workers' doses should therefore be tracked over long time periods,

and the responsibilities and arrangements for achieving this should be clearly established and documented.

6.71. The arrangements should be such as to ensure that, for each itinerant worker, an up to date record of the doses received and the status of health surveillance is available. This could be in the form of an output from a centralized database of workers' exposure records or an individual document on radiological monitoring (sometimes referred to as an 'individual radiation passbook') or alternative individual dose record. Before starting contract work at a facility where radiation sources are used, the worker's occupational exposure and health surveillance records should be made available to the management of the facility so that an appropriate plan for protection and safety can be established.

6.72. The worker's record of occupational exposure should be kept up to date while they are working on-site, either by the management of the facility or by the contractor, depending on who has the relevant responsibility. To avoid delays in updating the record, estimated doses (based, for instance, on the results of workplace monitoring) could be recorded pending receipt of the results of the worker's personal monitoring data. This provides a useful indication of the worker's dose for the next facility manager if the worker has moved on to another facility in the meantime. It is the responsibility of the employer of the itinerant worker to ensure that the worker's record of occupational exposure is kept up to date.

Training

6.73. In a facility in which radiation sources are used as part of normal operation, itinerant workers carrying out contract work in an area with no implications for protection and safety (e.g. cleaning, painting, general maintenance or construction in a supervised area) will require only a minimal knowledge of radiation protection and will need to be provided with only very basic information on any relevant precautions to be followed while in the supervised area. Conversely, itinerant workers who are required to carry out operations in controlled areas that necessitate complex tasks may need to be provided with training on topics such as access procedures, precautions to be taken, the use of personal protective equipment and procedural requirements. Itinerant workers bringing their own sources into a facility should be adequately trained in the safe use of these sources. It is the responsibility of the employer of the itinerant workers to ensure that training is provided. The management of the facility might need to be consulted on the level and content of the training required for the performance of contracted tasks in the workplace at the facility.

6.74. In some situations — typically where the contractor has only limited experience of working with radiation — the management of the facility may provide the contractor and its employees with necessary information on protection and safety, including information relating to on-site emergencies. Depending on the circumstances, such information could take the form of notices, written instructions or the content of formal training. In other situations, the contractor may take responsibility for training, but the management of the facility should nevertheless provide, before the work commences, information about the risks relevant to the work and about any special training necessary. At a large establishment, the management of the facility could help to provide suitable training (in so far as it is relevant to the facility) either on behalf of the contractor or as a separate contractual arrangement. This training should be at a level similar to the training that the management of the facility provides for its own employees.

6.75. Where the contractor takes responsibility for the training, it should assess the training needs of its employees. In consultation with the management of the facility and a qualified expert, as necessary, it should draw up a training programme that provides the appropriate level of training and information for any forthcoming work at the facility. In doing this, consideration should be given to the following:

(a) The nature of the work to be carried out in the foreseeable future;
(b) The potential for exposure associated with this work;
(c) The extent of training already provided and qualifications obtained;
(d) Site specific requirements at the facilities to be visited (e.g. entry procedures, the use of personal protective equipment, and emergency procedures).

6.76. Several levels of training might need to be provided, depending on the nature of the work to be carried out. For example, only basic awareness training in radiation protection might be required for the majority of the workers, but more comprehensive training will be necessary for those staff who will act as radiation protection officers.

Review of protection and safety

6.77. The arrangements and procedures established by the management of a facility for radiation protection and safety for itinerant workers should be reviewed periodically to ensure that they remain appropriate and relevant to the work. If the same itinerant workers are on-site for a protracted period of time, their working practices should be reviewed and audited at appropriate

intervals to assess the level of compliance with the arrangements and to identify any weaknesses in the procedures. Likewise, when new itinerant workers are about to commence work, the arrangements and procedures should be discussed with the contractor and the opportunity should be taken to review their continued validity.

6.78. In carrying out the review, account should be taken of the following:

(a) Changes in the working environment;
(b) Legislative and regulatory changes;
(c) Any modifications to working practices;
(d) The level of adherence to current arrangements;
(e) The practicability of current arrangements;
(f) The adequacy of emergency plans;
(g) The effectiveness of previously used arrangements and current arrangements in maintaining doses as low as reasonably achievable;
(h) The need for changes to the radiological evaluation, the safety assessment for the planned work and the level of interaction with the regulatory body;
(i) Lessons to be learned and operational experience.

6.79. Item (g) in para. 6.78 is especially important. The effective optimization of doses received by itinerant workers is a principal objective of the arrangements and procedures. In assessing the adequacy of the arrangements, the management of the facility should therefore review the records of occupational exposure for itinerant workers while they were on-site, and should satisfy itself that they are appropriate to the type of work being undertaken. This review should be carried out in consultation with the other employers involved and, possibly, with advice from a suitable qualified expert.

6.80. The outcome of the review will probably be a series of actions to be taken to rectify, improve or enhance the arrangements and procedures. These actions should be implemented as soon as reasonably practicable and, preferably, before itinerant workers next perform the tasks at the facility that are to be assessed. The findings of the assessment of the adequacy of the arrangements should be communicated to the workers affected and to their employers for their input and for incorporation into any revised contractual agreements and local rules and procedures.

6.81. Contractors that have sources under their control should also review their internal arrangements and procedures at regular intervals. As a registrant or licensee, the contractor is responsible for restricting the doses received by its employees and for optimizing radiation protection. The contractor

should have procedures in place for the ongoing review of results from dosimetry. As discussed, the arrangements and procedures for long term work at a single facility should be reviewed periodically in consultation with the management of the facility. The contractor should also review any ongoing arrangements and procedures that are followed for all site work, for example programmes for workers' health surveillance, procedures for maintenance of equipment, and arrangements for keeping records of the location, description, activity and form of each source for which it is responsible.

Issues associated with specific types of facility

Nuclear installations

6.82. Rigorous requirements have to be met before itinerant workers are granted access to a nuclear installation, owing to the potential for such workers in certain areas to receive high doses. These requirements may include adherence to some or all of the following procedures:

(a) The contractor should enter the following information on an access approval form: (i) individual information regarding the worker; (ii) the contract reference; (iii) details of the employer; (iv) the professional skills of the worker together with relevant certificates; and (v) the expected duration of the operation. The contractor should then send the form to the management of the facility who should then add a description of the areas where access is permitted and the period of validity of the access permit to the facility and to the supervised areas and the controlled areas therein. For access to areas with high (or potentially high) dose rates, a specific type of permit should be used. The access procedure for a nuclear power plant can take several days to process.
(b) On arrival of the itinerant worker at the nuclear power plant, a check should be made of all the information mentioned in para. 6.82(a), as well as: (i) the worker's fitness for work; and (ii) the worker's dose record over the current calendar year, over the past twelve months and the past five years.
(c) Specific training should be provided on particular facility conditions and for actions required in the event of an emergency.
(d) A check should be made of the compatibility of the skills of the itinerant worker with the work to be performed.
(e) Itinerant workers should be able to justify their access to a controlled area by producing a radiation work permit developed in accordance with the facility's work management system (see para. 3.96).
(f) An individual dose objective for the itinerant worker should be established.

6.83. Special procedures should be adopted for itinerant workers on short term contracts, such as the following:

(a) An individual dose objective should be calculated on a pro rata temporary basis.
(b) Access to areas of high (or potentially high) levels of radiation should be restricted or prohibited.

6.84. In tasks involving high or potentially high dose rates, the following special training and procedures should be provided for itinerant workers:

(a) A pre-work review, involving a detailed description of the work to be done, technical data, and dosimetric and environmental conditions.
(b) A preliminary procedure to carry out the work, together with an associated dose estimate.
(c) Training on a mock-up or, where reasonably feasible, a representative simulation of the actual job site or, if necessary, a briefing using descriptors of the job site (e.g. photographs or videos).
(d) Feedback on this training, including the exposure time, difficulties in carrying out tasks, phases to be improved, specific tools to be developed and the number of people in the workplace simultaneously.
(e) Anticipation, to the extent possible, of potential breakdowns of tools or equipment and of other operational incidents. This facilitates the formulation of corrective actions and the training of workers to carry out such actions in a manner that keeps doses as low as reasonably achievable.
(f) Improvement of working procedures and estimated doses, and optimization of protection and safety.
(g) Final training in accordance with such optimized procedures.

Facilities for performing medical exposures

6.85. The use of radioactive sources, accelerators and radiation generators for therapeutic purposes and the use of X ray generating equipment for diagnostic and interventional purposes are universal practices with the potential for high doses to workers. Equipment engineers and maintenance workers often fall into the category of itinerant workers. In addition, it is common practice for radiological medical practitioners, medical physicists, medical radiation technologists and other health professionals with specific duties in relation to medical uses of radiation to work in several hospitals and clinics. While they will be employed primarily by one hospital or hospital group, they will be acting as contractors in others. These workers should receive training in radiation protection in advance

in their initial pre-qualification training and will be working in accordance with very similar procedures at each hospital.

6.86. The critical issue in terms of itinerant workers in facilities for medical uses of radiation is the adequacy of the dosimetry arrangements. The workers should be provided with dosimeters by their primary employer and are likely to wear these dosimeters at every location. However, this practice can create difficulties when a high radiation dose is recorded on a dosimeter. In this situation, it might not be possible to determine from where the high dose was received and, thus, which employer is responsible for undertaking any investigation or corrective actions.

6.87. Suitable dosimetry arrangements will entail the worker wearing a separate dosimeter for each employment location, possibly with the dosimeter from the principal employer being worn at all locations for primary record keeping purposes. These dosimetry arrangements should be made after consultation with all parties involved.

6.88. In addition to dosimetry arrangements, itinerant workers should receive specific training to become familiarized with equipment, such as accelerators and X ray generating equipment, in all the facilities in which they will be working. This specific training on equipment should include operational details and safety aspects.

6.89. The adequacy of measures for protection and safety where radiation generating machines or unsealed sources are used should be ensured. Radiation monitoring equipment suitable for the characteristics of the radiation fields should be available. Whole body or partial body shielding between the source and medical personnel is often used as a means of reducing dose. Personal protective equipment suitable for the situation (such as protective aprons and gloves, face or eye shields and thyroid collars) should also be made available where appropriate.

6.90. When unsealed sources are used by the staff of the facility or of the contractor, rules and procedures for the control of surface contamination and airborne contamination, and the potential need for individual measurement programmes or supplementary workplace monitoring to assess whether measurable intakes of radionuclides occurred, are of relevance and should be considered. The prior radiological evaluation and discussions among all parties involved will be helpful in decision making.

6.91. Certain precautions should also be taken to avoid unintentional and accidental medical exposures of workers (and patients) that could occur as a result of maintenance work performed by itinerant workers:

(a) Sometimes, when itinerant workers perform maintenance services, changes are made to the default settings of the system (e.g. fluoroscopy modes in which the X ray beam is pulsed). Such changes should be registered so that the users of the systems are aware of them, and the responsible medical physicist at the facility should be informed personally.
(b) To avoid the possibility of an accident resulting from the temporary deactivation of a safety interlock during maintenance of a system by an itinerant worker, backup measures should be in place to prevent the clinical use of the system in such circumstances.
(c) After any work performed by an itinerant worker that affects radiation aspects or image quality aspects of the system, a detailed report should be prepared and submitted to the head of the service where the work was performed.
(d) After any maintenance is performed on a system by an itinerant worker, the system should be left in a state ready to be used with patients. Sometimes, after repair of a film processor, the cassettes are loaded with exposed films, and this can lead to some patients being irradiated twice when the system is next used because the first images were not usable.

Mines involving exposure due to radon and/or due to naturally occurring radioactive material

6.92. Radon concentrations in underground mines depend on the ventilation conditions and can therefore reach high levels in some locations. The mining of uranium ore (and sometimes of certain other minerals) can involve external exposure and internal exposure of workers due to naturally occurring radioactive material. The hiring of contractors, both short term and long term, is commonplace in mines. The question of who is best placed to take responsibility for radiation protection measures (including training, health surveillance and the use of personal protective equipment) with respect to itinerant workers depends very much on the nature of the contract work, which can vary widely, as illustrated by the following two examples:

(a) In some mines, contractors are hired to carry out normal day to day mining operations that may be conducted on a large scale and may continue for a long time. In such situations, it may be best to place responsibility for the management and control of radiation exposure of itinerant workers

with the management of the mine. The management should already have the necessary competence and infrastructure in place, and this competence and infrastructure will almost certainly be greater than that possessed by the contractor.

(b) Contractors may be hired to carry out specialized, non-routine tasks that do not form part of the day to day operation of the mine, such as the installation and maintenance of plant and equipment in the mine, excavation of ore passes[15] and sinking of shafts. Such tasks may sometimes involve higher exposure levels than those encountered during normal operations. It is possible, in such circumstances, that the contractor may be better placed to take responsibility for the radiation protection of its employees because of the specialized nature of the work and because the contractor performs this work on a routine basis and is likely to be more familiar with the particular radiation hazards involved. The contractor also has the advantage of being more easily able to keep track of its employees' radiation doses over long periods. On the other hand, the contractor's experience in carrying out such specialized work may have been gained mostly in situations where the radiological hazards were insignificant, in which case the responsibility for radiation protection may be better placed with the management of the mine, even though the work is of a specialized nature. The mine management should then familiarize itself with the radiation hazards associated with such specialized work.

6.93. The full range of options with respect to the assignment of operational responsibilities should be kept open and, as a general rule, the responsibility should lie with the employer having the greatest levels of competence and infrastructure for radiation protection for the tasks in question. Because many workplaces in mines are remote and relatively inaccessible, supervision of work activities can be difficult. There should be close and sustained interaction between the management of the mine and the contractor.

Facilities for the extraction and processing of minerals

6.94. Facilities for the extraction and processing of minerals normally involve the use of radioactive materials and/or radiation generators. Sealed sources, often with very high activities, are used extensively in measurement and control devices. Widespread use is made of industrial X ray generating equipment for testing the integrity of piping and pressure vessels. Unsealed radioactive materials

[15] An ore pass is a vertical or inclined chute created in underground mining operations for the downward transfer of ore.

are often used as tracers, such as in oil and gas pipelines [26]. In addition, the presence of minerals and mineral processing residues can result in exposure due to naturally occurring radioactive material [24–29].

6.95. Extensive use is made of contractors in such facilities, not all of whom have the necessary specialist knowledge in protection and safety to be able to take responsibility for the control of exposure of their workers. It is common practice in the chemical industry and in the oil and gas industry to use contractors for specialized jobs, such as the removal of scale and sediment from the interior of vessels, or the demolition and removal of a redundant plant, and these operations may involve working on parts of the plant that are contaminated with naturally occurring radioactive material. Itinerant workers in these situations often work at a particular facility for much less than a year, but they could be exposed to radiation that, if sustained, would give rise to annual doses approaching or exceeding the relevant dose limits. The occupational exposure of such workers should be carefully managed.

6.96. The nature of many specialist tasks involving exposure of itinerant workers to naturally occurring radioactive material with relatively high activity concentrations (e.g. the removal of radium rich pipe scale) is such that there may be significant scope for optimization of doses. It may be possible to achieve substantial reductions in doses with relatively simple modifications to the work (see, for instance, Ref. [26]). The management of the facility and the management of the contractor should both be alert to the possibility that fulfilment of the requirements for optimization may be overlooked more easily in specialized tasks involving itinerant workers than in routine normal operation of the facility.

6.97. In many cases, the contractor's knowledge of protection and safety is limited. The contractor's employees should be made aware of the implications of the work for radiation protection and the procedures to restrict exposure. The management of the facility and the contractor should discuss the radiation protection aspects of the work at the planning stage. The topics covered should include the following:

(a) The hazards posed by sealed sources (e.g. nuclear gauges) and by naturally occurring radioactive material (e.g. radium rich scale) in various parts of the plant;
(b) The presence of controlled areas or supervised areas;
(c) Procedures to be followed to optimize protection so that exposure is as low as reasonably achievable;
(d) The use of appropriate personal protective equipment;

(e) Supervision;
(f) Dose assessment and maintenance of dose records;
(g) Waste management;
(h) Training;
(i) Actions to take if ventilation, dust control or other relevant control systems fail or are taken out of service.

6.98. The site operator should, if necessary, assist the contractor in performing a prior radiological evaluation and in developing local rules and procedures. In view of the nature of the work and the precautions to be taken, the contractor's employees should receive training in the hazards of radiation exposure, pathways of exposure, the procedures to be followed for restricting exposure and the duties of the radiation protection officer. The site operator should, if necessary, arrange this training on behalf of the contractor.

6.99. The management of the facility should also, if necessary, discuss with the contractor any non-radiological risks at the facility or specifically at the work site where the itinerant workers will be. The management should ensure that mutually agreed techniques are developed for the management of such risks in a coherent manner with the radiation risks.

6.100. The nature of specialized tasks involving exposure of itinerant workers due to naturally occurring radioactive material with relatively high activity concentrations is such that there may be significant opportunities for optimization of radiation protection. It may be possible to achieve substantial reductions in projected doses with relatively simple modifications to the work plan. An example is in the use of engineered controls to reduce the buildup of scale, sludge and sediments or to facilitate maintenance work involving the removal of accumulated contaminants. Changes in the local rules and procedures for this type of work might also be found to reduce doses with a reasonable allocation of resources. Contractors and the management of the facility should be alert to the possibility that meeting the requirements for optimization could require a high level of attention by the management to specialized tasks involving itinerant workers.

7. MONITORING AND ASSESSMENT OF OCCUPATIONAL EXPOSURE

ASSESSMENT OF EXTERNAL EXPOSURE

Monitoring programme

7.1. Doses received by workers from external exposure should be assessed and can, in most circumstances, be readily assessed from the results of a systematic programme of individual monitoring. Doses may also be assessed from the results of workplace monitoring. GSR Part 3 [2] sets out the requirements with regard to the use of individual monitoring and workplace monitoring for dose assessment purposes (see para. 3.116).

Individual monitoring

7.2. Where individual monitoring of workers is to be performed, each worker should be provided with an integrating personal dosimeter.

7.3. Individual dosimetry should be performed by a dosimetry service approved by the regulatory body. The regulatory body should require such a service to supply dosimeters capable of measuring $H_p(10)$, $H_p(3)$ and/or $H_p(0.07)$, as appropriate, with adequate accuracy for all relevant types of radiation. Recommendations and guidance on the management system for dosimetry service providers are given in Section 8.

7.4. For controlling individual exposure on a day to day basis, or during a particular task, it should be considered whether it is necessary to use supplementary dosimeters of the direct reading type (i.e. active dosimeters). Direct reading dosimeters can provide estimates of an individual's dose with a frequency greater than that provided by typical routine dosimetry and can give information on dose rates. Such a dosimeter can be useful for optimization purposes.

7.5. While an active dosimeter is usually used only for purposes of dose control, it can also be used, with prior approval by the regulatory body, as a replacement for the dosimeter designated by the regulatory body for record keeping purposes (the dosimeter of record). In such cases, the same procedures for approval by the regulatory body should apply. The active dosimeter should be of a suitable design for use as the dosimeter of record. It should, for instance, have an adequate energy range, sensitivity, linearity and precision; it should be reliable; and

sufficient quality control measures and periodic calibration procedures should be in place. It should be taken into consideration that active dosimeters (especially electronic dosimeters) often perform poorly in pulsed radiation fields. This should be an important consideration when, for instance, measuring the dose to the lens of the eye, $H_p(3)$, in image guided interventional procedures in medical uses of radiation. Performance tests for electronic dosimeters for pulsed fields of ionizing radiation should be conducted in accordance with Ref. [45].

7.6. In most cases, a single dosimeter worn on the trunk is adequate. This dosimeter should be placed in the position at which the highest exposure at 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. Conversely, if the radiation is primarily from the back, the dosimeter should be worn on the back of the torso (see para. 7.121).

7.7. In an inhomogeneous radiation field, it may be useful for workers to wear additional dosimeters on other parts of the body in order to obtain a better assessment of the effective dose received. In some situations — for example in medical uses of radiation, where protective clothing such as lead aprons can be used — it is advisable to use one dosimeter under the protective clothing and one on an unshielded part of the body. The readings from the two dosimeters can then be combined by the use of suitable algorithms to give an estimate of the total effective dose. There are many algorithms available and the accuracy depends on many factors, such as the thickness of any lead apron worn, the use of a thyroid shield and exposure parameters. Further information on the use of such algorithms can be found in Refs [46–48].

7.8. If a worker is liable to receive an equivalent dose to the extremities, skin or lens of the eye that is a sizeable fraction of the relevant dose limit specified in paras 3.35 and 3.39, the individual dosimetry employed should be capable of providing the information needed for an assessment of the equivalent dose to the tissue or organ concerned. In situations with non-homogeneous exposure conditions for which whole body monitoring does not provide an adequate estimate of the dose to the skin, extremities or lens of the eye, these tissues and organs should be monitored separately. For example:

(a) Monitoring of hands and fingers should be considered for workplaces where extremities are particularly close to the radiation emitter or radiation beam, such as in situations where radioactive sources are handled in research, nuclear medicine and dismantling operations.

(b) Monitoring of extremities, including monitoring of feet, should be considered in interventional cardiology or radiology and in workplaces in nuclear medicine.
(c) Monitoring of skin should be considered for workplaces where skin is close to the radiation emitter or radiation beam or can become contaminated, for instance in the handling of unsealed sources.
(d) Monitoring of the lens of the eye should be considered in workplaces where the eyes are particularly close to the radiation emitter (which can also be a source of stray radiation) or the radiation beam. Workers for whom exposure of the lens of the eye might be important and for whom monitoring should be considered include workers in the medical sector, such as staff working in close proximity to patients in image guided interventional procedures, staff carrying out certain activities in nuclear medicine, staff involved in manual brachytherapy, staff involved in computed tomography guided biopsy and cyclotron engineers. Other examples of workers who could receive significant doses to the lens of the eye include workers in nuclear facilities such as those involved in the fabrication of mixed oxide fuels, in laboratory studies using glove boxes and in decommissioning.

7.9. When extremity dosimeters are used, they should be worn in positions that will measure the dose to the areas of the body expected to receive the highest dose. Often, the location of the maximum dose to the skin or to an extremity is not known in advance, or it is not practicable to wear a dosimeter at these locations. In such cases, a correction factor should be used to estimate the maximum dose [49].

7.10. When it is necessary to monitor the dose to the lens of the eye, the personal dose equivalent $H_p(3)$ should ideally be measured. However, suitable $H_p(3)$ dosimeters are not yet widely available and in certain circumstances the measurement of $H_p(0.07)$ or sometimes $H_p(10)$ can provide a sufficiently accurate estimate of $H_p(3)$ [11]. More details are provided in Ref. [50]. The need for a separate eye lens dosimeter and its positioning on the body depend on the type, energy, direction and homogeneity of the radiation field, as well as on the use of shielding:

(a) For neutron radiation, where homogenous radiation fields are usually present, separate eye lens dosimetry is not necessary because neutron whole body monitoring usually gives a conservative estimate of the dose to the lens of the eye, irrespective of the energy and direction of incidence of the radiation (see para. 246 of Ref. [9] and also table 1 in Ref. [51] in comparison with table A.41 in Ref. [9]).

(b) For photon radiation, separate dosimetry for the lens of the eye is usually the only suitable method for determining the dose to the lens of the eye:
   (i) If the radiation field is inhomogeneous, the dosimeter should always be located near the eyes, if possible in contact with the skin and facing towards the radiation source.
   (ii) It is usually acceptable to measure $H_p(0.07)$ but not $H_p(10)$ [11, 52]; however, the measurement of $H_p(10)$ may also be acceptable if the mean photon energy is greater than about 40 keV and if the radiation is incident mainly from the front or the person is moving in the radiation field [52].
   (iii) If eye shielding in the form of lead glasses is used, the dosimeter should preferably be located behind the eye shielding; where this is not practicable, the dosimeter should be worn above, or next to, the eyes and possibly covered by a filter that mimics the attenuation provided by the lead glasses.
   (iv) If shielding for the trunk (e.g. a lead apron) is used, monitoring near the eyes is necessary because monitoring behind the shielding underestimates the dose to the lens of the eye.

(c) For beta radiation, monitoring is necessary only if the maximum beta energy exceeds 700 keV, since beta radiation of lower energy does not penetrate to the lens of the eye:
   (i) If eye shields (e.g. glasses) are used that are thick enough to absorb the beta radiation,[16] only photon radiation should be considered, but account should be taken of any bremsstrahlung contributions (both outside and behind the shielding) produced by high energy beta radiation.
   (ii) If adequate eye shields are not used, separate dosimetry for the lens of the eye is necessary and $H_p(3)$ is the quantity that should be measured.
   (iii) As beta radiation fields are usually rather inhomogeneous, the dosimeter should be positioned near the eyes.

7.11. For some categories of worker, it might be sufficient to use computational tools to estimate the individual dose. For example, cosmic radiation fields in aircraft are fairly uniform and predictable. Computer codes have been developed for assessing the doses received by aircrew from cosmic radiation and have been validated against measurements (see para. 5.80).

[16] For example, about 10 mm of polymethylmethacrylate is sufficient to absorb beta radiation from ^{90}Y.

7.12. The period of deployment of the dosimeter (the monitoring period) should be established by the management on the basis of advice, as appropriate, from a qualified expert or radiation protection officer and dosimetry service provider. Account should be taken of the type of work being performed, the anticipated exposure associated with the work, the characteristics of the dosimeters (e.g. fading characteristics), the overall limit of detection of the dosimetry system and, if applicable, any additional requirements by the regulatory body. Unless exposures are particularly low or uniform in time, a monitoring period of one month should generally be used. Where the characteristics of the dosimeter allow, monitoring periods as long as three months may be acceptable for exposures that will generally lead to doses well below the relevant dose limit. A monitoring period of between a week and a month may be appropriate, where the rate of exposure is very non-uniform. Shorter monitoring periods, such as one week or even the duration of a specific procedure, may be advisable when setting up new procedures, when optimizing working conditions or when there is a high potential for exposure. If daily monitoring is required, a direct reading dosimeter should be used.

Workplace monitoring

7.13. Careful consideration should be given to the selection of locations for workplace monitoring and to the number of instruments deployed. The locations selected for workplace monitoring should be representative of worker occupancy, as determined on the basis of expected operational activities. If the radiation field is well characterized and uniform in space and does not vary significantly with time, it should be considered whether the installation of only a few instruments for workplace monitoring, or even just a single instrument, could be justified to be sufficient. In contrast, more monitoring instruments should be used if the dose rate varies significantly with time or in space. The use of portable instruments may be helpful, provided that supporting documentation is maintained to specify the place and time of each measurement.

7.14. The frequency of routine monitoring of the workplace should depend on the occupancy factor and the expected changes in the radiation environment:

(a) Where no substantial alterations to the protective shielding or to the processes conducted in the workplace are to be expected, routine monitoring should be used only occasionally, for checking purposes.
(b) Where changes in the radiation field in the workplace are to be expected, but are unlikely to be rapid or severe, periodic or occasional checks, mainly

at pre-established locations, should be made, which will usually give sufficient and timely warning of deteriorating conditions.

(c) Where sudden unexpected increases in exposure might result in a significant dose being received by a worker, provision should be made for the continuous monitoring of exposures.

(d) Where individual doses are assessed on the basis of the results of routine workplace monitoring, that workplace monitoring should be continuous and should be representative of all working areas in the workplace.

Choice of monitoring system

Personal dosimeters

7.15. The choice of a personal dosimeter should be based on the conditions in the workplace, such as the type of radiation and its energy distribution and directional distribution, the range of expected doses and dose rates, and the environmental conditions.

7.16. A dosimeter of the following types in particular should 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) Extremity dosimeters, giving information on $H_p(0.07)$ for beta–photon radiation.

(d) Eye lens dosimeters, giving information on $H_p(3)$ or $H_p(0.07)$ for beta–photon radiation (and for neutrons, if neutron sources are being handled, $H_p(10)$ can provide an approximate estimate of $H_p(3)$); dosimeters designed specifically for $H_p(3)$ are not yet widely available (however, see para. 2.38).

(e) Neutron dosimeters, giving information on $H_p(10)$.

7.17. For radiation fields for which only photon radiation should be considered, it is usually sufficient to measure only $H_p(10)$. A simple dosimeter is therefore adequate in most practical situations. For a wide range of photon energies, thermoluminescent dosimeters, optically stimulated luminescent dosimeters, photoluminescent glass dosimeters or photographic film dosimeters should be considered, provided that they exhibit an adequate energy and angular dependence. In addition, many active dosimeters (or semi-active dosimeters, such as the ‘direct ion storage’ dosimeter) are available that can reliably measure $H_p(10)$.

7.18. The value of $H_p(10)$ can be estimated with a single detector that should have an energy dependence such that the output signal is acceptably proportional to the absorbed dose in tissue (i.e. it is tissue equivalent), and that should be covered with material of a thickness corresponding to a thickness of 10 mm of soft tissue. Such a dosimeter should be responsive to the backscattered radiation from the body. When the detector is not acceptably tissue equivalent, multiple detectors should be used and their measurement results should be combined using a suitable algorithm.

7.19. 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 less than 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 personal dose equivalent at a depth of 0.07 mm.

7.20. For the measurement of $H_p(0.07)$, a simple, single element dosimeter may be sufficient. 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) can be assessed.[17] For example, a measurement made using 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.

7.21. The selection and use of extremity dosimeters should take account of practical considerations in terms of the persons who wear them. For example, the maximum skin dose on the hand is often at the tip of the finger, but for some groups of workers, it may be difficult to wear extremity dosimeters on the fingers, especially on the fingertips. Also, it is not always known in advance where the maximum skin dose will occur. Problems may arise because of requirements for sterilization or because the dosimeters need to be worn under gloves. There may also be problems of contamination associated with the dosimeter. In such situations, there may be severe constraints on the design and size of the dosimeter. If no suitable dosimeter is available, a pragmatic solution should be found (e.g. the use of a dosimeter at the base of the finger instead of on the fingertip) and correction factors should be applied where necessary.

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[17] In discussing the measurement and effects of beta radiation, 'thicknesses' of material are often expressed in units of milligrams per square centimetre 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.

7.22. Most types of neutron dosimeter cannot provide information on personal dose equivalents due to neutron radiation with sufficient accuracy over the whole energy range of interest. Extra effort is necessary if individual monitoring for neutrons is necessary. Since gamma radiation is always present in neutron fields, a photon dosimeter should always be worn together with a neutron dosimeter. In some neutron fields, the ratio of personal dose equivalent due to neutron radiation to personal dose equivalent due to gamma radiation has been found to vary by orders of magnitude. Personal dose equivalents due to neutron radiation cannot therefore be derived with sufficient accuracy from measurements of personal dose equivalent due to gamma radiation by assuming a constant ratio for a given workplace.

7.23. Doses from exposure due to thermal, intermediate and fast neutrons can be assessed by using dosimeters of various types, such as an albedo dosimeter, a track etch dosimeter, a bubble detector or an electronic dosimeter. However, each type of neutron dosimeter has its own specific limitations in terms of neutron energy range, sensitivity, practical usefulness and photon sensitivity. The choice of a neutron dosimetry system is therefore not straightforward and will depend on many practical aspects.

7.24. One major limitation of existing neutron dosimetry systems is the energy dependence. No neutron dosimeter can, at the same time, measure thermal, intermediate and fast neutrons with the same accuracy as that obtainable for the measurement of photon radiation with photon dosimeters. When the neutron doses are substantial, a more detailed study of the neutron spectrum in the workplace is therefore needed. With this information, a local energy correction factor for the dosimeter readings should be applied. This local energy correction factor could be significantly influenced by the directional distribution of the neutron field.

7.25. The choice of a dosimeter for use in a particular radiation field may require a normalization factor to be applied in order to minimize uncertainties in the measurement of $H_p(10)$ and in the estimation of effective dose.

7.26. There are situations in which the radiation field experienced by a worker could increase unexpectedly and significantly (e.g. by a factor of ten). For the control of doses in such situations, supplementary dosimeters should be worn that can give early information on short term changes in the radiation field in the working environment. An example of a dosimeter of this type is the active warning dosimeter, which provides an audible or visual alarm if a certain level of dose or of dose rate is exceeded.

7.27. For operations of short duration in high radiation fields, special monitoring programmes should be designed that include the use of active 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). Active dosimeters for extremity monitoring are now available.

7.28. Further information on personal monitoring systems for the assessment of external exposure is presented in Appendix II.

Workplace monitoring systems and instruments

7.29. A workplace monitoring instrument should be appropriate for its intended use. Care should be taken to verify that the instrument is suitable for the type of radiation to be measured and that its results are not seriously affected by other types of radiation that might be encountered.

7.30. A workplace dose rate monitoring instrument should generally have the following characteristics:

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

7.31. In areas where the possibility of a sudden unexpected increase in exposure necessitates the continuous monitoring of the workplace (see para. 7.14(c)), workplace monitoring instruments should be permanently installed and should be fitted with appropriate audio and/or visual alarms to warn of unacceptable conditions. The display may be routed to a control room, where appropriate, for initiating prompt action.

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

7.33. Workplace monitoring can also be performed with passive dosimeters, which provide a wide dynamic range. In general, however, such dosimeters are not ideally suited to dose assessment applications, particularly where dose rates might vary significantly with time, as they give no information about the time dependence of the radiation field.

7.34. Spectrometers can be a useful supplement to workplace monitoring instruments and are necessary when the information about the radiation spectrum will further support the performance of the workplace monitoring instrument.

7.35. While it is possible to use workplace monitoring at relevant locations for estimating doses to the lens of the eye, no workplace monitoring instruments are currently available for measuring the directional dose equivalent $H'(3, \Omega)$; therefore, special care is needed in selecting alternative instruments. The considerations that apply in this regard are the same as those for the measurement of the personal dose equivalent $H_p(3)$ (see para. 7.10).

7.36. The measurement of cosmic radiation fields on board passenger aircraft is described in Refs [53–55]. Currently, such measurements are not made on a routine basis for purposes of assessment of exposure, since it has been shown that the doses received by aircrew can be reliably calculated by using computer codes and taking flight routes and altitudes as input data (see paras 5.80 and 7.11). Where such measurements are required on a non-routine basis, instruments measuring the ambient dose equivalent $H^*(10)$ should be used [38]. Instruments sensitive to neutrons as well as to low linear energy transfer radiation are required. Some instruments such as tissue equivalent proportional counters, silicon semiconductor linear energy transfer spectrometers and recombination ionization chambers are capable of measuring dose components for both high and low linear energy transfer radiation. For this reason, such instruments, in particular the tissue equivalent proportional counter, have been suggested as reference instruments for cosmic radiation measurements. Alternatively, for dosimetric purposes the field can be divided into a component for particles of low linear energy transfer (≤5 keV/μm) and a component for particles of high linear energy transfer (>5–10 keV/μm); or alternatively into two slightly different components, the non-neutron component and the neutron component, which includes the contribution to the dose equivalent rate by high energy protons. The deposition by particles of low linear energy transfer can be determined by using ionization chambers, scintillation counters, silicon based detectors, passive luminescence detectors or ion storage devices. The component of high linear energy transfer can be measured by using special neutron survey meters (with an

extended energy range response), passive track etch detectors, bubble detectors (superheated drop detectors) or fission foils with damage track detectors.

7.37. Personal dosimeters are, in principle, not suitable for workplace monitoring, as the measurement quantities are different. The dose equivalent quantity for workplace monitoring is defined free in air, and the conversion coefficient from air kerma has no dependence on the angle of radiation incidence. The quantity for personal monitoring is defined in a phantom, and the conversion coefficient has a strong dependence on the angle of radiation incidence, especially at low energies. Where there are compelling reasons for using a personal dosimeter for workplace monitoring, for example by mounting it on a wall in a controlled area, such use should at least be accompanied by a careful consideration of the associated additional uncertainty. The results of a type test in terms of $H^*(10)$ can be used to estimate this uncertainty (see paras 7.94 and 7.95).

7.38. Further information on workplace monitoring instruments for the assessment of external exposure is given in Appendix III.

Specifications for monitoring equipment

Personal dosimeters

7.39. The essential dosimetric performance specifications for personal dosimeters should be such as to meet the objectives of individual monitoring. Information relating to specifications for dosimetric performance can be found in various publications, including Refs [9, 56–61].

7.40. A basic objective of personal dosimetry is to provide a reliable measurement of the operational quantities $H_p(10)$, $H_p(3)$ and $H_p(0.07)$ for almost all practical situations, independent of the type, energy and direction of incidence of the radiation, and with a prescribed overall accuracy. Other characteristics of dosimeters that should be considered from a practical point of view include their size, shape and weight, and their identification.

7.41. The accuracy that can be expected when making measurements with individual dosimeters in the workplace is discussed in para. 251 of Ref. [56], 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*

(*Paragraph 271, Publication 60* [15]). 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."

Strictly speaking, this statement applies to the assessment of effective dose and equivalent dose, but, for doses below the relevant annual dose limit, it can also be applied to the operational quantities.

7.42. The statement of the ICRP quoted in para. 7.41 should be taken to mean that, for doses of the order of the annual dose limits, the apparent annual doses received by an individual — $H_p(10)$, $H_p(3)$ and $H_p(0.07)$, 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 personal dose equivalents that would be indicated by an ideal dosimeter worn at the same point at the same times.

7.43. For single measurements of the operational quantities, the ICRU [58] recommends that:

"in most cases, an overall uncertainty of one standard deviation of 30% should be acceptable. …The error of instruments may substantially exceed this limit at some radiation energies and for certain angles of incidence, but conform to it when they occur in a radiation field with a broad energy spectrum and broad angular distribution."

7.44. Concerning the determination of a value for the recording level (i.e. the dose above which the recording of doses is required), the ICRP states in para. 232 of Ref. [56] 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."

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) is acceptable, where R is given by:

$$R = L \cdot \frac{\text{monitoring period (months)}}{12} \tag{24}$$

Here, 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. Consequently, the minimum level of detection should be at least the recording level. Guidance on minimum levels of detection and other characteristic parameters in measuring radiation can be found in Ref. [62].

7.45. Thus, the recommendations of the ICRP in Ref. [56] 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.

7.46. 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 [63]. 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 [64]. The upper limit R_{UL} is given by:

$$R_{UL} = 1.5\left(1 + \frac{H_0}{2H_0 + H_1}\right) \tag{25}$$

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. (24)). The lower limit R_{LL} is given by:

$$R_{LL} = \begin{cases} 0 & \text{for } H_1 \le H_0 \\ \dfrac{1}{1.5}\left(1 - \dfrac{2H_0}{H_0 + H_1}\right) & \text{for } H_1 \ge H_0 \end{cases} \tag{26}$$

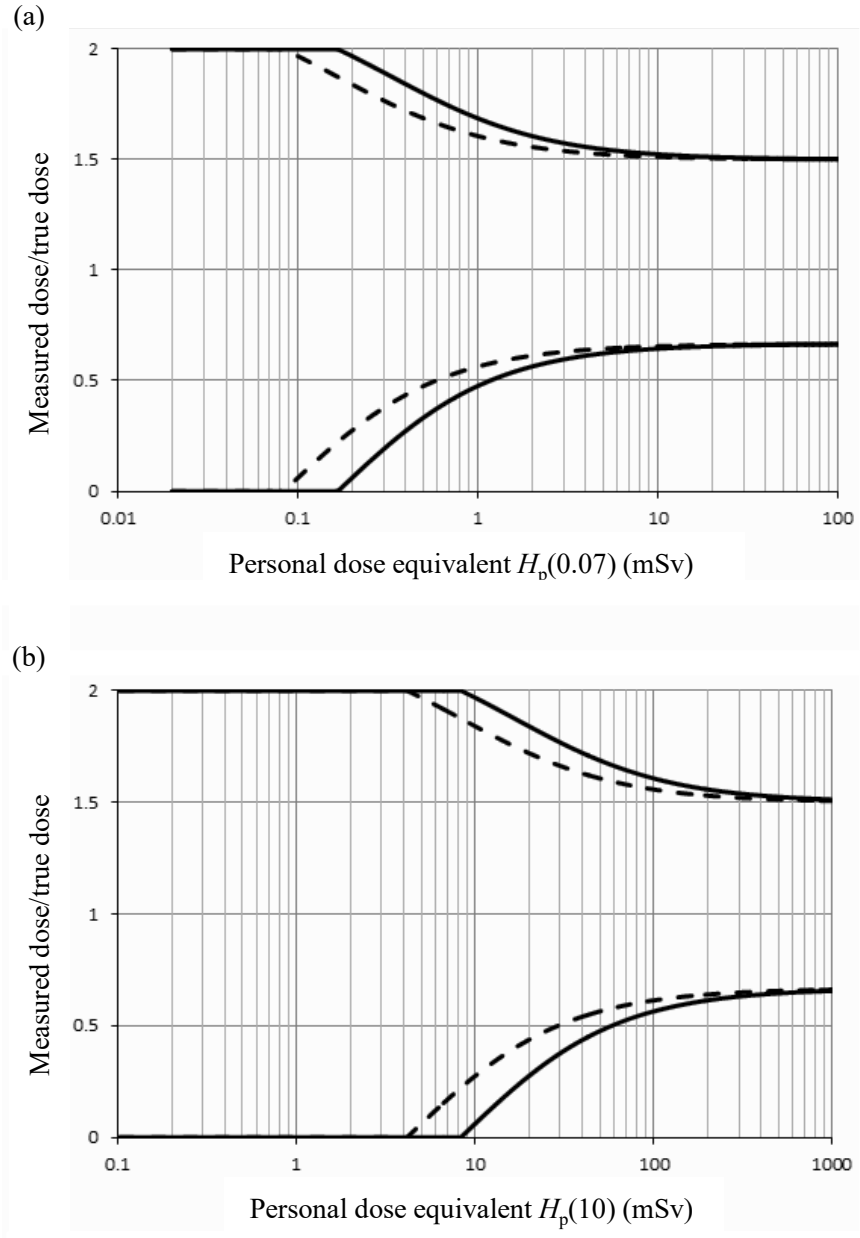
7.47. For $H_p(10)$, with a monitoring period of 1 month, H_0 is 0.08 mSv (using 1 mSv in Eq. (24)). For $H_p(0.07)$, with a monitoring period of 1 month, H_0 is 4.2 mSv (based on 10% of the annual limit of 500 mSv for extremities or the skin). For $H_p(3)$, with a monitoring period of 1 month, H_0 is 0.17 mSv (based on 10% of the annual limit of 20 mSv for the lens of the eye). These recording levels are, of course, dependent on the monitoring periods. The accuracy intervals for $H_p(10)$ and $H_p(0.07)$, the most widely used quantities, are shown graphically in Fig. 4. 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.

7.48. The performance criteria presented in these paragraphs should be used for demonstrating that the recommendations of the ICRP [56] on overall accuracy have been followed. 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.

7.49. For doses to the extremities from exposure to low energy electrons or positrons, the required accuracy is achievable for some designs of dosimeters, but there can be difficulties associated mainly with the thickness of the detector and/or the covering.

7.50. From considerations of the response characteristics of personal neutron dosimeters in current use, and from results of intercomparisons, there are certainly difficulties in meeting the accuracy criteria for measurement of neutron doses. Even with a criterion of 50%, it might not be possible with any current design of dosimeter to meet the criterion over the full range of neutron energies possibly encountered in the workplace. However, for those neutron energies for which there are the greatest difficulties, the contributions to the total dose are generally small. In practice, therefore, a combined standard uncertainty of 50% should be achievable for single measurements in actual workplace fields. The use of a workplace field specific correction factor should enable an overall uncertainty for the assessment of annual effective dose within the limit of a factor of 1.5 to be achieved.

7.51. Where the external field has both a photon and a neutron component, the overall uncertainty is derived from the uncertainties for the two assessments or measurements. If, as is usually the case, the photon component is larger, a larger uncertainty for the neutron component can be accommodated, while still meeting the general criterion for the total dose. In general, contributions from intakes of radionuclides also have to be included. For these contributions, the combined uncertainties may be substantially greater than 50%.



Note: Dashed lines — monthly monitoring periods; solid lines — 2 month monitoring periods.

FIG. 4. Acceptable upper and lower limits for the ratio of the measured dose to the conventional true dose as a function of dose for a depth of (a) 7 mm and (b) 10 mm.

7.52. By using knowledge of the energy and angular spectra of the workplace fields, the uncertainty of a dose assessment can be reduced by applying correction and/or normalization factors. This can be determined by carrying out in-field calibrations or by using information on the workplace field characteristics combined with the energy and angular characteristics of the dosimeter.

7.53. The detailed determination of radiation energy and directional distributions requires the use of specialists and specialized equipment. The measurements can thus be time consuming and expensive. In such cases, an alternative method can be used. The readings of the routine dosimeter can be compared with on-phantom readings of specialized devices which give a better determination of the operational quantities, but are generally not suitable for routine use. Multiple dosimeters can be used on the same phantom to mimic rotation of the worker.

7.54. The determination of field specific correction factors should be the responsibility of the employer but should be carried out in consultation with the radiation protection officer or qualified expert or the dosimetry service, as appropriate, using information supplied by the dosimetry service on the characteristics of the dosimeter.

7.55. 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, the following:

(a) Low cost;
(b) Low weight, convenient size and shape, and convenient and reliable fastening 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 a long period;
(h) Adaptability to various applications (e.g. measurement of body dose and extremity dose);
(i) Availability of, and ease of, calibration;
(j) Suitability for automatic processing.

7.56. For dosimetry of the extremities, particular attention should be paid to the mechanical strength of the dosimeters and to their temperature and humidity resistance, as these dosimeters are often used in abnormal working environments.

Workplace monitoring instruments

7.57. Workplace monitoring instruments used for dose assessment should be calibrated in terms of the operational quantities $H^*(10)$ and $H'(0.07, \Omega)$. They should operate within prescribed criteria for overall accuracy, taking into account the dependence on radiation energy, direction of incidence, temperature, radiofrequency interference and other quantities of influence. As with personal dosimeters, the energy and direction dependences of the response in particular should be considered.

7.58. In line with ICRU recommendations (see para. 7.43) on the acceptable uncertainty value for single measurements of the operational quantities in individual monitoring, an overall uncertainty of one standard deviation of 30% would be appropriate and should be used for workplace monitoring instruments. This value should be applied to performance under laboratory test conditions (standard test conditions), and it may not be achievable under normal operational conditions.

7.59. In addition to the energy and the angular response, several factors can influence the accuracy and reliability of measurements. The factors that should be assessed include:

(a) Ability to withstand shock and vibration;
(b) Independence of atmospheric pressure on the response;
(c) Dust tightness;
(d) Water resistance;
(e) Independence of dose rate on the response;
(f) Correctness of the response in pulsed fields (as applicable);
(g) Insensitivity to electric fields 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.

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

7.61. In some industrial activities involving naturally occurring radioactive material, such as mining and oil and gas production, conditions can be particularly

harsh. In such conditions, the design and construction of workplace monitoring instruments should be suitably rugged. There may also be a risk of flammable atmospheres. Workplace monitoring instruments used in such applications should be designed and constructed to meet the requirements for intrinsic safety. This limits the choice of suitable instruments because most do not meet such safety requirements.

Estimation of uncertainties

7.62. The assessment of uncertainty in measurement is the basis for quantifying the accuracy of the measurements. International guidance on the metrological aspects of dosimetry can be found in publications developed by the Joint Committee for Guides in Metrology.[18] The two fundamental reference documents are the International Vocabulary of Metrology: Basic and General Concepts and Associated Terms (VIM) [65] and Evaluation of Measurement Data: Guide to the Expression of Uncertainty in Measurement [66]. Further guidance relating to uncertainty in measurement can be found in Refs [60, 67–73].

7.63. In the evaluation of uncertainty, all knowledge of the dosimeter and its associated evaluation system (e.g. thermoluminescent dosimeter reader, densitometer and track counting system) or of the workplace monitoring instrument, both from experience and from type testing (see paras 7.72–7.81, 7.94 and 7.95), should be used, possibly in combination with information from the client or customer, such as information on local exposure and storage conditions.

7.64. The evaluation of the uncertainty should use a mathematical model of the dosimetry system. This mathematical model can be given as:

$$Y = f(X_1, X_2 ...) = f(X) \tag{27}$$

where Y is the output quantity or measurand, for instance $H_p(10)$, and X is an array containing the input and influence quantities of the measurement system.

[18] The Joint Committee for Guides in Metrology comprises representatives of the International Bureau of Weights and Measures, the IEC, the International Federation of Clinical Chemistry and Laboratory Medicine, the International Laboratory Accreditation Cooperation, the International Organization for Standardization, the International Union of Pure and Applied Chemistry, the International Union of Pure and Applied Physics, and the International Organization of Legal Metrology.

The evaluation of the uncertainty then consists of two stages: the formulation stage and the calculation stage.

7.65. The formulation stage consists of:

(a) Defining the output quantity *Y*.
(b) Determining the input quantities in array *X*. These are all the quantities that affect the value of the output quantity, in this case the radiation field characteristics. Typical input quantities include the following:
    (i) Dose rate, energy and angle of incidence;
    (ii) Characteristics of the measurement system (e.g. sensitivity as a function of energy and angle, dosimeter fading and characteristics of the dosimeter evaluation system such as developer temperature and reader sensitivity);
    (iii) Characteristics of the calibration system;
    (iv) The dose from exposure due to the natural background radiation, which should be subtracted (see paras 7.128–7.132).
(c) Developing a model relating the input quantities to the output quantity. In most cases, the model is already largely available in the form of the algorithm that is routinely used to calculate the dose from film density or from track detection light output using numerous parameters such as calibration and normalization factors or coefficients.
(d) Assigning a probability density function to each of the input quantities. This assignment is done using all available knowledge of the dosimetry system and the measurement conditions.

7.66. In a 'Type A' evaluation of uncertainty, the assignment of the probability density functions is based on statistical analyses. The standard uncertainty for a Type A evaluation, with an associated standard deviation, is identified from a series of measurements. Examples of parameters with Type A uncertainties are:

(a) Measurements of film density or of light output of a thermoluminescent dosimeter reader;
(b) Blank signal of the reader system;
(c) Sensitivity of the individual detectors.

7.67. For many of the other input quantities, a 'Type B' evaluation should be applied, which is based on a scientific judgement of the uncertainty. Type B uncertainties are those that cannot be reduced by making repeated measurements. The following are usually considered to be sources of Type B uncertainties:

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

7.68. The calculation stage consists of propagating the probability density functions of the inputs through the measurement model $Y = f(X)$ into a probability density function of the output. From this probability density function of the output, the following summarizing quantities should be calculated:

(a) The expectation value, being the central value of the probability density function that is taken as an estimate y of the dose Y;
(b) The standard deviation that is taken to be the combined uncertainty $u_c(y)$ in the dose Y;
(c) A coverage interval that contains Y with a specified probability.

7.69. If it is thought that the probability density function of the dose Y follows a Gaussian (normal) distribution, then one standard deviation each side of the mean corresponds to confidence limits of about 68%. It is therefore 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 two or three, corresponding to confidence limits of approximately 95% or 99%, respectively. The numerical value taken for the coverage factor should be clearly indicated.

7.70. For the calculation stage, essentially two methods are available:

(a) The framework based on the law of propagation of uncertainties and the central limit theorem [66];
(b) The Monte Carlo method, which uses statistical sampling from the probability density functions of the input quantities to evaluate the convolution integral of the probability density functions [69].

7.71. From a radiation metrology perspective, it is not meaningful to report doses in more detail than the standard uncertainty allows. So, for example, in systems

with a standard uncertainty in low doses of less than 0.1 mSv, the doses can be reported in intervals of 0.01 mSv. In systems with a greater standard uncertainty, the doses can be reported in intervals of 0.1 mSv.

Testing of personal dosimetry systems

Type testing

7.72. Type testing of a dosimetry system for external exposure involves testing the performance characteristics of the system as a whole under a series of irradiation conditions and storage conditions. In particular, those sources of uncertainty discussed in paras 7.64–7.67 should be quantified. This largely involves investigation of the variation of response of the dosimeter 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 conditions and humidity conditions, and the ability to respond properly at 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 fields and magnetic fields, and its ability to withstand mechanical shock and vibration. The tests do not concern only the dosimeter itself but the whole system, including any readout equipment.

7.73. The result of a type test should be a detailed description of all of the properties of a given type of dosimeter. The results of type testing should be analysed in terms of performance criteria (see paras 7.48–7.56), and they 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 are to be used. As long as the type of dosimeter and the readout equipment is unchanged, the type test remains valid.

7.74. Dosimetry systems should preferably be type tested in accordance with the relevant standards of the IEC and/or the International Organization for Standardization (ISO) and/or equivalent national standards, and they should have passed the relevant test. Failure of any part of the test should be clearly detailed and the reasons for the failure should be considered.

7.75. All the radiation fields used in type tests should be well characterized and should be traceable to national metrology standards. Several ISO standards give guidance on establishing reference radiation fields for photon, beta and

neutron radiation [74–83]. Additional equipment may be needed for measuring environmental quantities of influence, mechanical effects and electromagnetic fields. Not all of these items of equipment are required at the dosimetric service site; it is sufficient if they are available at the testing laboratory.

7.76. Several standards for type testing exist. For active personal dosimeters, Ref. [84] covers photon, beta and neutron radiation. For passive personal dosimeters, Ref. [85] covers photon and beta radiation. Since these two standards are compatible, the type test results are comparable, regardless of whether the detector is of the active type or of the passive type. For passive personal neutron dosimeters, Ref. [86] is available.

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

7.78. As a result of a type test according to the relevant standard specified in para. 7.76, rated ranges of use for all quantities of influence are determined. The suitability of a dosimeter for a given workplace can be judged by comparing these rated ranges with those required for that workplace.

7.79. Because the operational quantity for individual monitoring relates to the measurement of personal dose equivalent $H_p(d)$ within the body, dosimeters for this operational quantity should be type tested on an appropriate phantom to emulate backscatter from, and attenuation by, the body. If the dosimeter performs adequately on the phantom, it can be assumed that it will perform adequately on the body.

7.80. Personal whole body 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. Extremity dosimeters should be irradiated on the pillar phantom, in the case of wrist dosimeters, or on the rod phantom, in the case of ring dosimeters, in accordance with Refs [74–77]. For doses to the lens of the eye ($H_p(3)$), the design of a suitable phantom has been the subject of discussion. When dosimeters for the quantity $H_p(0.07)$ are used for determining the dose to the lens of the eye:

(a) They should be optimized for such use on the slab phantom (i.e. their energy and angular dependence should be type tested on the slab phantom and they should be calibrated on the slab phantom).

(b) Alternatively, it should be ascertained that the dosimeters correctly detect the radiation scattered back from the body behind them (i.e. the head). This is usually the case for ring dosimeters with a back layer of plastic with a thickness of about 1–3 mm [87].

It has been shown that, in the case of photons, measurement of the quantity $H_p(0.07)$ with dosimeters sensitive to backscatter radiation and calibrated on any ISO phantom provides a conservative approximation of the dose to the lens of the eye [11, 87, 88].

7.81. Conversion coefficients relating the physical quantities (fluence and air kerma) to the operational quantities ($H_p(10)$, $H_p(3)$ and $H_p(0.07)$) are given in various publications, including Refs [76, 79, 80, 83, 85, 86, 89].

Performance testing

7.82. In addition to the type testing of a personal dosimetry system, in which the functioning of the whole system is carefully analysed in order to verify that it meets the accuracy criteria, performance testing should be conducted at regular intervals (typically annually) to demonstrate that this standard of performance is maintained.

7.83. Performance tests carried out externally by an identifiable laboratory serve as a check on the reliability of the dosimetry system and on the consistency of its method of application. The approval of a dosimetry service by the regulatory body should involve a review of both the type testing results and the initial performance testing results. Ongoing compliance with approval procedures should be based on the results of external performance testing.

7.84. External performance testing necessitates careful consideration of the dose range, the types and energies of the radiation to be measured, the uncertainty of the dose estimates, and the measurement process, including traceability and calibration. The results obtained should meet specific performance criteria, with reference to a standard where applicable.

7.85. In addition, performance tests carried out externally or internally may serve as a check on the consistency of the measurement procedures and laboratory practice (as part of an internal quality assurance programme conducted in accordance with a relevant international standard such as Ref. [90]).

7.86. Three types of performance test are in general use:

(a) In a blind test, the dosimetry service provider is not aware of the tests and cannot use selected dosimeters or special evaluation procedures for the tests. One approach is the invention of an independent 'dummy' customer and irradiation of the dosimeters under controlled conditions independent of the service provider. Most service providers use a dummy customer for their internal performance testing for quality assurance.
(b) In a surprise test, the dosimetry service provider is aware of forthcoming tests but does not know the actual test date in advance. It is possible to use selected dosimeters but not to use special evaluation procedures.
(c) In an announced test, the dosimetry service provider is aware of the tests and can use selected dosimeters and special evaluation procedures.

7.87. An intercomparison exercise among dosimetry service providers can be regarded as an announced performance test. Generally, the results of such intercomparison exercises are published but they are not identified with the names of the participants. Participation in such intercomparison exercises is often a requirement for approval and also a part of the quality management system.

7.88. Further guidance on performance testing can be found in Ref. [91].

Routine testing

7.89. The purpose of routine testing 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 for photon dosimeters). This type of test also serves to normalize the overall sensitivity of the system. Routine tests should usually be carried out by the dosimetry service provider, and should be repeated at regular intervals, preferably monthly. In contrast, quality assurance tests to monitor specific aspects of system performance are generally performed every readout day.

7.90. Routine testing, which includes calibration, is a means by which the sensitivity, precision and accuracy are verified, usually for a single radiation type and energy. The tests required in a quality assurance programme may include routine testing.

7.91. The introduction of a dummy customer is one possible routine test. Dosimeters from the dummy customer are exposed to a known dose over each exposure period and undergo the same treatment as the normal dosimeters. A follow-up of the doses reported for this dummy customer gives a good idea of the performance of the normal dosimeters.

7.92. The results of routine tests should be followed up closely, for instance by the use of control charts, where warning levels and action levels are specified to trigger necessary actions by the dosimetry service provider.

Summary

7.93. A summary of the recommended testing programmes for personal dosimetry systems is given in Table 3.

TABLE 3. SUMMARY OF TESTING PROGRAMMES FOR PERSONAL DOSIMETRY SYSTEMS

	Performer of test	Frequency of test
Type testing	Manufacturer or authorized type testing organization	Once, typically prior to marketing to end users
Performance testing	Authorized testing organization	Annually
Routine testing	Dosimetry service provider	Monthly
Routine testing (quality assurance tests)	End user or dosimetry service provider	Daily or every readout day, prior to dosimeter processing

Testing of workplace monitoring instruments

Type testing

7.94. The type testing of workplace monitoring instruments demonstrates the suitability of an instrument to perform adequate measurements in the workplace environment and should involve the same general approach as that described in paras 7.72–7.81 for personal dosimetry systems. The procedures for measurement of the energy response and angular response of workplace monitoring instruments are similar to those used for personal dosimeters, except that radiation exposures in workplace monitoring would normally be free in air (i.e. without phantom).

7.95. The IEC has published standards for most types of workplace monitoring equipment. 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 and angular response) and the environmental, electrical and mechanical performance. The relevant IEC standards and their applicability are given in Table 4.

Pre-use testing

7.96. Workplace monitoring instruments should be tested before they are first used to ensure that they conform to type test data. Testing should cover the range of dose rates that could reasonably be expected to be encountered. Ranges for which an instrument has not been tested should be clearly identified and documented.

7.97. Pre-use tests should be designed to identify credible faults such as miscalibration or incorrect assembly of the detector. Pre-use testing should also provide a baseline for subsequent routine testing. It is usually possible to select a restricted series of tests that can provide adequate confidence in an instrument's performance. Detailed guidance is provided in Ref. [98].

Periodic testing

7.98. Once a workplace monitoring instrument is in use, periodic testing should be carried out to indicate any deterioration in its performance. Periodic testing should be carried out at least once a year and should involve a subset of the tests used in pre-use testing. Examples of reference types of radiation that may be used are:

(a) For photon dose rate monitors, the 0.662 MeV gamma emission from ^{137}Cs;
(b) For neutron dose rate monitors, ^{241}Am–Be neutrons;
(c) For beta dose rate monitors, the 0.662 MeV gamma emission from ^{137}Cs plus a beta source of suitable energy;
(d) For beta contamination monitors, beta emissions at or below the minimum energy for which the monitor is to be used;
(e) For workplaces involving naturally occurring radioactive material, an appropriate reference source.

7.99. Simpler periodic tests should be carried out on a more frequent basis:

(a) Most workplace monitoring instruments should be regularly checked by using a suitable check source to ensure proper functioning. These checks should be carried out monthly, weekly or even daily, depending on the type

TABLE 4. INTERNATIONAL ELECTROTECHNICAL COMMISSION STANDARDS FOR WORKPLACE MONITORING INSTRUMENTS FOR EXTERNAL EXPOSURE

Standard	Applicability
IEC 60532:2010 [92]	Applies to installed dose rate meters, warning assemblies and monitors that are used to prevent a minor radioactive release or mitigate its consequences, or minor degradation of fuel, within the design basis of the nuclear power plant or nuclear facility, and to warn personnel or to ensure their protection during or following events that involve or result in release of radionuclides in the nuclear power plant or the nuclear facility, or risk of radiation exposure. In IEC 61226:2009 [93], this equipment is typically classified as category A, B, C or 'not classified'. The main technical changes with regard to the previous edition are updates to take account of the requirements of IEC standards published since 1996.
IEC 60846-1:2009 [94]	Specifies the design requirements and performance characteristics of dose equivalent (rate) meters intended for the determination of ambient dose equivalent (rate) and/or directional dose equivalent (rate) as defined in Ref. [58]. Applies to dose equivalent (rate) meters and/or monitors for the measurement of ambient dose equivalent (rate) and/or directional dose equivalent (rate) from external beta radiation, X rays and gamma radiation.
IEC 60846-2:2015 [95]	Applies to portable or transportable dose equivalent (rate) meters and/or monitors for the measurement of ambient and/or directional dose equivalent (rate) from external beta radiation, X rays and gamma radiation in emergencies. Applies directly to dose equivalent (rate) meters intended for the determination of the personal dose equivalent or dose equivalent rate from external beta radiation and/or X rays and gamma radiation of energies up to 10 MeV in emergencies.
IEC 61005:2014 [96]	Specifies requirements for the performance characteristics of neutron ambient dose equivalent (rate) meters, and prescribes the methods of testing to determine compliance. Specifies general characteristics, general test procedures, radiation characteristics, electrical, mechanical, protection and safety, and environmental characteristics, and also the identification certificate.
IEC 61017:2016 [97]	Applies to transportable, mobile or installed equipment intended to measure environmental air kerma rates from 30 nGy/h to 10 μGy/h due to X rays or gamma radiation of energy between 50 keV and 1.5 MeV. Specifies general characteristics, general test procedures, radiation characteristics, electrical, mechanical, protection and safety, and environmental characteristics, and also the identification certificate.

of instrument. The choice of source and ranges tested should be appropriate for the type of monitoring being conducted.

(b) Battery checks, zeroing and tests to demonstrate an adequate response should be carried out regularly as part of the quality assurance programme to ensure that the equipment continues to function satisfactorily and has suffered no obvious damage.

7.100. Following testing, a sticker should be affixed to the instrument. This sticker should provide relevant information, including the organization performing the test, the test certificate number and the date of the test or the date when the next test is due, as appropriate.

7.101. A summary of the recommended testing programmes for workplace monitoring instruments is given in Table 5.

TABLE 5. SUMMARY OF TESTING PROGRAMMES FOR WORKPLACE MONITORING INSTRUMENTS

	Performer of test	Frequency of test
Type testing	Manufacturer or authorized type testing organization	Once, typically prior to marketing to end users
Pre-use testing	Manufacturer, end user or authorized testing organization	Once, prior to placing instrument into service
Periodic testing	End user or authorized testing/ calibration organization	Annually or more frequently, depending on the stability of the instrument and its intended use

Calibration of instruments

7.102. Calibration is the operation that, under specified conditions and in a first step, establishes a relationship between the quantity values (with measurement uncertainties) provided by measurement standards and corresponding indications (with associated measurement uncertainties), and, in a second step, uses this information to establish a relationship for obtaining a measurement result from an indication.

7.103. Calibration should not be confused with adjustment of a measurement system, with ‘self-calibration’ or with verification.

7.104. For all measurement methods, instruments should be regularly calibrated, and this calibration should be traceable to recognized national standards. This may be effected either by using reference sources that have been calibrated previously against primary standards, or by using reference instruments that have been calibrated previously against primary standards by a national primary laboratory or at an acknowledged reference laboratory that holds appropriate standards.

7.105. The reference calibration of a personal dosimetry system (passive or active) should be repeated at regular intervals, for example every one or two years. More frequent periodic checks (see paras 7.89–7.92 for routine testing) should be carried out on the dosimetric performance of the dosimetry system. For passive systems, some simple checks of the readout system should also be performed every readout day, for example using irradiated detectors.

7.106. To determine the reference calibration factor, the radiation field should be well characterized. For the periodic determination of the reference calibration factor of a dosimetry system, it is usually sufficient to use a radioactive source such as ^{137}Cs or ^{60}Co for photon radiation, ^{90}Sr/^{90}Y for beta radiation and ^{252}Cf for neutron radiation. These fields should have traceability to a national metrology institute. Such reference fields and the calibration procedures are described in Refs [74–83]. For neutron radiation, it may also be useful to carry out a calibration in simulated workplace fields, in accordance with Refs [99, 100].

7.107. The reference calibration factor may then be combined with a number of correction factors to be applied in specific conditions of use.

7.108. In addition, every dosimeter should have a traceable individual normalization or calibration factor. For reusable dosimeters, the individual normalization or calibration factor should be checked periodically and should be adjusted if necessary.

7.109. Periodic repeated internal calibrations should be undertaken for passive (solid state) dosimeters to adjust the normalization or calibration factors for changes due to repeated use, or to confirm that their performance has not changed. A suggested frequency is every ten uses or every two years, whichever comes first. An individual normalization factor may also be necessary and should be considered for active dosimeters.

Approval of dosimetry services

7.110. According to para. 3.73(c) of GSR Part 3 [2], the regulatory body is responsible for the authorization or approval of service providers for individual monitoring and calibration services. An approved dosimetry service provider can be described as one such service provider who is responsible for the calibration, reading or interpretation of individual monitoring devices and whose capacity to act in this regard is recognized by a regulatory body or other relevant authority.

7.111. The purpose of approval is to recognize and verify that a dosimetry service provider is technically competent, is able to generate technically valid results, and has adequate administrative, technical and management systems.

7.112. For a service provider to be approved, the service provider should be able to provide an acceptable degree of accuracy in the assessment of dose, to achieve and maintain a high level of reliability, to communicate the results of routine dose assessments to the employer and/or the regulatory body within a reasonable period of time, and to rapidly communicate the results of dose assessments made in the event of an accident or other incident or occurrence. In addition to satisfying technical requirements, an approved service provider should satisfy relevant management system requirements (see Section 8).

7.113. The approval process may involve the following aspects which should be considered:

(a) Submission of a report containing information about the dosimetry system. The technical documentation typically covers type test results, dosimetry procedures and calibration traceability, as well as the management system, including the organizational structure, personnel, equipment and quality control protocols and procedures.
(b) Accreditation of the management system in accordance with a relevant international standard such as Ref. [90].
(c) Certification that the dosimetry system is traceable to the appropriate national standard and is based on conversion coefficients for the operational quantities in accordance with international recommendations and standards.
(d) An irradiation performance test at unknown doses in unknown situations.
(e) On-site inspection and assessment of the laboratory by dosimetry experts who evaluate aspects such as staff (including training), equipment, facilities, calibration and dosimetry procedures, in accordance with what is stated in the approval documentation.

7.114. External performance testing as part of approved procedures should be carried out to demonstrate that the essential performance specifications are routinely maintained (see paras 7.83 and 7.84). The results should verify the type testing data.

7.115. An approval performance testing programme may be subdivided into different irradiation categories to suit different classes of dosimeter (i.e. categories 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 dose ranges.

7.116. Approval performance tests should be carried out at regular intervals. Such tests may be organized by the regulatory body or other relevant authority or may involve participation in international external intercomparisons.

Interpretation of measurements and dose assessment

Personal dosimetry

7.117. For radiation protection purposes, the measured operational quantities $H_p(10)$, $H_p(3)$ and $H_p(0.07)$ are interpreted in terms of the protection quantities effective dose, equivalent dose to the lens of the eye, and equivalent dose to the skin and extremities, respectively.

7.118. For photons, $H_p(10)$ will, in most practical situations, provide a reasonable estimate of the effective dose E that avoids both underestimation and excessive overestimation. For neutrons, $H_p(10)$ can underestimate the effective dose for some energy ranges and field geometries. In such cases, information on the energy distribution and directional distribution of workplace fields is necessary to apply suitable corrections.

7.119. The close correspondence between E and $H_p(10)$ is based on the assumption of uniform whole body exposure. Coefficients have been calculated for conversion from the fundamental quantities (particle fluence, air kerma and tissue absorbed dose) to effective dose in anthropomorphic phantoms representing adult humans, and to the operational quantities using ICRU phantoms. The ratios of the operational quantities and protection quantities are an indication of the quality of estimation of the protection quantities for different energies and directional distributions [89].

7.120. For doses near or above the dose limit, or above a fixed investigation level, confirmation should be obtained that measurements of the operational quantities provide a good estimate of the protection quantities. This should be done, in particular, for neutron doses or inhomogeneous exposures. To do this, information should be obtained on the uniformity of the field, the energy and direction distribution of the field, the position at which the dosimeter is worn and the dosimeter response characteristics.

7.121. In cases where the worker moves about the workplace, four types of multidirectional field are generally considered:

(a) Radiation incident predominantly from the front half space (anterior–posterior geometry);
(b) Radiation incident from the rear half space (posterior–anterior geometry);
(c) Radiation incident symmetrically from all directions perpendicular to the body (rotational geometry);
(d) Radiation incident from all directions including above and below (isotropic geometry).

It can be assumed that $H_p(10)$ measured by a personal dosimeter worn on the chest approximates the effective dose sufficiently accurately, at least for anterior–posterior, rotational and isotropic geometry. For posterior–anterior geometry (e.g. for the driver of a vehicle transporting radioactive material), the dosimeter should be worn on the back. Thus, one dosimeter worn on the front (or rear) of the trunk generally provides a satisfactory assessment of the effective dose. More detailed information on the interpretation of dosimeter results obtained under various geometric exposure conditions is provided in Ref. [101].

7.122. For certain radiation fields, the operational quantities might not be a good approximation of the protection quantities because of the energy spectrum of the field. This is the case, in particular, for neutrons in the energy range of 4–20 MeV and more than 50 MeV. Such factors might be determined by a good experimental characterization of the workplace field. Monte Carlo simulations can also be very useful for this purpose.

Workplace monitoring

7.123. In many cases, workplace monitoring should be used to characterize the workplace for the purposes of determining whether restrictions on the movement of workers within that workplace are necessary. In such cases, it is assumed conservatively that a worker is located for the entire working period in that

part of the workplace where the dose rate is highest. However, when workplace monitoring is used for the purposes of dose assessment, realistic estimates of the periods of occupancy should be obtained and used. In workplaces where the dose rates may vary significantly with time, the occupancy of the workplace should be recorded, so that the periods of occupancy can be applied to the relevant dose rate to assess exposure. Additional information on workplace monitoring is provided in Ref. [56].

7.124. If appropriately designed and accurately calibrated instruments are used, a quantity measured in the workplace, together with appropriate occupancy data, can provide the basis for an adequate estimate of the effective dose received by a worker or of the equivalent dose in the tissues and organs of a worker. The operational dose quantities $H^*(10)$ and $H'(0.07, \Omega)$ defined for workplace monitoring will provide an adequate estimate of the effective dose and the skin dose. As explained in para. 7.37, instruments for measuring quantities defined in free air (e.g. kerma) generally do not have the correct energy response for the measurement of $H^*(10)$.

7.125. It should be taken into consideration that the quantity $H^*(10)$ could significantly overestimate the value of $H_p(10)$, as measured with a dosimeter on an individual, and hence could also overestimate the value of the 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.

7.126. For situations in which the extremities or the unprotected skin of the body might be locally exposed to radiation, the directional dose equivalent $H'(0.07, \Omega)$ should be used to provide an adequate estimate of the equivalent dose. The quantity $H'(0.07, \Omega)$ should also be used to provide an adequate estimate of the equivalent dose to the lens of the eye from exposure to photon radiation (see para. 7.10). For multidirectional fields, the instrument should be rotated in the radiation field and the maximum value of dose indicated by the instrument should be used in order to avoid any underestimation of the dose to the skin or the dose to the lens of the eye. The operator should be aware of the possible existence of point sources or narrow beams which could give rise to misleading readings.

7.127. Workplace monitoring instruments are calibrated in radiation fields that irradiate the volume of the detector 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. operational fields close to point sources or

narrow beams). These situations should be given special attention. It should be considered whether it is 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 a hundred [102]. One technique is to use a matrix of point sources to simulate source geometries of interest [103].

Background subtraction

7.128. The zero dose indication of a dosimetry system comprises the readout system background plus the intrinsic background of the detector. Intrinsic backgrounds of detectors can be determined for detectors individually or in batches. In the latter case, the contribution of the uncertainty to a single result will be larger. For batch determination of intrinsic backgrounds, attention should be paid to the sampling procedure.

7.129. The dosimeter indication will, after subtraction of the zero indication (blank indication), and after the application of correction factors and calibration factors, give the gross dose, also known as the measured value. The gross dose will, in general, include a contribution from natural background radiation in addition to any dose from the worker's occupational exposure.

7.130. The methods for the subtraction of natural background radiation are to use either an average value (usually a national average) or to use specific customer values or location values. For dosimeters issued for a monthly wear period, the use of an average background value, although adding to the total uncertainty, will for many services still enable the accuracy requirements to be met. For longer wear periods (e.g. up to three or four months, which is appropriate where exposures are predictably low), more specific values of natural background radiation (i.e. based on the customer location) or control dosimeters (see para. 7.131) should be used.

7.131. At locations where the natural background radiation is significantly greater or less than the national average, the dose rate due to local natural background radiation should be taken into account. The variation in local background radiation can be taken into account by the use of control dosimeters, which are supplied by the dosimetry service to the customer, and stored at the location where the workers' dosimeters are kept when not in use. In some cases, doses in transit should be subtracted. The determination of the dose rate due to local natural background radiation can also be done using a method based on the analysis of the results for issued dosimeters. Such methods are based on the

assumption that the majority of issued dosimeters are exposed only to natural background radiation.

7.132. Additional considerations should apply for active personal dosimeters, since they can accumulate doses from exposure to natural background radiation only when they are in use, rather than continuously. For active personal dosimeters issued on a shift basis, methods should be established to subtract the correct amount of dose attributable to exposure to natural background radiation, especially when using active personal dosimeters that are claimed to have low detection limits. Alternatively, the contribution from exposure to natural background radiation may be neglected.

ASSESSMENT OF INTERNAL EXPOSURE

Monitoring programme

7.133. The assessment of doses received by workers from exposure due to intakes of radionuclides may be based on the results of individual monitoring involving one or more of the following types of measurement:

(a) Sequential measurements of radionuclides in the whole body or in specific organs, such as the thyroid or the lung;
(b) Measurements of radionuclides in biological samples, such as excretions or breath;
(c) Measurement of activity concentrations in air samples that are collected using personal air sampling devices worn by the worker and that are representative of the air breathed by that worker.

7.134. For some radionuclides, individual monitoring based on measurements of activity in the body or in biological samples may not be feasible because of the types of radiation emitted and the sensitivity of detection of the monitoring methods.

7.135. In some situations, it should be considered whether it is necessary or preferable for the assessment of doses received by individual workers to be based on the results of workplace monitoring (see para. 3.118).

7.136. For workers engaged in industrial activities involving naturally occurring radioactive material, internal exposure due to the inhalation of radionuclides in the ^{238}U decay series and the ^{232}Th decay series in dust particles is often the

dominant pathway because of the inherently dusty nature of many such activities. In such workplaces:

(a) Air sampling, rather than biological sampling or whole body counting, should be used as the best way of assessing doses and providing the information that is needed for optimization.
(b) Particular attention should be paid to the characterization of the airborne dust in terms of its particle size distribution, its activity concentration (which may differ from that of the bulk material) and the lung absorption class(es) of the radionuclides concerned.

Further information is given in Ref. [103].

7.137. The choice of measurement technique should be determined by factors such as:

(a) The radiation emitted by the radionuclides;
(b) The biokinetic behaviour of the contaminant;
(c) The degree to which the contaminant is retained within the body, with account taken of both biological clearance and radioactive decay;
(d) The necessary frequency of measurements;
(e) The sensitivity, availability and convenience of appropriate measurement facilities.

7.138. A facility for individual monitoring should ideally be situated in a building remote from other laboratories or operations that give rise to the emission of radionuclides or penetrating radiation which could interfere with measurements. The monitoring area for direct measurement, containing shielded detectors and associated electronic equipment, would normally occupy a ground floor or basement location in view of floor loading specifications. There should also be waiting rooms for people coming for measurement, showers, toilets and rooms for the changing of clothes, and also separate rooms for collecting or dealing with excretion samples.

7.139. The laboratory for analysis of excretion samples should be constructed in the same way as any other radiochemical laboratory. It should not also be used for the analysis of other, high activity process samples such as reactor coolant, in order to avoid any cross-contamination. Precautions for the handling of potentially infectious material should be taken into account when planning space for dealing with, or storage of, excretion samples.

7.140. Further information on the design and implementation of internal monitoring programmes for workers can be found in Refs [103–106].

Routine monitoring

7.141. Routine monitoring of internal exposure should be conducted on a fixed schedule for selected workers. Monitoring of internal exposure has certain limitations that should be considered in the design of an adequate monitoring programme:

(a) Monitoring does not directly measure the committed effective dose received by the individual. For instance, biokinetic models are necessary: to relate the activity levels in an excretion sample to the activity levels in the body at the time the sample was taken; to relate the body content at the time the sample was taken to the original intake; and to calculate the committed effective dose from internal exposure due to the estimated intake. Further information on the biokinetic models used is given in Appendix IV.
(b) Measurements can be subject to interference from other radionuclides present in the body, including radionuclides of natural origin. It may be necessary to establish the body content of radionuclides of natural origin and of artificial origin from previous intakes, especially where such non-occupational intakes are unusually high. Radiopharmaceuticals administered for diagnostic purposes or for therapeutic purposes can interfere with bioassay measurements for some time after their administration, depending on the properties of the agent administered and on the radionuclides present in the workplace. Workers should be requested to report any administration of radiopharmaceuticals to their supervisors so that it can be determined whether or not adequate monitoring of internal exposure can be performed.
(c) The results of an individual monitoring programme for the estimation of chronic intakes might depend on the time at which the monitoring is performed. For certain radionuclides with a significant early clearance component of excretion, there may be a significant difference between measurements taken before and after a weekend break. Such cases should be reviewed individually [13, 16, 107]. For radionuclides with long half-lives, the amount present in the body and the amount excreted will depend on, and will increase with, the number of years for which the worker has been subject to intakes. In general, the activity retained from intakes in previous years should be taken to be part of the radiation background for the current year.

(d) The analytical methods used for individual monitoring sometimes do not have adequate sensitivity to detect the activity levels of interest. Information on detection limits achievable for individual radionuclides is given in Ref. [13]. More specific information on detection limits for inhaled intakes of ^{232}Th and its progeny for various measurement techniques is given in tables 7, 8, 94 and 95 of Ref. [25].

7.142. In situations in which air samples are used as the basis for assessing the doses received by workers from exposure due to the inhalation of airborne radionuclides, the airborne activity concentration can be determined using stationary air samplers or personal air samplers. The dose is assessed from the airborne activity concentration by using generic or site specific assumptions about the form of the material (particle size and chemical form) and the breathing rate and exposure period of the worker. Stationary air samplers for the monitoring of airborne dust have relatively high flow rates, typically about 20 L/min, and are deployed at predetermined fixed locations in the workplace. Personal air samplers, on the other hand, have relatively low flow rates, typically about 2 L/min, and are worn on the lapel; the pack containing the pump and battery is worn on a belt and is connected to the sampling head by a flexible tube. Care should be taken to ensure that the sampling head is so positioned that the sampled air is reasonably representative of the air breathed. Personal air samplers are not always sufficiently rugged or convenient to wear in harsh working conditions.

7.143. Personal air samplers, in combination with other direct and indirect methods, have increasingly been used in preference to stationary air samplers, since they provide more reliable monitoring [104, 105]. The air sampled by a stationary air sampler might not be representative of the air breathed by the worker, resulting in doses from exposure due to dust inhalation being significantly underestimated, sometimes by several orders of magnitude and particularly in workplaces where the resuspension of dust by the worker's activities is a significant factor. On the other hand, the use of stationary air samplers may result in a significant overestimation of the dose if the worker is not continuously stationed in a dusty area. Where practicable, therefore, personal air samplers should be used in preference to stationary air samplers in all cases where short term spatial and temporal variations of airborne activity concentrations are to be expected and where the concentrations of radon progeny are likely to be highest.

7.144. In some workplaces, particularly those associated with mining and mineral processing operations, there may be difficulties in applying personal air sampling to every exposed worker for all of the time. Where this is the case, monitoring strategies usually involve the assignment of workers to work

categories that reflect the general nature and scope of the work activities. In many cases, however, the exposure is not uniform within a work category, since a worker can, during the course of the work shift, spend time in different exposure environments. A further complication arises in accounting for the wearing of respiratory protective equipment.

7.145. Surface contamination monitoring can be used to indicate the potential for intake of radionuclides and the need for more detailed workplace monitoring. However, surface contamination measurements do not provide a suitable basis for internal dose assessment because of the large uncertainties associated with parameters such as resuspension factors.

7.146. In order to determine the appropriate frequency and type of individual monitoring, the workplace exposure conditions should be characterized. The radionuclides in use and, if possible, their chemical and physical forms should be known. Consideration should also be given to the potential for these forms to change under accident conditions (e.g. the release of uranium hexafluoride into the atmosphere, resulting in the production of hydrogen fluoride gas and uranyl fluoride). The chemical form and physical form (e.g. particle size) of the material determine its behaviour in the respiratory tract and its subsequent biokinetic behaviour in the body. These in turn determine the excretion routes and rates, and hence the types of excretion sample that might need to be collected and their frequency of collection.

7.147. Where bioassay monitoring is used, the method of measurement and the frequency of measurement should be capable of detecting an intake that results in a specified fraction of the dose limit. It should therefore be verified that an intake of this level is likely to be detected. An intake would not be detected if, as a result of radioactive decay and biological clearance, the body content or daily excretion of the radionuclide were to decline to a level below the detection limit of the method employed. The fraction $m(t)$ of an intake remaining in the body for direct measurement or being excreted from the body for indirect measurement depends on its effective half-life and on the biokinetic behaviour of the radionuclide, and is a function of the time period t since the intake. Thus, an intake I and the resulting committed effective dose $E(50)$ would be missed if the product $I \times m(t)$ were less than the detection limit. Typically, the frequency of monitoring should be such as to ensure that intakes corresponding to more than 5% of the annual dose limit can be detected.

7.148. The required frequency of monitoring is thus strongly dependent on the sensitivity of the measurement technique. Techniques for measurement should

be as sensitive as possible. The costs of using the most sensitive techniques and the shortest possible sampling interval should be balanced against the radiation detriment associated with doses that might be underestimated or that might be missed if less sensitive methods or less frequent measurements were to be used.

7.149. A further consideration in establishing a schedule for bioassay sampling is the uncertainty in estimating the intake due to the unknown time of an intake within the monitoring period. It is recommended in Ref. [13] that monitoring periods generally be selected such that, on assuming an intake to have occurred at the midpoint of the monitoring period, any underestimation of the intake would not exceed a factor of three.

7.150. Maximum values of recommended monitoring intervals for various radionuclides and various measurement techniques are given in Refs [108, 109].

7.151. A graphical approach to the determination of monitoring intervals has also been proposed that takes into account uncertainties in material specific parameters (e.g. absorption and particle size distribution), as well as uncertainties in the time of intake [110]. Information on the detection limit for a particular measurement technique is used to determine a monitoring interval appropriate for the dose level of interest.

7.152. In some cases, one or more of the stipulations referred to in paras 7.147–7.151 cannot be satisfied because of a lack of analytical sensitivity, unacceptably long counting times for direct measurements or unacceptably short sampling intervals for collection of excretion samples, particularly in the case of faecal sampling to monitor the inhalation of insoluble (lung absorption type S) particulates. In such cases, dose assessment should be based on alternative types of measurement such as personal air sampling or workplace monitoring.

Task related monitoring

7.153. Task related monitoring is by definition not routine (i.e. it is not regularly scheduled). Such monitoring is conducted to provide information about a particular operation and to provide, if necessary, a basis for decisions on the conduct of the operation. It is particularly useful when short term procedures are carried out under conditions that would be unsatisfactory for long term use. Task related monitoring should be conducted in the same way as routine monitoring, unless the circumstances of the operation dictate otherwise; for example, if the radionuclides involved were different or if the probability or potential magnitude of internal exposure were significantly greater.

Special monitoring

7.154. Special monitoring may be necessary as a result of a known or suspected exposure, or an unusual incident, such as a loss of containment of radioactive material, as indicated by an air or surface sample, or following an accident. Special monitoring is most often prompted by a result of a routine bioassay measurement that exceeds the derived investigation level, but it may also result from a measurement on an occasional sample such as a nose blow or nasal swab, or surface contamination wipe.

7.155. In accident situations, the medical care and treatment of any victim of the accident take priority. Once the victim's medical condition is stable, the following steps should be followed:

(a) External contamination is to be removed, and it is to be ensured that workers have showered and washed their hair before making direct bioassay measurements.
(b) The whole body content of radionuclides is to be established as quickly as possible.
(c) The collection of all excretions is to be ensured.
(d) Other biological samples, such as nasal swabs and mouth wipes, are to be collected (this can be performed during medical treatment or decontamination procedures).
(e) Samples of the contaminant are to be collected for further analysis of the radionuclide composition and of the physical and chemical properties of individual radionuclides.

These steps facilitate a more reliable estimation of the committed effective dose from internal exposure, which, together with the possible dose received from external exposure, is of prime importance in cases of suspected high exposure.

7.156. Special monitoring prompted by an incident is, in terms of measurement techniques, not usually conducted any differently from a routine measurement, although it should be considered whether improved sensitivity or a faster processing time may be necessary. The laboratory should be advised that the sample analysis or the direct measurement has priority over routine measurements, and the frequency of subsequent monitoring may be changed. The laboratory should also be informed that samples may have a higher than normal level of activity. The measurement technique can then be tailored to the special monitoring situation and any necessary precautions can be taken to prevent contamination of other samples. For instance, if counting rates are so

high that dead time losses prevent the proper collection of data, the measurement geometry should be changed and body counting should be performed at a greater distance from the detector, following which a recalibration of the system should be performed. Similar measures should be taken in a radiochemistry laboratory when excretion samples (especially faeces) with high contents of radionuclides are to be analysed.

Methods of measurement

7.157. Intakes of radionuclides can be determined by either direct or indirect measurement methods. Direct measurements of gamma radiation or X ray photons (including bremsstrahlung) emitted from internally deposited radionuclides are frequently referred to as body activity measurements, whole body monitoring or whole body counting. Indirect measurements are measurements of activity in samples which can be either biological (e.g. excretions) or physical (e.g. air filters).

7.158. Each type of measurement has advantages and disadvantages, and the selection of one rather than another should largely depend on the nature of the radiation to be measured.

7.159. Direct methods are useful only for those radionuclides that emit photons of sufficient energy, and in sufficient numbers, to escape from the body and to be measured by an external detector. Many fission products and activation products fall into this category. Incorporated radionuclides that do not emit energetic photons (e.g. ^{3}H, ^{14}C, ^{90}Sr/^{90}Y, ^{239}Pu) can usually be measured only by indirect methods. However, some beta emitters, especially those with high energy emissions, such as ^{32}P or ^{90}Sr/^{90}Y, can sometimes be measured 'directly' via the bremsstrahlung produced. Such bremsstrahlung measurements, because of their relatively low sensitivities, should not usually be employed for routine monitoring.

7.160. Recommendations on the principles of measurement and on the instruments used are given in Ref. [111] and are summarized in Appendix V.

7.161. Direct measurements, where they are possible, offer the advantage of a rapid and convenient estimate of the total activity in the body, or in a defined part of the body, at the time of measurement. The direct measurement of body content or organ content is, therefore, to be preferred for dose assessment if it is sufficiently sensitive, for example the measurement of ^{137}Cs and ^{131}I. However, whole body measurements and measurements for individual organs suffer from

greater uncertainties in calibration, especially for low energy photon emitters. Making direct measurements may necessitate the worker being removed from any work involving radiation exposure for the period over which the retention characteristics are measured. Direct measurements usually necessitate special, well shielded (and therefore expensive) facilities and equipment.

7.162. Direct measurements are useful in qualitative as well as quantitative determinations of radionuclides in a mixture that may have been inhaled, ingested or injected. In addition, direct measurements can assist in identifying the mode of intake by determining the distribution of activity in the body. Intake by the inhalation route of insoluble (lung absorption type S) aerosols containing gamma emitting radionuclides can be easily and accurately detected by the lung counting technique, for instance measurement of U_3O_8; the intake is likely to be undetected using a bioassay technique. Another example is for radioactive iodine, where the thyroid counting system can quantify the uptake of radioiodine by the thyroid. Sequential measurements, where they are possible, can reveal the redistribution of activity, and can yield information about the total body retention and the biokinetic behaviour of radionuclides in the body.

7.163. Indirect measurements generally interfere less with workers' assignments, but they require access to a radiochemical analytical laboratory. Such a laboratory may also be used for measuring environmental samples, but high level measurements (e.g. measurements of reactor water chemistry) should be performed in separate laboratories. Measurements performed on excretion samples are used to determine the rate of loss of radionuclides from the body by a particular route. A biokinetic model needs to be used to determine the body content and intake from the results of such measurements. Because of the ability of radiochemical analyses to detect low levels of activity, measurements performed on excretion samples usually provide a sensitive means of estimating the activity in the body.

Detection limits and decision thresholds

7.164. Measurement methods have certain limits of detection that arise from the level of natural background radiation, from statistical fluctuations in counting rates and from factors relating to sample preparation and analysis. The concepts of the detection limit and decision threshold and their application to in vivo and in vitro activity measurements are documented in Refs [62, 112, 113].

7.165. The measured number of gross counts N_G is the sum of the counts due to background radiation N_B (natural background radiation and radionuclides other than the one of interest) and the net counts N_N due to the monitored radionuclide:

$$N_G = N_B + N_N \tag{28}$$

The activity in the sample or in the body is calculated by dividing the net counts by an appropriate efficiency factor ε.

7.166. The detection limit can be evaluated for a given radionuclide and measurement procedure before the sample measurement takes place. It specifies the minimum activity in the sample (for indirect methods) or in the body (for direct methods) that can be detected with a specified probability β of a false negative. The detection limit allows a prior decision to be made as to whether a measuring method is suitable for the given monitoring programme.

7.167. Once the measurement has been made, the measured net count rate should be compared with the decision threshold. The decision threshold is defined such that if the count rate is greater than decision threshold, then it can be said that the sample or the body contains the monitored radionuclide with a specified probability α of a false positive. If the measured count rate is less than the decision threshold, it cannot be concluded that the radionuclide is absent; however, the activity in the sample or in the body, if present, is less than the detection limit.

7.168. For cases where N_B is sufficiently large (greater than about 30) for the Poisson distribution to be approximated by a normal distribution, decision threshold and detection limit (expressed in terms of count rates) can be calculated as follows:

$$\mathrm{DT} = k_{1-\alpha}\sqrt{\lambda_B\left(\frac{1}{T_B}+\frac{1}{T_S}\right)} \tag{29}$$

$$\mathrm{DL} = \left(k_{1-\alpha}+k_{1-\beta}\right)\sqrt{\lambda_B\left(\frac{1}{T_B}+\frac{1}{T_S}\right)} \tag{30}$$

where

$k_{1-\alpha}$ and $k_{1-\beta}$ are the $1-\alpha$ and $1-\beta$ percentiles of the normal distribution,respectively (the probabilities α and β are generally taken to be 5%, in which case $k_{1-\alpha} = k_{1-\beta} = 1.645$);

λ_B is the background count rate, which can be determined by background measurements in the absence of the activity in the sample;

and T_B and T_S are the durations of the background measurement and of the sample measurement, respectively.

7.169. Representative values of the detection limits for different radionuclides and methods of measurements are given in Ref. [109]. Reference [113] gives examples of the application of the calculation of detection limit, decision threshold and other quantities for a selected number of bioassay techniques.

7.170. The theoretical background to the definition of decision threshold and detection limit is based on the application of Bayes' theorem; this differs from the theoretical background to the definition of minimum detectable activity or minimum detectable amount [114]. For monitoring techniques for internal contamination, however, the values of minimum detectable amount and of detection limit are in most cases equal, provided that the uncertainty associated with the counting efficiency is negligible.

7.171. Further clarification and applications of these concepts for direct methods and indirect methods can be found in Refs [19, 112, 113].

Calibration

Direct methods

7.172. Whole body counters and organ counters should be calibrated with a phantom that simulates the human body or organ and that contains a known quantity of the required radionuclides, either in solution, in sealed sources within the phantom or in the form of a permanent source in a solid tissue substitute matrix.

7.173. The most convenient whole body phantom for general purposes consists of an assembly of plastic containers filled with standardized radioactive aqueous solutions. This concept has been extended to the development of phantoms based on a collection of polyethylene cylinders with circular or elliptical cross-sections.

The appropriate proportions of each section of a phantom representing the adult body are given in Ref. [115]. Phantoms representing different age groups are described in Ref. [116].

7.174. Phantoms have been developed that are not filled with aqueous solutions of radionuclides, and so are less subject to spillage or contamination. Some phantoms are filled with organic gels containing radionuclides [116]. Alternatively, numerous separate point sources can be inserted into polyethylene bricks from which phantoms of various body heights and weights can be easily built, as shown in Ref. [117]. Properly prepared phantoms are also available for the thyroid, the lungs and, for bone seeking radionuclides, the knee or skull [115]. Several publications present different styles and applications of phantoms, tissue substitutes and phantom construction (see Refs [118–123]).

7.175. Methods of calibration using phantoms are relative methods. Some absolute methods do not require a radioactive standard for calibration, but reference standards should always be used to confirm a calibration. Mathematical phantoms, developed for Monte Carlo calculations of detection efficiency, are used for such calibrations (see Refs [124–127]). The advantage of such phantoms is that different distributions of radionuclides in the body and also different sizes, shapes and geometrical relations between internal organs can be simulated. However, thorough comparisons of calculated efficiencies with the measured values should be performed in order to ensure the accuracy of the calibration. This should be done in particular when low energy photons are to be measured or when the radionuclides in the body are not homogeneously distributed.

Indirect methods

7.176. Methods of calibration depend on the instruments used. Standard radionuclide solutions, tracer radionuclides or stable isotopes of the elements to be determined (e.g. strontium) should be obtained.

7.177. When gamma emitting radionuclides are to be measured, the gamma counts obtained from the sample should be compared with those from a standard containing a known amount of the specific radionuclide and measured in the same counting geometry. When gamma spectrometry is used, curves of efficiency versus energy for different geometries should be constructed from measurement standards in those geometries. In preparing efficiency curves, the relative yields of the various gamma emissions from different radionuclide standards should be considered.

7.178. Internal and external standards should be used for beta emitters analysed by liquid scintillation counting. Care should be taken to ensure that the same quenching conditions exist for the standards and the samples.

7.179. Many radiochemical techniques rely on separation procedures for which recovery can be quite variable and depend to some extent on the matrix of samples to be analysed. Methods should be used that allow the determination of the yield of chemical separation or extraction procedures. For this purpose, a known amount of a tracer radionuclide (e.g. ^{243}Am for ^{241}Am) should be added to the sample as early in the procedure as possible to permit direct measurement of chemical recovery.

7.180. Descriptions of various calibration methods for indirect counting are given in Ref. [128].

Performance criteria

7.181. A full description of performance criteria for direct and indirect methods is given in Ref. [62].

7.182. The relative bias is a measure of how close the assessed activity is to the actual activity. This criterion should be verified with phantoms or test samples containing a known value of activity A_{ai}. The individual relative bias B_{ri} for the ith measurement in a series with respect to the correct value of the measurand is defined as:

$$B_{ri} = \frac{A_i - A_{ai}}{A_{ai}} \tag{31}$$

where A_i is the value of the ith measurement in the series being tested.

7.183. The relative bias B_r for this series of measurements is given by the average of the individual relative biases B_{ri}:

$$B_r = \sum_{i=1}^{n} \frac{B_{ri}}{n} \tag{32}$$

where n is the number of test measurements ($n > 5$).

7.184. The repeatability s_{B_r} of the measurement method is defined as the relative dispersion of the values of B_{ri} from their mean B_r:

$$S_{B_r} = \sqrt{\frac{\sum_{i=1}^{n} \left(B_{ri} - B_r\right)^2}{n-1}} \tag{33}$$

7.185. B_r should be between −0.25 and +0.50 for values of A_{ai} that are five to ten times greater than the detection limit. The value of S_{B_r} should be less than or equal to 0.4. When S_{B_r} is greater than 0.4, appropriate corrective actions should be taken.

Uncertainties in monitoring measurements

7.186. As indicated in paras 7.62–7.71, general guidance for uncertainty assessment is given in Refs [65, 66]. Further guidance in line with Ref. [66] can be found in Refs [60, 67–73].

7.187. The result of the uncertainty evaluation should be realistic for the application. The amount of effort put into the uncertainty evaluation should be commensurate with its purpose in terms of radiation protection.

7.188. In programmes for monitoring the intakes of radionuclides, the evaluation of uncertainties in the measurements enables the following:

(a) The making of objective decisions on whether the result is compatible with previous intakes or is to be considered as a new intake;
(b) The identification of data from outliers;
(c) Statistical analyses of the results of the fitting procedures used to evaluate intakes from more than one data point.

7.189. In the case of a measurement of activity in the body or in a biological sample, it can be assumed that Type A uncertainties (see para. 7.66) arise only from counting statistics, which can be described by the Poisson distribution, while Type B uncertainties (see para. 7.67) arise from all other sources of uncertainty.

7.190. In Ref. [109], it is assumed that the overall uncertainty of a measurement can be expressed in terms of a log-normal distribution. The geometric standard deviation of the distribution is given the term scattering factor (SF). The total uncertainty is assessed as:

$$\mathrm{SF} = \exp\sqrt{\sum_i \ln\left(\mathrm{SF}_i\right)^2} \tag{34}$$

where the summation is performed over all Type A and Type B uncertainties. According to Ref. [129], this assumption is considered valid when the Type A uncertainties are relatively small, that is:

$$\frac{\ln(SF_B)}{\ln(SF_A)} > 3 \tag{35}$$

where SF_A and SF_B are the scattering factors for Type A and Type B uncertainties, respectively.

7.191. Reference [110] includes a compilation of typical values of the various uncertainty components for various direct and indirect monitoring methods.

7.192. Examples of Type B uncertainties for in vivo measurements include:

(a) Counting geometry errors.
(b) Positioning of the individual in relation to the detector and movement of the person during counting.
(c) Determination of the thickness of the chest wall.
(d) Differences between the phantom and the individual or organ being measured, including:
    (i) Geometric characteristics;
    (ii) Density;
    (iii) Distribution of the radionuclide within the body and organ;
    (iv) Linear attenuation coefficient.
(e) Interference from radioactive material deposits in adjacent body regions.
(f) Spectroscopy resolution and peak overlap.
(g) Electronic stability.
(h) Interference from other radionuclides.
(i) Variation in background radiation.
(j) Activity of the standard radionuclide used for calibration.
(k) Surface external contamination of the person.
(l) Interference from natural radioactive elements present in the body.
(m) Calibration source uncertainties.

7.193. Examples of Type B uncertainties for in vitro measurements include:

(a) Quantification of the sample volume or weight;
(b) Errors in dilution and pipetting;
(c) Evaporation of solutions in storage;
(d) Stability and activity of standards used for calibration;

(e) Similarity of chemical yield between the tracer and radioelement of interest;
(f) Blank corrections;
(g) Background contributions to, and fluctuations in, radionuclide excretion;
(h) Electronic stability;
(i) Spectroscopy resolution and peak overlap;
(j) Contamination of sample and impurities;
(k) Source positioning for counting;
(l) Density and shape variation from the calibration model;
(m) Assumptions about homogeneity in calibration;
(n) For liquid scintillation counting, differences in quenching between the sample and calibration standard.

7.194. If the samples are collected over periods of less than 24 h, they should be normalized to an equivalent 24 h value (see para. V.22). This normalization introduces additional sources of Type B uncertainty — the uncertainty in the collection period and the uncertainty relating to biological (inter- and intra-subject) variability.

Interpretation of measurements and dose assessment

7.195. The intake of radionuclides and the resulting committed effective dose should be assessed from the results of monitoring measurements in accordance with the scheme presented in Fig. 2 and paras 2.48–2.53. In the case of routine monitoring, it should be assumed that the intake has occurred at the midpoint of the monitoring period.

7.196. In some cases, the measured value M should be processed before being divided by the fraction $m(t)$ to obtain the intake. For instance, urine samples collected over a period of less than 24 h should be normalized to an equivalent 24 h value.

7.197. According to Refs [108, 110], intake and dose should not be assessed if the measured value M is below the critical value M_c, defined in Ref. [108] as that value of the measurement result below which it should be considered unnecessary to evaluate the intake or dose explicitly, since the annual dose may be regarded as insignificant even if that intake were to be repeated for all monitoring periods in the accounting year.

7.198. The annual committed dose value for which the assessment is regarded as insignificant is specified as 0.1 mSv in Refs [108, 109]. Thus, for N monitoring

periods per year, the critical value M_{c_j} (in becquerels) associated with the intake of radionuclide j in any monitoring period is given by:

$$M_{c_j} = \frac{0.000\,1}{N \cdot e(g)_j} m(t_0)_j \tag{36}$$

where $e(g)_j$ is the committed effective dose per unit intake (dose coefficient) for ingestion or inhalation of radionuclide j, as appropriate (in sieverts per becquerel), and $m(t_0)$ is the fraction of the intake remaining in the body or in the excretion sample after an elapsed time period t_0 between the intake and the time of sampling. The intake is usually assumed to occur at the midpoint of the sampling period, in which case Eq. (23) applies for t_0.

7.199. The measured value (if above the decision threshold) should be recorded in order to document the fact that the measurement was carried out and to provide information to support any possible future reassessment of dose.

7.200. Values of M_c for a value of insignificant dose of 0.1 mSv, for various radionuclides and for typical settings for the monitoring programme, are given in Refs [108, 110]. The calculation of these M_c values was based on the parameters defined in Ref. [15], and on the bioassay functions (retention functions and excretion functions) and dose coefficients specified in Refs [13, 130].

7.201. For many radionuclides, values of the retained fractions or excreted fractions $m(t)$ and of the dose coefficients $e(g)$ are given for different lung absorption types or for different values of intestinal absorption. The most appropriate choice of value for a given situation should be based on knowledge of the physicochemical characteristics of the materials present in the workplace. Tables III.2B and III.2C of GSR Part 3 [2] give gut transfer factors and lung absorption types, respectively, for various chemical forms of the relevant elements. In some cases, little information might be available on the characteristics of the intake, in which case the most restrictive value (i.e. the one indicating the highest dose) should be used.

7.202. The values of the retained or excreted fractions $m(t)$ and of the committed effective dose coefficients $e(g)$ given in Refs [13, 14], respectively, are for specific routes of intake and should not be used directly for assessing doses from injection into the blood, from transfer to the blood at wound sites or from absorption through the skin.

7.203. Measurements of airborne activity concentration can be compared directly with values of derived air concentration as an input to the evaluation of workplace conditions. However, the interpretation of measurements of airborne activity concentration for purposes of dose assessment can be difficult because they correspond to the concentration of radionuclides in the air at the location of the sampler, which may not necessarily be in the breathing zone of the worker. However, a personal air sampler placed on the worker's lapel or on protective headgear can collect a sample that is representative of the activity concentration in air that the worker has inhaled, except in cases where the sample comprises only a few particles. Measurements of air activity concentration, combined with measured exposure times and assumptions about breathing rates, can be used to estimate the intake. This is the best method for determining intakes of radionuclides in the ^{238}U decay series and the ^{232}Th decay series by workers engaged in industrial activities involving naturally occurring radioactive material (see para. 7.136). This method can also be used to determine intakes of other radionuclides, such as ^{14}C (in particulate form), ^{239}Pu and ^{235}U, for which direct methods and other indirect methods of assessment of activity in the body are not sufficiently sensitive.

7.204. The control of exposure due to ^{222}Rn progeny in existing exposure situations does not usually require the calculation of effective dose. Reference levels for exposure due to ^{222}Rn progeny are expressed in terms of the time weighted average ^{222}Rn gas concentration (in becquerels per cubic metre). However, a factor for calculating the effective dose arising from a given exposure due to ^{222}Rn progeny is necessary in those special situations in which occupational exposure due to ^{222}Rn progeny is subject to the requirements for planned exposure situations (see para. 3.161). This is because in such situations it is necessary to ensure that the limits on effective dose are not exceeded. In addition, the conversion of occupational exposure due to ^{222}Rn progeny to effective dose enables the exposure to be compared with occupational exposures due to other sources such as exposures to external gamma radiation and exposures due to inhalation of radionuclides in dust.

7.205. The committed effective dose is usually determined from the exposure due to ^{222}Rn progeny rather than from the intake, using the expression:

$$E_{\text{inh}} = H_{\text{RnP}} P_{\text{RnP}} \tag{37}$$

where

E_{inh} is the committed effective dose (mSv) via inhalation of ^{222}Rn progeny;
H_{RnP} is the committed effective dose per unit potential alpha energy exposure (mSv per mJ·h·m^{-3});
and P_{RnP} is the potential alpha energy exposure (mJ·h·m^{-3}).

7.206. Various estimates have been made, and continue to be made, of H_{RnP}, the committed effective dose per unit exposure due to ^{222}Rn progeny. Estimates derived from epidemiological studies on mine workers have tended to give lower values than those derived using a dosimetric approach. The United Nations Scientific Committee on the Effects of Atomic Radiation recommends a dose per unit equilibrium equivalent exposure of 9 nSv per Bq·h·m^{-3} [23], which, when expressed in terms of dose per unit potential alpha energy exposure, equates to a value of 1.6 mSv per mJ·h·m^{-3}. The ICRP recommends the use of dose coefficients based on biokinetic and dosimetric models [131].

7.207. A similar situation exists with respect to intakes of ^{220}Rn and its progeny. An equilibrium equivalent exposure due to inhaled ^{220}Rn progeny of 1 Bq·h·m^{-3} is considered to give rise to a committed effective dose of 40 nSv [132]. On the basis of this value, the committed effective dose per unit potential alpha energy exposure is about 0.5 mSv per mJ·h·m^{-3}. For an annual exposure period of 2000 h, it can be deduced that:

(a) A time weighted average ^{212}Pb activity concentration of 1 Bq/m^3 in air corresponds to a committed annual effective dose of about 0.08 mSv.
(b) A time weighted average potential alpha energy concentration of 1 μJ/m^3 corresponds to a committed annual effective dose of about 1 mSv.

Use of workplace specific, material specific and individual specific data

7.208. The reference parameter values of the models used for the calculation of retention functions, excretion functions and dose coefficients are based on the ‘reference person’ or ‘reference worker’, as defined by the ICRP [16]. The models and their parameters have been developed for defined physical forms and chemical forms of radionuclides. In some circumstances, it is likely that the physical forms or chemical forms of the radionuclides in use in a given workplace will not correspond to the reference parameter values used for the biokinetic models. In such circumstances, an analysis of the particle size and the solubility of samples of airborne radioactive particles can assist in the development of more reliable assessments of dose.

7.209. Even if all the assumptions in the reference biokinetic models are appropriate for a given workplace, there will still be differences between individuals in excretion rates and other biokinetic parameters for the same intake of a radionuclide. In these circumstances, the development of material specific and individual specific models should be considered.

7.210. In addition, the assessment of dose following an accidental exposure needs more specific information about the time and pattern of intake, about the physicochemical form of the radionuclides and about the characteristics of the individual (e.g. body mass). Moreover, routes of exposure other than those for which the values of $m(t)$ and $e(g)$ have been calculated may be relevant in accident situations (e.g. the absorption of radionuclides through intact skin or through a wound). Biokinetic models for these routes of exposure are described in Appendix IV.

7.211. If intakes are small, the reference models are likely to be adequate for estimating the resulting doses. However, if the estimate of an intake corresponds to a significant proportion of the dose limit, the development of parameters for a biokinetic model that are specific to the materials or individuals in question should be considered for estimating the committed effective dose more accurately.

7.212. According to Ref. [108], this specific assessment should be performed if the dose assessed by means of the standard evaluation exceeds the investigation level as defined in Ref. [106].

7.213. According to the graded approach adopted in Ref. [109], it is suggested that information specific to the workplace should be used when the dose assessed with the reference models (standard evaluation) exceeds 1 mSv. It is also suggested that information specific to the individual should be taken into consideration when the dose assessed by means of the reference models (standard evaluation) exceeds 6 mSv. Such specific biokinetic models can be developed from sequential direct and indirect measurements of the exposed workers.

7.214. The deposition of inhaled dust particles in the respiratory tract is influenced by the particle size. A common example of conditions in which there is a need for information specific to the material is where the particle size distribution of airborne dust differs significantly from that assumed in the

reference models (i.e. an activity median aerodynamic diameter (AMAD)[19] of 5 μm and a geometric standard deviation of 2.5 [133]). For example, in industrial activities involving naturally occurring radioactive material, the AMAD of airborne dust is typically in the range of 1–20 μm. Dose coefficients for an AMAD of 1 μm (in addition to those for an AMAD of 5 μm) are specified in Ref. [14]. For AMADs other than 1 μm and 5 μm, the fractions of inhaled radioactive particles deposited in the various regions of the respiratory tract should be determined from the respiratory tract model of the ICRP and an appropriate dose coefficient should be calculated.

7.215. Where information on the particle size distribution is necessary for the correct interpretation of the intake of radionuclides and subsequent dose assessment, the airborne particle size distribution should be determined by using, for instance, a cascade impactor. As a minimum, air sample measurements should include measurement of the concentration of the respirable fraction of airborne particulates.

7.216. More specific information may also be necessary on the absorption types in body fluids of the material after inhalation or ingestion, as appropriate (see Ref. [110] for one such an evaluation).

7.217. In industrial activities involving naturally occurring radioactive material, a worker can be subject to internal exposure due to the inhalation of airborne dust particles containing radionuclides in the ^{238}U decay series and the ^{232}Th decay series. Such radionuclides are generally contained within a matrix of non-radioactive elements and their compounds, in which case these matrices determine the solubility of the particles. It is therefore appropriate to choose, for the contained radionuclides, a single lung absorption type corresponding to the solubility of the mineral matrix [103]. Many types of industrial naturally occurring radioactive material, including metalliferous ores, mineral sands and radium rich scale, are resistant to all but the most vigorous forms of chemical attack. Therefore, for radionuclides contained in dust particles associated with such material, lung absorption type S should be assumed.

[19] The AMAD is the aerodynamic diameter at which 50% of the airborne activity in a specified aerosol is associated with particles smaller than the AMAD, and thus the remaining 50% of the activity is associated with particles larger than the AMAD. In internal dosimetry, the AMAD is used for the simplification as a single ‘average’ value of aerodynamic diameter that is representative of the aerosol as a whole. It is used for particle sizes typically greater than 0.5 μm, for which deposition depends principally on inertial impaction and sedimentation.

7.218. Since the retrospective determination of particle characteristics following an exposure can be difficult, consideration should be given to obtaining advance information specific to the material when setting up monitoring programmes for workers. The analysis of samples from the workplace of air and surface contamination can also assist in the interpretation of bioassay measurements; for example, by measuring the ratio of ^{241}Am to $^{239+240}$Pu when direct measurement of ^{241}Am in the lung is used to assess plutonium intakes or the solubility of inhaled particles [134, 135].

7.219. In some workplaces, intakes are determined from measurements of dust mass concentrations in air.[20] In such cases, the calculation of the intake requires knowledge of the activity concentration (activity per unit mass) of the airborne dust particles. Sometimes, the composition of the airborne dust particles, and hence its activity per unit mass, can be assumed to be that of the process material. Alternatively, the dust particles might need to be subjected to chemical analysis to determine their composition, or the activity concentration of the dust particles (activity per unit mass) might need to be determined directly by radiometric analysis.

7.220. The variability between individuals, and even the variability in the daily excretion rate for the same individual, will often be more significant than the differences between a reference biokinetic model and a model developed specifically for a given individual. To reduce some of this variability, collection periods for excretion samples should be sufficiently long (e.g. 24 h for urine and 72 h for faeces).

7.221. The use of modelling parameters specific to the individual (e.g. the transfer rates of the systemic biokinetic model) should be rare under routine circumstances. If modifications are introduced to the biokinetics or to other anatomical characteristics of the model, the parameter values of the ICRP for calculating the equivalent dose to a tissue or organ, or the committed effective dose, cannot be used, since they are based on the reference person or the reference worker (as defined in Ref. [16]).

7.222. Special attention should be paid to the interpretation of bioassay measurements after the use of interventional techniques for blocking the uptake of radionuclides or for enhancing their excretion, such as the administration of diuretics, laxatives, or blocking or chelating agents, as well as after the removal

[20] In industrial activities involving naturally occurring radioactive material, mass concentrations of dust in air are often monitored for purposes of industrial hygiene.

of contamination or surgical intervention at a wound site. These techniques influence and modify the biokinetic behaviour of the incorporated radionuclides, thus invalidating the use of the standardized modelling approach for estimating intake and dose from the bioassay measurements.

7.223. In such cases, alternative approaches should be employed, such as discarding data on excretion for excretion samples collected during the period in which excretion rates can be assumed to have been influenced by the treatment, or modifying the standard models in order to take into account the effect of the treatment. Examples of analyses performed after the administration of the chelating agent Ca-DTPA (a calcium salt of diethylenetriaminepentaacetic acid) in cases of accidental intakes of actinides can be found in Refs [136–144]. Bioassay measurements for dose assessment purposes are performed after a certain time period, post-treatment with Ca-DTPA, until the excretion of the radionuclide stabilizes in urine samples.

Uncertainties in dose assessments

7.224. The models that have been developed by the ICRP for describing the behaviour of radionuclides in the body, and hence for assessing intakes, provide the most up to date methods available for dose assessment. However, the reliability of the estimates of doses depends on the accuracy of the models and on any limitations on their application in particular circumstances.

7.225. In particular, knowledge of the time of the intake(s) of radionuclides into the body and of whether the intake was acute or chronic is essential for a reliable dose estimate. According to Refs [145, 146]:

(a) The assumption that the intake occurred at the midpoint of the monitoring period might have a tendency to overestimate the true intake.
(b) The assumption of a constant chronic intake over the whole monitoring period produces an unbiased estimate of the true intake provided that the measurement and the excretion function are accurately known or are at least unbiased.

7.226. Another source of uncertainty in the process of dose assessment is the knowledge of the route of intake and the physicochemical characteristics of the radionuclides that have entered the body. For inhaled radionuclides, the particle size is particularly important in influencing deposition in the respiratory system, while for ingestion the gut absorption factor f_1 can substantially influence the committed effective dose. For routine monitoring when exposures are well

within those corresponding to the dose limits, the default parameters set out in GSR Part 3 [2] may be sufficient for assessing the intakes. For exposures approaching or exceeding those corresponding to the dose limits, more specific information on the physical and chemical form of the intake, and on the characteristics of the individual may be necessary and should be obtained to improve the accuracy of the modelling predictions.

7.227. The models used for dose assessments have the following sources of uncertainty:

(a) The structure of the biokinetic model.
(b) The human biokinetic data used in the formulation of the model.
(c) The extrapolation of biokinetic data from animals to humans (interspecies extrapolation).
(d) The extrapolation of biokinetic data from one chemical element to a chemical analogue on the assumption of close physiological similarities (inter-element extrapolation).
(e) The variability in the population.
(f) The following physical and anatomical parameters of the computational models used to assess the dose deposited in a target region by the radiation emitted by an incorporated radionuclide:
    (i) The energy and intensity of the radiation emitted;
    (ii) The interaction coefficients of the emitted radiation in tissues;
    (iii) The elemental composition of the tissues of the body;
    (iv) The volume, shape and density of the target organs in the body;
    (v) The spatial relationship of the organs within the body.

ASSESSMENT OF EXPOSURE IN EMERGENCIES

7.228. High levels of exposure of accidentally exposed workers might be associated with nuclear or radiological emergencies, such as an emergency at a nuclear power plant, a criticality accident [147], an emergency at an industrial irradiation facility, or an emergency involving a lost or stolen source. The assessment of such exposures can begin by using data from personal and workplace monitors, but other more sophisticated and highly specialized retrospective dosimetry techniques, such as chromosomal aberration analysis, electron spin resonance, accident simulation and computer modelling, can also be used, as discussed in Ref. [32] (see also paras 7.239–7.243).

7.229. In situations in which individual doses to emergency workers could greatly exceed those expected under normal working conditions and could approach the thresholds for severe deterministic effects specified as levels of acute dose in table IV.1 of GSR Part 3 [2] and table II.1 of GSR Part 7 [30], special attention should be paid to the capabilities of dosimeters and to the application of measurements and calculation methods needed for the assessment of RBE weighted organ doses [32] (see para. 2.33).

External exposure

7.230. The choice of a personal dosimeter depends on the type of radiation and on the information that is necessary for determining the RBE weighted absorbed dose AD_T for tissue or organ T. The following types of dosimeter may be used:

(a) Photon dosimeters and neutron dosimeters giving information on the personal dose equivalent $H_p(10)$ for evaluation of AD_T in tissues and organs such as red marrow and the lung.
(b) Eye lens dosimeters, giving information on $H_p(3)$ for beta–photon radiation. Since such dosimeters for the lens of the eye are not yet widely available, it might be necessary to use $H_p(10)$ as the starting point in estimating the dose to the lens of the eye in cases of accidental exposure, although in accidents in industrial radiography this is likely to underestimate the dose to the lens of the eye.
(c) Extremity dosimeters, giving information on the skin dose at a depth of 0.4 mm, for beta–photon radiation (and for neutrons if criticality is expected) for evaluation of AD_T in the dermis for the palm of the hand and the sole of the foot.

7.231. Because of the difference between the RBE of neutrons in the development of severe deterministic effects (a value of 3) and the radiation weighting factor w_R for neutrons (a value of about 12 for most neutron spectra), special care should be taken in using individual monitoring of neutron exposure to evaluate AD_T in certain tissues and organs, as discussed in Ref. [32].

7.232. For extremity dosimetry in emergencies, especially for the hand, a simple, single element dosimeter should be sufficient. 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 40 mg/cm^2 (or 0.4 mm) can be assessed (see para. 7.230(c)). However, if such dosimeters are not readily available, suitable alternative methods using $H_p(0.07)$ or $H_p(10)$ dosimeters may be used.

7.233. To avoid the need for a special additional accident dosimeter, the routine personal dosimeter should be capable of providing information on $H_p(10)$ from photons up to at least 10 Gy [148]. It should be recognized that certain dosimeters, such as film dosimeters, might not be capable of achieving this at all energies.

7.234. The wearing of warning (alarm) dosimeters (or dose rate meters) can be effective in preventing 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.

7.235. Information on dosimetry in the event of criticality accidents involving fissile materials is provided in Ref. [148].

Internal exposure

7.236. The conceptual framework for the assessment of internal doses in emergencies is illustrated in Fig. 5.

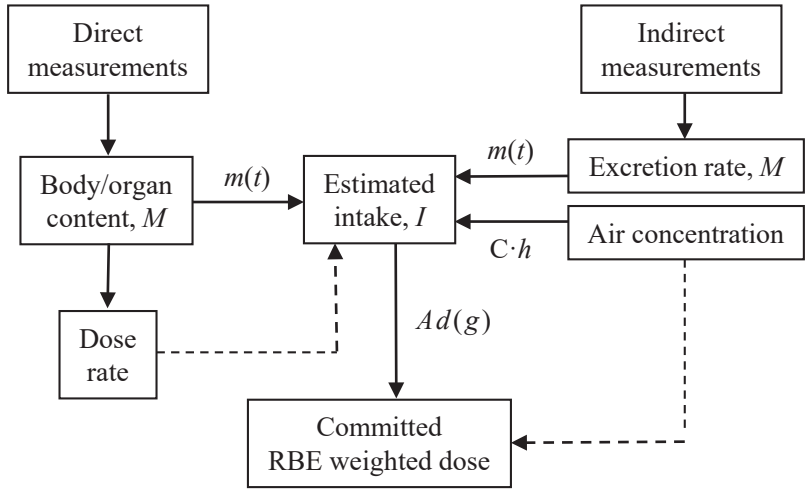


FIG. 5. General scheme for the assessment of internal doses from monitoring measurements in emergencies.

7.237. To derive the value of committed RBE weighted dose, the intake is multiplied by the appropriate dose coefficient (RBE weighted dose per unit intake) for ingestion ($Ad(g)_{T,j,ing}$) or inhalation ($Ad(g)_{T,j,inh}$) of radionuclide j, as appropriate. The committed RBE weighted dose for a period of 30 d after acute intake should be used as an indicator of the probability of developing severe deterministic effects. The committed dose can be seriously underestimated if the dose coefficient $Ad(g)_{T,j}$ is applied directly to the measured value M_j rather than to the inferred intake. Recommended values of the retention rates and excretion rates $m(t)_j$ for use for certain radionuclides after acute intake for inhalation and for ingestion by workers are given in appendix XII of Ref. [32]. In the case of incorporation of a mixture of radionuclides, intakes are assessed separately for each radionuclide and are multiplied by the respective dose coefficients. Committed RBE weighted doses in a tissue or organ from intake of different radionuclides could be summarized. Values of the coefficient of committed RBE weighted dose for a period of 30 d after acute intake for inhalation and ingestion by workers are given in tables 18 and 19 of Ref. [149].

7.238. In the case of a combination of accidental internal exposure and external exposure, an evaluation of the risk of developing severe deterministic effects should be based on the exposure history of accidentally exposed workers as given in Ref. [149].

7.239. Additional information on accidental internal or external exposures can be obtained long after an accident by the application of retrospective dosimetry techniques to biological samples taken from the exposed individuals, to personal effects on the exposed individuals or to other items that were present at the accident site. An overview and a description of such techniques are given in Refs [33, 150, 151] and are summarized in the Annex to this Safety Guide.

7.240. The choice of retrospective dosimetry technique depends on, among other things, the type of radiation emitter involved and the time elapsed since the accident, according to the stability with time of the signal which is measured. The premature chromosome condensation fragment technique, gamma-H2AX assays and the evaluation of changes in blood cell counts or serum proteins should be used only within a few hours of the exposure.

7.241. Luminescence measurements in polymers, hair and nails are effective only for a few days after the exposure, owing to a substantial rate of signal fading. Somewhat slower signal fading is observed for manufactured materials, such as glass, electronic components (such as those in mobile telephones) and memory chips (such as those incorporated into cash cards and credit cards),

enabling them to be used for dose reconstruction purposes for up to a few weeks after the exposure.

7.242. Assays of dicentric chromosomes, micronuclei, translocations or mutations in cells can be successfully employed several weeks or even years after the exposure, as well as electron paramagnetic resonance measurements in tooth enamel, the measurement of activated calcium or the measurement of luminescence signals in quartz extracted from bricks or other fired building materials. The biodosimetry methods may not be appropriate for low dose exposures of less than 50–100 mSv.

7.243. Various numerical methods are used for the retrospective estimate of individual doses. Most of these are based on Monte Carlo radiation transport codes that simulate radiation transport and deposition in tissues starting from known (measured) or estimated information about the radioactive source and its position or distribution in the environment.

SKIN CONTAMINATION

7.244. Skin contamination will lead to external exposure and sometimes even to internal exposure, depending on the radionuclides involved, the chemical forms present and the activity concentrations.

Principal objectives

7.245. The principal objectives for the monitoring and assessment of exposure and contamination of the skin 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

7.246. For strongly penetrating radiation, the limitation on effective dose generally provides sufficient protection for the skin from stochastic effects.

Except in situations involving hot particles (see para. 7.247), no further consideration of skin monitoring is necessary.

7.247. Situations may arise in which exposure due to hot particles is possible. This can lead to spatially non-uniform exposure from discrete radiation sources with dimensions of up to 1 mm. While compliance with dose limits is a principal objective, the ICRP notes that acute ulceration is a particular end point to be prevented [152]. 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 (0.10–0.15 mm), 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 particles. The emphasis should be on identifying and controlling those operations that could give rise to such hot particles.

Weakly penetrating radiation

7.248. For weakly penetrating radiation, the equivalent dose to the skin is limited to 500 mSv in a year, averaged over 1 cm^2 of the most highly irradiated area (Schedule III of GSR Part 3 [2]). The nominal depth of measurement is 7 mg/cm^2 (0.07 mm).

Monitoring skin contamination

7.249. 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 equivalent dose over an area of 100 cm^2. In most monitoring for skin contamination, the reading is compared with a derived limit and the contamination is reduced when practicable. The derived limit should be the level (normally expressed in units of becquerels per square centimetre) that is considered to be capable of causing exposure equal to the relevant dose limit. The derived limit is usually established by taking account of all potential pathways of exposure (not just exposure of the skin). 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 that includes the contamination. These estimates are often extremely imprecise, especially if the radiation from the contaminant might 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 might also be transferred into the body, causing internal exposure.

7.250. The calibration of surface contamination monitors is discussed in Refs [153–155]. The type testing of contamination monitors is discussed in Refs [156, 157].

RECORDS OF OCCUPATIONAL EXPOSURE

7.251. Paragraph 3.105 of GSR Part 3 [2] states that:

> "Records of occupational exposure shall include:
>
> (a) Information on the general nature of the work in which the worker was subject to occupational exposure;
> (b) Information on dose assessments, exposures and intakes at or above the relevant recording levels specified by the regulatory body and the data upon which the dose assessments were based;
> (c) When a worker is or has been exposed while in the employ of more than one employer, information on the dates of employment with each employer and on the doses, exposures and intakes in each such employment;
> (d) Records of any assessments made of doses, exposures and intakes due to actions taken in an emergency or due to accidents or other incidents, which shall be distinguished from assessments of doses, exposures and intakes due to normal conditions of work and which shall include references to reports of any relevant investigations."

7.252. As well as use in demonstrating compliance with the legal requirements, record keeping may be used for several additional purposes, such as the following:

(a) Demonstrating the effectiveness of the optimization process;
(b) Providing data for the compilation of dose distributions;
(c) Evaluating trends in exposure and thus providing the information necessary for the evaluation of the radiation protection programme;
(d) Developing effective procedures and programmes for monitoring;
(e) Providing exposure data from new medical procedures and programmes;

(f) Providing data for epidemiological and research studies;
(g) Providing information that may be necessary for litigation related purposes or for workers' compensation claims, which can arise years after the actual or claimed exposure.

Record keeping for individual monitoring

7.253. The individual occupational exposure record should be linked uniquely to the relevant worker.

7.254. Dose records should preserve the consistency of data fields to enable the reconstruction of results at any later time. Dose records should permit coordination with other required records (e.g. they should permit a linkage with data from workplace monitoring).

7.255. For each monitoring period, the record should comprise:

(a) A unique identification of the individual and the undertaking.
(b) Information on doses from previous monitoring periods (i.e. for an annual period and for an appropriate five year period).
(c) The results of dose assessments for external exposure and the method of assessment, including, as appropriate:
    (i) The personal dose equivalent for exposure to strongly penetrating radiation, $H_p(10)$;
    (ii) The personal dose equivalent for exposure to weakly penetrating radiation, $H_p(0.07)$;
    (iii) Other dose values, such as $H_p(0.07)$ derived from extremity dosimeters, $H_p(3)$ for the lens of the eye, dose values from the use of multiple dosimeters (e.g. in the case of double dosimetry with the use of a lead apron) and estimated dose values calculated from simulations (e.g. for doses received by aircrew from exposure to cosmic radiation).
(d) The results of dose assessments for internal exposure and the method of assessment, including:
    (i) The committed effective dose $E(50)$;
    (ii) The values of the measured quantity (e.g. retention value or daily excretion value) and details of the models used for the assessment, including the results of whole body counting, thorax counting and/or thyroid counting and the assessed committed effective dose;
    (iii) If appropriate (e.g. in the case of overexposure), the committed equivalent dose to the most highly exposed tissue, $H_T(50)$.

(e) The notional dose substituting for missing values, artefacts or surrogates, for instance in the case of lost or damaged dosimeters or samples (see para. 7.258).

7.256. In evaluating the readings of personal dosimeters, it is virtually impossible to distinguish between photon radiation and beta radiation. It is, therefore, not feasible to attempt to identify (and report) the components of $H_p(0.07)$ due to exposure to beta radiation and exposure to gamma radiation separately. However, because the different types of high linear energy transfer radiation have different quality factors, neutron doses should be recorded separately. It should be remembered that doses from photon exposure, neutron exposure and beta exposure are to be combined to determine the total personal dose equivalent.

7.257. The recording level in the context of individual monitoring should be a formally defined level of effective dose (or equivalent dose) or of intake above which a result from a monitoring programme is of sufficient significance to require the measured value or calculated value to be included in a dose record. Other results can be covered by a general statement in the record that no unrecorded results exceeded the recording level. The fact that a measurement has been made should be recorded even in these cases. The best way of doing this may be to enter a zero in the records. If this is done, it should be made clear that the 'zero entry' refers to a dose below the recording level.

7.258. In some circumstances, a dose assessment might not be available for the period when a radiation worker was (or should have been) monitored. This can happen when a dosimeter has been damaged, lost or exposed, or when it has recorded a dose that, on investigation, is declared invalid. In such cases, the record keeping system should provide for the introduction of a notional dose estimated or assessed by the regulatory body or by an authorized person. Notional doses should be marked as such in the dose record so that they can be distinguished from doses assessed from dose measurements made by the approved monitoring service. If no assessed dose or estimated dose is provided, the recorded value should be left blank, so that it is distinguishable from a dose below the recording level (recorded as a zero dose).

7.259. For those individuals who need to use extremity dosimeters (including their use as dosimeters for the lens of the eye), separate records should be kept for the exposure of each extremity (or for exposure of the lens of the eye) for the period during which the extremity dosimeter is being worn.

7.260. Typical records generated in a monitoring programme for internal exposure include both directly relevant data and supporting documentation. The records should ensure the traceability of the measurements and of the dose assessment. Directly relevant information includes:

(a) Data on samples, such as the date and time of collection and evidence of a chain of custody;
(b) Raw data from measurement devices, techniques used for the measurements (direct or indirect techniques) and counting rates in specific energy bands;
(c) Measurements of background levels, and standards and calibration data for the counters;
(d) Calculated results, such as activity content of the body or daily excretion rates, and their statistical analyses;
(e) Calculated estimates of intake and the biokinetic models from which they were derived;
(f) Estimated committed effective doses and the dose conversion factors used.

7.261. Individual dose records should include any assessed equivalent doses or intakes. Details of any involvement in abnormal events should be included, even if no estimates of exposure could be made. Records referencing the objectives, the monitoring methods and the models used for data analysis and interpretation should be retained. Such records may be needed for future interpretation of the dose records. Traceability of the measurements and of the dose assessment should be ensured for the future interpretation of the dose records.

7.262. In accidents, or for a potential intake that may be close to or above a regulatory limit, interim results should be entered into the exposure record so that appropriate administrative actions and other actions can be instituted. The results should include the result of the measurement, the implied intake value on the basis of the appropriate biokinetic model, and the implied committed effective dose on the basis of the corresponding dose coefficient. Recommendations for follow-up monitoring and for workplace restrictions can be made if appropriate. The source of the information reported should be clearly identified, as should a point of contact for any additional information.

7.263. The uncertainties in the measured values and the calculated values should be reported. As an alternative, the dosimetry service can produce a leaflet or report in which specific information relating to the measurement procedure and its characteristics (limitations), including the uncertainties, are shown.

7.264. With respect to the confidentiality, availability and integrity of dose records:

(a) Access to premises, files, archives, computers and servers where personal information is handled and stored should be restricted.
(b) The circulation of information, particularly by means of electronic information networks, should be secure.
(c) There should be procedures for backing up dose records and equivalent security for backup copies.
(d) Similar security measures should be taken in the use of active personal dosimeters and associated software.
(e) Provision should be made for the destruction of documentation or other media containing confidential information that no longer needs to be kept.
(f) The recorded data should be protected against unauthorized or unintentional modification, so as to preserve the integrity of the data.

7.265. Consideration should be given to the establishment of a national dose registry as a central point for the collection and maintenance of dose records. The storage of information at the national dose registry should be such as to allow workers, during and after their working life, to retrieve information on the doses they received while occupationally exposed. Long term storage of such information in a national dose registry also serves the following purposes:

(a) It prevents the loss of data on individual doses in the event that the registrant or licensee ceases its activities in the State concerned.
(b) It allows periodic analysis of all data on exposures collected in order to characterize the situation at the national level with regard to occupational exposure.

Record keeping for workplace monitoring

7.266. Information should be recorded that:

(a) Demonstrates compliance with regulations;
(b) Identifies significant changes in the working environment;
(c) Includes details of radiation surveys (e.g. date, time, location, radiation levels, instruments used, surveyor and other comments);
(d) Includes reports received about the workplace where compliance with the standards could be adversely affected;
(e) Details any appropriate actions taken.

7.267. Records documenting the designation and location of controlled areas and supervised areas should be kept. Records should also be kept of radiation surveys, including the date, time and location, the radiation levels measured, and any comments relevant to the measurements made. Records should identify the instruments used and the individual performing the survey. Even if data from workplace monitoring are not used for dose assessment, the data should be maintained for future verification of workplace conditions.

7.268. A suitable record of the calibration of monitoring equipment should include identification of the following:

(a) The equipment;
(b) The accuracy of calibration of the equipment over its range of operation and for the type of radiation that it is intended to monitor;
(c) The date of the test;
(d) The calibration standards used;
(e) The frequency of calibration;
(f) The name and signature of the qualified person under whose direction the test was carried out.

Record retention periods

7.269. Paragraph 3.104 of GSR Part 3 [2] states that:

> "Records of occupational exposure for each worker shall be maintained during and after the worker's working life, at least until the former worker attains or would have attained the age of 75 years, and for not less than 30 years after cessation of the work in which the worker was subject to occupational exposure."

7.270. For records of individual exposure, the retention period should be taken as applying not only to the retention of the worker's occupational exposure records but also to the retention of the calibration records for the personal monitoring equipment used for determining such occupational exposures.

7.271. The regulatory body should decide which parts of the records of occupational exposure should be maintained by the management for regulatory purposes, and it should specify retention periods for each of these parts of the records.

7.272. A retention period of five years is generally recommended for the records of workplace monitoring and the records of the calibration of the workplace monitoring instruments. However, many records of workplace monitoring, for example the full details of a particular radiation survey, are temporary in nature and are relevant only for the lifetime of an established review period, and there may be no necessity to retain such records for extended periods. Other records might relate to decisions about specifications for the workplace, and these records may be relevant for the lifetime of the workplace. For example, records documenting the creation of designated areas should probably be retained for as long as those designated areas exist.

7.273. The retention periods specified in paras 7.269 and 7.272 reflect the recommended minimum requirements to be set by the regulatory body with regard to record retention. The management may choose to retain more detailed records relating to specific operations that could, for example, be used in future optimization of protection. Such operations might include activities for maintenance or refurbishing.

8. MANAGEMENT SYSTEM FOR PROVIDERS OF TECHNICAL SERVICES

GENERAL CONSIDERATIONS

8.1. Any technical service providers for protection and safety should be qualified by certain procedures. The services provided by technical service providers can be divided into two categories:

(a) Consultancy and maintenance services, including:
  (i) Radiation safety consultancy;
  (ii) Shielding calculations;
  (iii) Modelling for dose assessment, containment and ventilation;
  (iv) Maintenance services covering both in-house operations and services contracted with an outside organization;
  (v) Decontamination services for the decontamination of, for example, equipment and pipes.
(b) Calibration and testing and assay services, including:
  (i) Monitoring services, including individual monitoring, workplace monitoring and environmental monitoring;

(ii) Calibration and calibration verification services for monitoring devices and radiation sources.

8.2. The management system for service providers in radiation protection and safety should be graded to the scope of their activities. The service provider should document its management system, which can include policies, processes and procedures, and instructions. The management system should be documented to the extent necessary to ensure the quality of the service provided.

8.3. The management system for a service provider should cover work carried out in permanent facilities, at sites away from permanent facilities, or in associated temporary or mobile facilities.

8.4. Protection and safety should be of paramount importance for all service providers that use radiation in their activities. The management system of a service provider using radiation should meet the requirements and recommendations of all relevant IAEA safety standards [5, 6].

8.5. Where a service provider is part of a larger organization, the organizational arrangements should be such that departments that could have conflicting interests, such as the production department, the commercial marketing department or the finance department, do not adversely influence the service providers' ability to comply with the requirements of their management system.

8.6. The service provider, if it wishes to be recognized as a third party organization, should be able to demonstrate that it is impartial and that it and its personnel are free from any undue commercial, financial or other pressures that might compromise their technical judgement. The third party organization should not engage in any activities that could endanger trust in its independence of judgement and integrity in relation to its services.

8.7. As in many States, this demonstration of fulfilment should be achieved through third party audit or accreditation to internationally accepted management standards such as Ref. [90]. It should be the responsibility of the service provider to carry out its activities in such a way as to satisfy the needs of its customers.

Safety culture

8.8. For a service provider, safety culture should be established by:

(a) Promoting the knowledge of relevant safety standards within the organization;
(b) Carrying out a risk analysis of the procedures applied;
(c) Establishing proper rules and procedures, and observing regulatory requirements to keep risks at a minimum;
(d) Periodically evaluating the implementation and effectiveness of these rules and procedures;
(e) Engaging the relevant management and staff;
(f) Periodically training the staff, in accordance with an established training programme, to follow the rules and procedures correctly;
(g) Discussing the established training programme with trained staff;
(h) Periodically updating the training programmes and coordinating them with the requirements of legal and regulatory bodies, which should check the effectiveness of these programmes;
(i) Disseminating and promoting knowledge of accidents and other incidents to learn from their occurrence, and any reoccurrence, and to improve the safety culture;
(j) Eliciting safety related proposals from the staff through an incentive system.

Applying a graded approach to the application of management system requirements

8.9. The graded approach usually adopted by service providers is such that any differences in the controls to be applied to the products or services should be identified within each process and should be based on the influence of the process on the quality of the final product.

8.10. In the graded approach adopted, account should also be taken of the size and functions of the organization. Smaller organizations will not have the personnel to fulfil all the functions with separate staffing. However, it remains critical that the functions — including promoting safety culture, ensuring independence, documentation and record keeping — be fulfilled to achieve the performance outcomes given herein.

Documentation of the management system

8.11. Documents[21] may be organized in any relevant medium used in the organization provided that an appropriate system of control is used.

8.12. The documentation of the management system is often contained in a quality manual that includes, or that makes reference to, the supporting documents, including:

(a) A description of the management system.
(b) Management documents, for instance documents relating to some of the topics covered in paras 8.49–8.70.
(c) Detailed work processes and job descriptions.
(d) Additional technical documents and data, including:
    (i) Databases of radionuclides and technical databases;
    (ii) Operating manuals for equipment and software;
    (iii) Reagent data sheets;
    (iv) Requirements of the regulatory body or other relevant authority (as established in laws and regulations);
    (v) Managerial and technical standards.

8.13. The additional technical documents are often external documents that are not within the scope of influence of the service provider. Nevertheless, these documents and data should also be controlled.

8.14. The procedure that describes how documents are to be controlled within the organization should include a periodic review of valid documents to determine whether an update (revision) may be necessary.

[21] Documents may include: policies; procedures; instructions; specifications and drawings (or representations in other media); training materials; and any other texts that describe processes, specify requirements or establish product specifications.

MANAGEMENT RESPONSIBILITY

Management commitment

8.15. A 'management commitment' document should be signed by the senior management[22] to acknowledge the management's responsibility to meet the requirements to establish a management system, to provide the necessary resources, to guarantee the review and revision of the system as necessary, and to define the organizational policies and objectives that will govern the system. After it is issued, the document of the management commitment should be brought to the awareness of staff. In this context, 'necessary resources' can include the staff, infrastructure, working environment, information, supplies and partnerships, natural resources and financial resources necessary to accomplish the objectives of the organization.

Customer satisfaction

8.16. For organizations providing technical services in protection and safety, interested parties[23] are typically customers, staff, regulatory bodies, suppliers, the public and owners. Of these, customers should be accorded the most importance. The interests of other interested parties can generally be satisfied by observing existing laws, rules and regulations.

8.17. A process should be established for identifying and documenting the requirements for fulfilling a contract for service. This should include the identification of:

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[22] Senior management means the person, or group of people, who directs, controls and assesses an organization at the highest level. Many different terms are used, including, for example, chief executive officer, director general, executive team, plant manager, top manager, chief regulator, managing director and laboratory director.

[23] An interested party in this context is a person, group, company or other entity with an interest in the performance of an organization, business or system. Those who can influence events may effectively become interested parties — in the sense that their views need to be considered. Interested parties have typically included the following: customers; owners and operators; employees; suppliers; partners; trade unions; regulated industries and professionals; scientific bodies; governmental agencies and regulators (local, regional and national) whose responsibilities cover nuclear energy; the media; the public (individuals, community groups and interest groups); and other States, especially neighbouring States that have entered into agreements providing for an exchange of information concerning possible transboundary impacts, or States involved in the export or import of certain technologies or materials.

(a) Customer requirements;
(b) Related statutory and regulatory requirements;
(c) Organizational resources necessary;
(d) Requirements for communication with the customer.

8.18. The organization should ensure that customers' reactions are considered. Feedback, including both favourable and unfavourable reactions, should be collected and evaluated. To this end, the management should establish a monitoring process under the management system that is designed to assess and analyse all customer reactions. The results should be used to enable the organization to take actions for the continual improvement of effectiveness and safety.

8.19. The organization should have a procedure in place stating how it protects client confidentiality, while recognizing and acceding to any legal requests to advise regulatory bodies of any breach of a regulatory request or limit, such as exceeding personal dose limits.

Organizational policies

8.20. Typically, a service provider would only have one organizational policy. The policy should be simple (concise) and easily understandable by all members of the organization (the staff). The organizational policy of the service provider should include brief descriptions of actions designed to address such matters as:

(a) Defining and maintaining the expected level of customer satisfaction;
(b) Identifying opportunities and needs for continual improvement;
(c) Ensuring commitment to providing the resources necessary to accomplish the task;
(d) Ensuring contributions of suppliers and partners (confirming that suppliers and partners are capable of providing goods and services that meet the established quality standards);
(e) Ensuring commitment to adopt professional good practices when providing services;
(f) Making the commitment to ensure the competence (qualification) of the personnel involved in the execution of services;
(g) Committing to meet the requirements of the relevant standards;
(h) Ensuring the necessary attention to protection, safety, health, quality, environmental, security, societal and economic aspects, as appropriate.

8.21. Once established, the organizational policy should be translated into measurable objectives. The achievement of these objectives should be checked during the management review. Equally, their adequacy for the existing management system should be evaluated during this management review.

Planning

8.22. A plan should be developed to provide the organization with a series of clearly defined objectives. This means that a series of goals or objectives should be established at different levels of the organization. These objectives should be established during the planning process, and they should be consistent with the organization's policy or policies. At the technical level, objectives should be quantifiable.

8.23. Information sources such as internal audit reports, process reviews and feedback from customers should all be considered in identifying appropriate objectives. As an example, an initial objective for a testing laboratory might be to provide a result to the customer that meets certain criteria for testing performance. Over time, if the organization consistently demonstrates its ability to meet those criteria, other factors, such as improving customer satisfaction through shorter turnaround times for tests, might be made additional objectives. Thus, objectives should be established after the consideration of many factors, including the current and future needs of the organization, the needs of the market served and regulatory requirements.

8.24. To ensure that the planning process remains focused on the defined objectives, planning activities should be systematic and should be documented. The senior management should have a responsibility to ensure that adequate resources are provided to make it possible to meet the defined objectives.

Responsibility and authority for the management system

8.25. In an organization that provides technical services in protection and safety, it is often the case that the top manager appoints one person as a management system manager. The management system manager should have appropriate experience in the tasks for which he or she is appointed to do and should have the authority, assigned in a written document, to do the following:

(a) Develop and manage the management system, which includes performing activities designed to ensure compliance with relevant standards, harmonizing procedures and documents, reviewing operations, identifying

and reporting any non-conformance (i.e. the non-fulfilment of a requirement) to the management and conducting training in awareness of the management system for the staff;

(b) Communicate on quality issues as may be required by the regulatory body or accreditation bodies;

(c) Communicate directly with the senior management at all times on issues relating to the management system;

(d) Act as the focal point for reports of problems regarding quality and suggestions for improvement;

(e) Stop any work that is not being performed according to established procedures;

(f) Perform periodic (usually annual) reviews of the management system.

CONDUCT OF PROCESSES

Provision of resources

8.26. Resources are items that are necessary for conducting processes. They include staff, equipment and supplies, information, physical facilities, infrastructure services, workplaces with appropriate conditions, and monetary funds.

Human resources

8.27. Issues such as staffing levels, education, training, experience, qualifications and periodic performance reviews should all be taken into account when considering human resources. The human resources available should be adequate to meet the predetermined requirements for staff.

Infrastructure and working environment

8.28. The infrastructural requirements of each process should be reviewed to identify the resources that will be required for the successful accomplishment of the stated objectives. For calibration and testing laboratories, where the workplace environment could influence the quality of the results, the regulatory body may impose additional requirements, such as special authorities to be used for calibration services to ensure the correct certification and calibration of equipment.

8.29. The process for the control of monitoring and measuring devices should be conducted so as to establish an effective means of ensuring, with a high degree of confidence, that the data that are generated by these devices, and which are to be used as the basis for reported results, conclusions and interpretations, are accurate within prescribed requirements. Monitoring and measuring devices include the instruments, software and calibration standards used to perform measurements and surveys.

8.30. The process should confirm that these devices are suitable for the intended use, and that they are tested, calibrated and verified as functional within specified performance limits. Physical protection of the devices should also be provided, with the goal of eliminating the potential for process errors.

8.31. Software used to collect data, and to perform calculations on the data collected, should be validated before being put into use and should be protected against unauthorized modification. The functionality of the software should be verified following any change made to the computer's basic operating system or network control parameters, or following any activity that could have an effect on the functionality of the software for the application. Consideration should be given to the need to retain (archive) the different software versions so as to be able to access older records generated by specific versions of the software.

8.32. Additional requirements established by other regulatory bodies may concern matters such as general protection and safety in the workplace and in associated facilities, protection of personal privacy and confidentiality of data, and backup of records kept in electronic media.

8.33. With regard to the working environment, consideration should be given to how best to combine the consideration of human factors and physical factors with achieving the goal of enhancing the performance of the organization.

Developing processes

8.34. The products of organizations providing technical services in protection and safety are the services themselves, which are delivered by using established processes. Development of new processes to supply new services (products) should be carefully planned.

8.35. The management of the organization providing technical services should nominate a technical project leader to be in charge of the planning for new processes. It should be the task of the project leader to schedule the planning

for the new process, by applying technical knowledge and experience together with knowledge of the product requirements that is necessary to the technical service concerned.

8.36. In the planning schedule, account should also be taken of the need to ensure the traceability of any measurement results to suitable standards, including the correct and consistent use of the International System of Units (SI), and for providing information on the uncertainties associated with these measurement results.

Process management

8.37. In an organization providing technical services in protection and safety, there are generally two types of process:

(a) Processes of the management system (administrative processes and key processes);
(b) Processes to deliver the services and products of the organization (technical processes and core processes).

8.38. In monitoring the performance of its processes to ensure that the processes remain effective and that customer satisfaction is provided, a service organization should review the following:

(a) Timeliness;
(b) Capability (ability to meet relevant requirements);
(c) Efficiency (resources allocated to the process and the possibilities for their reduction without compromising quality and compliance with regulatory requirements).

8.39. Data can be derived from different types of monitoring during the conduct of processes. The data can be put to use as a basis for decisions in the organization, by means of adequate analysis. The application of statistical methods to raw data may be especially useful in determining trends in the performance of persons and instruments by describing improvements or deteriorations. This may provide an opportunity for early action to prevent non-conformances. The application of similar statistical techniques to the monitoring of customer satisfaction, resource economics and the performance of suppliers, among other things, may also be useful.

Control of products

8.40. In organizations providing services for protection and safety, the product is generally controlled by controlling the production (i.e. service providing) process.

8.41. The processes of the organization should include any necessary measures to ensure that the delivered product or service fulfils the requirements and expectations of the customer:

(a) For consultancy services, these measures could include the following:
  (i) Additional calculations using other algorithms;
  (ii) Checks on data entry;
  (iii) Comparison of the results with previous experience.
(b) For measurement and calibration services, these measures could be:
  (i) Repeated tests (possibly done using different instruments for analysis);
  (ii) Checks on introduced blank or test samples;
  (iii) Plausibility tests on the results (done by applying expert knowledge).

The results should be recorded as proof of the control of the production process.

8.42. The conformance of the product, or of parts of it, should be ensured by specifying the conditions for identification, storage, handling, protection and delivery.

8.43. Moreover, when a product can be fully verified only after delivery, each process that contributes to its production should be verified to specify acceptable and suitable criteria for the equipment and methods used and the qualification of the personnel involved. A list of parameters linked to the proper completion of each step is generally useful to keep the process exact and consistent. Verification usually requires the production of records, such as checklists, to be completed and evaluated for the final value to be assigned. In practice, the checklist can have the form of a record in a database file, and the verification process can be established by means of a software routine.

8.44. If the creation of a product requires several steps, tracking of the product's status may be necessary, if required by regulation, to identify the output of each step. Generating a record, such as a checklist confirming the completion of all necessary steps, can be helpful.

8.45. Customers' property, including intellectual property, should be safeguarded throughout all of the production processes. Customers' property, and methods to

protect it, should be specified in advance. For example, only a limited number of people should be permitted access to data provided by customers.

8.46. In the case of a consultancy for radiation protection, the customer's property could include detailed information about the customer's facilities, data on exposures or sources, or details of any method developed by the customer in relation to the service requested. Moreover, the service provided in relation to radiation protection becomes the property of the customer, and associated information (i.e. reports on doses or calibrations) should be treated as confidential.

Communication

8.47. Communication in an organization providing services in protection and safety can be achieved by:

(a) Organizing regular meetings of key personnel;
(b) Using communication tools (e.g. electronic billboards and intranet);
(c) Having similar methods of internal communication.

Managing organizational change

8.48. When organizational changes are contemplated in service providing organizations, the guidance provided in GS-G-3.1 [6] should be followed to ensure that there is no adverse effect on the quality of the product or service.

MEASUREMENT, ASSESSMENT AND IMPROVEMENT OF PERFORMANCE

Monitoring the management system

8.49. For all phases in the development and operation of technical services, the technical service provider should define, plan and conduct activities for measurement and monitoring relating to the management system that are necessary to ensure conformance with applicable standards, laws and regulations, and to achieve improvements. These activities should include determining the need for, and specifying the use of, applicable methods, including statistical techniques.

8.50. The general process of performance measurement, analysis and improvement includes the following:

(a) Actions taken on an ongoing basis to monitor the overall effectiveness of the system, by identifying areas, through appropriate metrics, in which improvement may be appropriate.
(b) Application of basic statistical methods (e.g. histograms, distributional analysis and mean values) or qualitative analytical methods for monitoring data on customer satisfaction, the performance of equipment, measurement of throughput and similar indicators of the effectiveness of services provided to the customer.
(c) Actions taken on a proactive basis to prevent non-conformances, to improve the system and to optimize the service to the client; the internal audit process, together with activities for improvements, is part of these actions.
(d) Actions taken on a reactive basis to correct non-conformance identified by, among other things, self-assessment, complaints by clients or recommendations of an internal or external audit.

Self-assessment

8.51. Self-assessment is a tool used by those actually carrying out work to identify possibilities for improvement. If a service providing organization wishes to adopt the practice of self-assessment, it should follow the guidance provided in GS-G-3.1 [6].

Independent assessment

8.52. Audits may be spread over the year or undertaken concurrently. Conducting internal audits on a progressive schedule has several advantages:

(a) It helps to emphasize that the internal audit process is a continual activity designed to improve the management system.
(b) It helps to reduce the additional workload for individuals selected to conduct the audit.
(c) It is useful in promptly identifying items of potential non-conformance and areas in which improvements may be appropriate.
(d) It helps to monitor progress in accomplishing any corrective actions that may have been recommended in previous audits.

8.53. Independence to perform the audits can be achieved by creating a cross-audit department which works across functions (where resources allow). The mandate and scope of the auditing team or person should be clarified and communicated.

8.54. Internal audits could also be carried out for specific purposes and should be considered following customers' complaints, repeated non-conformances or major changes in the organization.

8.55. Internal auditors should be rotated through different aspects of technical applications within an organization. This can serve to increase job satisfaction by allowing employees to play an important part in maintaining the organization's management system.

8.56. The internal audit programme should address all of the elements of the management system.

8.57. It is common practice that an audit schedule encompasses all elements of the management system in all parts of the organization on an annual basis. The extent of the audit and the parts of the organization to be audited should be planned with consideration given to changes in staff or methods, workload, customer complaints, findings of previous audits and ongoing corrective or preventive actions.

8.58. Customers whose work may have been affected by problems identified during an audit should be notified in writing. For some findings, a formal system for corrective actions should be used; for others, there may be simpler remedies.

8.59. If it is necessary to check the effectiveness of corrective actions quickly, a follow-up audit should be considered. Corrective measures that have been undertaken should be analysed to evaluate their effectiveness.

Management system review

8.60. In addition to the review inputs described in GSR Part 2 [5], an organization providing services in protection and safety should consider the results of interlaboratory comparisons or proficiency tests.

8.61. Decisions made during the management review and any actions arising from them should be recorded. The report on the management review should include details of:

(a) The persons who were involved in the review;
(b) Factors that were considered;
(c) Decisions that were reached;

(d) Actions that were planned, the persons responsible for the actions and the time schedules that were decided upon;
(e) The provision for review and approval of the report.

8.62. Results should be incorporated into the laboratory planning system and should include the goals, objectives and action plans for the coming year. The management should ensure that planned actions are carried out within the agreed timescale and that their completion is documented. A comprehensive radiation safety audit will bring out the status of the management actions with regard to radiation protection and safety.

Non-conformances, corrective actions and preventive actions

8.63. For services in radiation protection and safety, non-conformances could include the following:

(a) Incorrectly entered data;
(b) Results obtained by applying incorrect algorithms;
(c) Incorrect calibration data or calibration factors;
(d) Measurement results produced by using instruments outside of their application range;
(e) Calibration data obtained by using the wrong irradiation conditions;
(f) Incorrect output data used for analysis;
(g) Incorrectly performed sampling or sample treatment.

8.64. An analysis of the impacts of revealed non-conformances on safety should be performed, followed by the notification of the management at the appropriate level.

8.65. A policy and procedure for the resolution of complaints received from clients or other parties should be in place. A corrective action procedure should be started after a complaint is made by, or feedback is received from, a customer, or upon the discovery of a non-conformance by staff or during an audit. Corrective actions should be commensurate with the magnitude of the problem and the associated risks. Records should be maintained of all complaints and of the resulting investigations and corrective actions.

8.66. A preventive action might have to follow a corrective action, or may be taken alone, or during the development of new testing or management procedures, or because of a decision taken during a management review. Preventive actions and corrective actions follow similar courses, the one prospective and the

other retrospective. While preventive actions are intended to eliminate the risk that non-conformances will occur, corrective actions are applied to existing non-conformances.

8.67. A corrective action begins with an investigation to determine the cause(s) of a problem. Depending on the nature of the problem, this investigation may be informal or it may be formal and extensive.

8.68. Some questions that should be considered in determining the root causes of a problem include:

(a) Has the issue been validated as a problem?
(b) Have the client's requirements changed?
(c) Have the characteristics of the sample changed?
(d) Has the working environment changed?
(e) Are the methods and procedures for performing the task adequate?
(f) Is there a need for additional staff training or development of skills?
(g) Does the relevant equipment function properly?
(h) Has the calibration of equipment been verified?
(i) Have the specifications of consumable supplies used in support of the operation in question been changed?

8.69. Preventive action is a proactive process to identify opportunities for improvement rather than a response to the identification of problems or to complaints. Apart from the review of the operational procedures, the preventive action might involve the analysis of data, including trend analyses and risk analyses, and analysis of the results of proficiency testing. The planning, development, implementation and monitoring of preventive actions will probably involve a pattern of activities similar to that for corrective actions, except that the activities are proactive in nature.

Improvement of services

8.70. The organization should always try to improve the services to the customer and the internal processes necessary to arrive at the product. The organization should review its performance and events that have taken place, and it should identify and make improvements.

ADDITIONAL GUIDANCE FOR PROVIDERS OF CALIBRATION AND TESTING SERVICES

Organization

8.71. It is important that organizations providing calibration or testing[24] services should seek accreditation by third parties to internationally recognized standards, such as Ref. [90].

8.72. To be sure that tests and calibrations are performed according to established quality standards, laboratories should provide for adequate supervision of calibration and testing staff, by persons who are familiar with methods and procedures, with the purpose of each test or calibration, and with an assessment of the results of tests or calibrations.

8.73. Laboratories should appoint deputies for key personnel, including the technical director and the quality manager, to provide continuity of qualified management even when primary individuals may be absent.

Review of requests, tenders and contracts

8.74. When reviewing requests, tenders and contracts, laboratory personnel should ensure that the appropriate test method or calibration method is selected and that it is capable of meeting the clients' requirements. The review of contracts should extend to any work that is to be subcontracted by the laboratory.

Subcontracting of tests and calibrations

8.75. Laboratories proposing to subcontract tests and calibrations should inform the affected clients in writing of the arrangements and, as appropriate, should gain the approval of the client, preferably in writing.

8.76. For calibration and testing laboratories, subcontracting means placing work that is within the scope of its accreditation with a third party outside the immediate control of the primary contracting laboratory. It does not include, for example, contracting with a reference laboratory to provide intercomparison samples, contracting with an employment agency to provide supplementary support workers, or similar activities. The level of competence of the subcontractor should be adequate for the technical services to be provided. This can be

[24] In some States, the term 'assay' is used instead of test.

demonstrated either by the subcontractor holding an equivalent accreditation in its own right or by the prime contractor completing a quality system audit of the subcontractor's operation.

8.77. The laboratory should maintain a register of all the subcontractors that it uses for tests or calibrations. The evidence should be recorded of how each subcontractor establishes its compliance with international standards (technical and managerial) that are applicable to the work in question.

Service to the client

8.78. In addition to maintaining good communication with clients, laboratories may be required to allow clients to monitor their performance. This can be accomplished by allowing the client reasonable access to the laboratory for the purpose of witnessing tests or calibrations, by providing the client with an opportunity to submit items for verification purposes, by using surveys of client feedback or by other means.

8.79. All activities involving monitoring by clients should be conducted in a manner that preserves the confidentiality of the laboratory's relationship with other clients. Feedback from the monitoring by clients should be documented and should be used to improve the management system.

Control of records

8.80. With regard to technical records, the laboratory should retain, to the extent practicable, the records of original observations, derived data and sufficient information to establish an audit trail, calibration records and a copy of each test report or calibration certificate issued for a specified period. The records for each test or calibration should include sufficient information to permit, if necessary, the identification of factors affecting uncertainty and to enable the test or calibration to be repeated under conditions as close as possible to the original conditions. The records should include the identity of the personnel responsible for sampling, performing each test or calibration, and checking the results.

8.81. Technical records are accumulations of data and information that result from carrying out tests or calibrations and which indicate whether specified values for quality parameters or process parameters were achieved. They may include forms, contracts, worksheets, workbooks, checklists, work notes, control graphs, external and internal test reports and calibration certificates, clients' notes, papers and information from client feedback. Observations, data and calculations should

be recorded at the time that they are made and it should be possible to link them to the specific task concerned.

8.82. Any mistake that occurs in a record should be crossed out (not erased, made illegible or deleted), and the correct value should be entered alongside it. All such alterations to records should be signed or initialled by the person making the correction. In the case of records stored electronically, equivalent measures should be taken to avoid the loss of, or changes to, the original data.

Internal audit

8.83. When audit findings cast doubt on the effectiveness of the operations or on the correctness or validity of the laboratory's test or calibration results, the laboratory should take timely corrective action, and it should notify clients in writing if investigations show that the laboratory results might have been affected.

Laboratory facility infrastructure

8.84. The management should provide adequate laboratory facilities to perform all processes under consistent and familiar conditions. The management should ensure that:

(a) Technical standards and requirements are fulfilled (facilities, computers and programs).
(b) Adequate technical documentation is available (handbooks, tables and manuals).
(c) Necessary environmental conditions (which may influence results) are well known, correctly maintained, documented, monitored and recorded (thresholds and the assignment of responsibility for stopping a task should be specified).
(d) Access to the facilities is restricted and monitored.
(e) Procedures for good housekeeping have been specified and documented.
(f) Work in one room should not disturb the process in an adjoining room.

Test and calibration methods and validation of methods

8.85. Each measurement method should be well documented in a procedure that describes the task step by step, if this is deemed necessary. The management should ensure that staff members are using an up to date method and that they carry out their daily work guided by these documented methods. The selected method should be well known (in terms of its accuracy, correctness, repeatability,

reproducibility and robustness), and the range of uncertainties in measurements should be known and should be shown on the measurement report. Each method of measurement should be validated in accordance with the laboratory's procedure for validation.

8.86. Consideration should be given, as appropriate, to the following:

(a) Methods should be planned methodically and documented in a form suited to the working style of the laboratory.
(b) The documentation should describe the method of measurement on a step by step basis, as appropriate, and should include guidance on how to keep the necessary records.
(c) As a first method of validation, the newly developed method of measurement should be tested using different parameters, and the results should be documented and assessed.
(d) An additional step of validation providing a 'go/no go' decision could be incorporated into the method.
(e) The actions to be taken when a deviation (error) occurs (i.e. who has to do what and when) should be determined.
(f) The data flow of measurement results (who needs what information, when and in which form, and how the backup of data can be ensured) should be organized.

Test and calibration equipment

8.87. The laboratory should possess adequate equipment to perform the necessary services for the customer, including sampling, sample preparation, measurement or calibration, calculations and reporting. The equipment necessary to produce the results of measurements should be functional and should be capable of being used for day to day measurements.

8.88. The following activities should be undertaken:

(a) Periodic and documented calibrations should be performed to guarantee correct results of measurements.
(b) Periodic and documented functional tests should be performed between the calibration times to test the correct functioning of the equipment.
(c) All maintenance work provided for by the equipment manufacturer should be done and should be documented in an equipment file.
(d) Training and periodic retraining of every equipment operator should be completed to ensure that staff members are familiar with the equipment.

8.89. All equipment and self-designed software should be clearly identified. This may be accomplished through documentation that is sufficient to enable the validation of software and the proper setting up of equipment.

8.90. Checks on outgoing and incoming equipment should be performed if a piece of equipment is used outside the laboratory.

8.91. All calculations, including those performed using commercial software (e.g. for spreadsheets), should be documented and validated.

Measurement traceability

8.92. To be sure that the measurement results are traceable, each measurement device that has an influence on the results should be calibrated before being put into service and at defined intervals thereafter. The standards used for these calibrations should be quantified in terms of the International System of Units (SI). In some cases — for example in connection with measurements relating to ^{222}Rn — participation in suitable international intercomparison exercises is also recommended for demonstrating confidence in the measurements.

8.93. Calibration services should ensure traceability of their standards and measuring instruments by means of an unbroken chain of calibrations or comparisons linking them to relevant primary standards, with consistent use of the International System of Units (SI) throughout this chain. For measurement services, this traceability can be achieved by using a calibration service.

8.94. To keep a calibration service or measurement service operational, it may be helpful to do the following:

(a) Organize information on all calibration standards used into a database file, giving:
  (i) Calibration data;
  (ii) Serial numbers of units calibrated;
  (iii) Date of last and next calibrations;
  (iv) Location and name of the tester.
(b) Store all calibration procedures and their outcomes, the calibration certificates, in the laboratory.
(c) Support periodic calibration with a time schedule programme.
(d) Keep calibrated spare parts available for important devices to shorten the downtime in case of a malfunction.

Sampling

8.95. If a testing laboratory also performs sampling, it should do so in accordance with accepted standards or documented procedures. If a subcontractor or a customer performs sampling, it should be ensured that the same restrictions and conditions apply as for the laboratory.

8.96. Consideration should be given, as appropriate, to the following points in implementing a procedure for sampling:

(a) The requirements of relevant standards and those of customers (e.g. in relation to the sampling location, sampling time, name of the person responsible for sampling and technical conditions) should be addressed.
(b) Any possible negative influence on the samples during sampling, transport of the samples, handling, storage and analysis should be avoided.
(c) Procedures should be well documented, and they should, as appropriate, use statistical methods as a basis for providing properly identified samples and sample data for the measurement process.
(d) Information should be provided to the customer if the sampling process reveals problems or errors, or in the event that the sampling was performed incorrectly.

Handling of items for testing and calibration

8.97. Test items and calibration items should be handled with care. To maintain their identity, the item and its description should never be separated. The laboratory should have a procedure in place that provides:

(a) Identification and labelling of incoming test items and calibration items;
(b) Reporting of any abnormalities found for the items handled;
(c) Instructions for handling, storage and transport, and on the necessary environmental conditions to be maintained for the test items and calibration items;
(d) Instructions on the return of the items to the customer or on approved disposal routes for the items.

Ensuring the quality of test results and calibration results

8.98. The laboratory should have a process and procedure in place to ensure continuous control of the quality of the services provided to the customer.

8.99. When designing such a process and procedure, consideration should be given, as appropriate, to:

(a) Using only certified (reference) materials for calibration purposes and internal quality control;
(b) Carrying out all measurements and calibrations in accordance with the applicable documentation;
(c) Participating in interlaboratory comparison exercises or proficiency testing programmes;
(d) Replicating tests or calibrations using the same or different methods;
(e) Retesting or recalibration of retained items;
(f) Correlation of results for different characteristics of an item;
(g) Using statistical methods, such as control charts, to determine the quality of calibration results over a longer time period so as to identify possible trends in the degradation of instruments.

Reporting of results

8.100. Results should be reported to the customer accurately and in a comprehensible way so as to meet the customers' needs and fulfil any requirements of the regulatory body.

8.101. The laboratory should devise a layout for its reports in which recognition is given to the:

(a) Requirements of the relevant regulatory bodies;
(b) Requirements of the relevant standards;
(c) Internal rules for reporting within the organization.

Care should be taken to clearly designate data coming from a subcontractor. The laboratory should have a procedure in place for changing reports in the event that errors are detected in the original version. All reports issued should be considered to be records and should be treated accordingly.

9. ENGINEERED CONTROLS, ADMINISTRATIVE CONTROLS AND PERSONAL PROTECTIVE EQUIPMENT

GENERAL CONSIDERATIONS

9.1. Where the physical design features of a facility do not provide sufficient containment or shielding of radioactive material, additional engineered controls using facility systems and components should be used to protect individuals. For example, adequately designed and properly controlled ventilation systems are an effective means of minimizing exposure in workplaces prone to airborne contamination, such as in underground mines and in buildings in which dry processing of radioactive minerals is carried out. Installed fume hoods, glove boxes and manipulators are also examples of engineered controls.

9.2. Appropriate monitoring should be performed to determine the adequacy and effectiveness of engineered controls. For instance, when engineered controls, such as ventilation, vacuum cleaners or containment devices, are used to reduce or to maintain activity concentrations of radionuclides in the work environment, air quality should be monitored. Generally, for installed physical design features, such as fume hoods, fixed location air sampling is preferred, whereas for temporary controls, such as portable ventilation or the use of vacuum cleaners, grab sampling is preferred. Real time air monitoring for determining the adequacy of installed controls may also be appropriate and should be made a requirement for some situations.

9.3. Temporary engineered controls, such as temporary shielding, containment devices and portable or auxiliary ventilation, may need to be used during non-routine operations such as maintenance, modifications, and decontamination and decommissioning. Planning for non-routine operations should include an evaluation of the potential for the spread of contamination and an evaluation of the effectiveness of the engineered controls in reducing such potential.

9.4. Temporary containment devices may be particularly useful in controlling the spread of contamination when leakages occur in the containment system or when maintenance work requires the containment system to be opened. These devices range in complexity from simple plastic catch basins suspended below leakage points to complex portable buildings used to enclose an entire work area. Many commercially available designs include provisions for glove ports and equipment ports, ventilation, and exit portals for reduction of contamination.

9.5. The exhausts from portable air handling systems used in contaminated areas, including vacuum cleaners, should be equipped either with high efficiency particulate air (HEPA) filters or with suitable adsorbers, as appropriate, and should be directed to installed systems that are so equipped. These provisions may not be necessary in areas where only tritium or radioactive noble gases are present, or when the material to be vacuumed is wet enough to preclude resuspension after entry into the system collection chamber. Improper use of vacuum cleaners and portable air handling equipment may result in the generation of airborne radioactive material or removable surface contamination. The prolonged use of air handling equipment may result in a significant buildup of radioactive material in ducts and filters. A radiological assessment of the operation of such equipment should be performed periodically by monitoring the exhaust air and accessible equipment surfaces.

9.6. When the use of physical design features, including specific engineered controls to limit individual exposures, is impractical or inadequate, administrative controls should be considered to ensure that protection and safety is optimized. Examples of administrative controls include the use of work authorizations and restrictions or controls on access to areas with the potential for contamination.

9.7. Control measures, such as quality in design, installation, maintenance and operation, together with administrative arrangements and instruction of personnel, should be used to the maximum extent possible before relying on personal protective equipment for ensuring the protection of workers. In circumstances in which engineered controls and administrative controls are not sufficient to provide adequate levels of protection of workers, personal protective equipment should be provided to restrict the exposures of workers.

SHIELDING

9.8. The provision of shielding can be an effective form of engineered control. At the design stage, an adequate thickness of the shielding material should be provided to give an acceptable level of protection to the workers in normal operation as well as in abnormal situations. The design of shielding should be such as to ensure that the individual external dose in normal working conditions is lower than the dose constraint. The adequacy of the shielding in abnormal conditions, including accident situations leading to maximum foreseeable (worst case) radiological consequences, should be evaluated and, where necessary, additional shielding or other engineered controls (e.g. interlocks) should be considered. The likelihood of an accident or other incident giving rise to an

unacceptable level of individual dose should be maintained at a very low level and any planned exposure situation that could cause the annual dose limit to be exceeded because of inadequate shielding should be prevented. The effectiveness of the shielding should be actively monitored by means of installed workplace monitoring instruments and/or by regular area surveys performed by suitably qualified personnel. Additional local shielding should be provided to reduce the radiation field as needed. Passive area monitors should also be used to determine doses integrated over time in various areas. The results should be analysed for trends, and the shielding should be improved, as appropriate.

9.9. Shielding should be considered in work involving X rays, gamma radiation, and neutrons and other high energy particles (which can include high energy beta particles). Appropriate shielding materials should be selected depending on the type of facility. In accelerator facilities, for example, shielding for the accelerators and the storage ring should be provided through a combination of various materials, as appropriate (e.g. concrete, lead, polyethylene and soil), and should be designed for normal operations by using conservative assumptions about beam loss to limit the maximum dose received by a worker. Additional guidance relating to the design and installation of structural shielding of gamma, electron and X ray irradiation facilities is provided in IAEA Safety Standards Series No. SSG-8, Radiation Safety of Gamma, Electron and X Ray Irradiation Facilities [158].

VENTILATION

9.10. The purpose of the primary ventilation system in a facility is to provide fresh air to workplaces to remove airborne contaminants generated by the operations. Careful attention should be given to the design of the ventilation network, including the calculation and verification of rates and velocities of air flow, to ensure that it is adequate for controlling airborne contamination. In many facilities, the control of airborne contamination is achieved by:

(a) Maintaining adequate negative pressure with respect to atmospheric pressure.
(b) Providing an adequate or prescribed number of air changes in the workplace.
(c) Providing the appropriate exhaust air, off gas cleaning systems (including scrubbers, adsorbers and/or HEPA filtration) so that the discharges from the facility will be within authorized limits. The discharge of the exhaust air should be through a stack of appropriate height to provide the necessary dilution for the releases to protect members of the public.

9.11. Ventilation is of crucial importance in underground mines, where workers can be exposed at elevated levels due to radon and airborne dust containing radionuclides of natural origin. The design of mine ventilation systems is complex and the measurement and analysis of air flows requires special skills. It is usual, therefore, for such mines to employ an appropriately qualified ventilation officer who reports directly to senior levels of the mine management.

9.12. The ventilation officer in a mine should have the responsibilities specified in para. 3.177.

9.13. In some workplaces, especially in underground mines and in buildings where dry processing of radioactive minerals is carried out, the fresh air supplied by the primary ventilation system might not be adequate to ventilate particular parts of the workplace. Examples of such workplaces include development ends in underground mine tunnelling operations and product bagging areas in facilities processing radioactive minerals. In these circumstances, auxiliary ventilation is commonly supplied to the affected parts of the workplace through flexible ducts. The positioning of auxiliary ventilation ducts should be such as to avoid recirculating eddies of contaminated air.

9.14. The proper functioning of the primary and auxiliary ventilation systems throughout the operating phase of the facility should be ensured and, if necessary, should be indicated as audiovisual alarms in the control room display panel or the radiation protection officer's display panel, so that prompt action for the protection of workers can be initiated. The employer should put in place a programme of inspection and maintenance of ventilation equipment, including main fans, auxiliary fans and any heating or cooling systems. This programme should be documented and recorded.

9.15. In underground mines, the design of the ventilation system should be an integral part of the planning and development process for the mine with the objective of achieving, where practicable, a 'one pass' or parallel ventilation system to ensure good air quality and to minimize the buildup of radon and airborne dust.

9.16. For the effective operation of primary and auxiliary ventilation systems in a facility:

(a) Air intakes and exhausts should be separated to the extent practicable.
(b) Ventilation should be considered an important safety related system. For equipment such as fans, blowers and HEPA filter systems consideration

should be given to the provision of standby systems, including alternative power supplies (such as diesel generators), where necessary. In this way, process systems can be shut down safely during the conduct of maintenance activities while all monitoring systems will continue to work. Consideration should also be given to real time indicators of system performance to alert operators to failures or malfunctions of exhaust systems.

(c) For the protection and safety of workers, every workplace should be supplied with air of a quantity and quality sufficient to ensure that exposure due to airborne contaminants, such as dust, radon and other radioactive gases, is minimized.

(d) Air velocities should be high enough to dilute the airborne contaminants but not so high as to cause settled dust to be resuspended.

(e) In the case of underground mines, the primary systems for ventilation and dust control should preferably be operated continuously; if the continuous operation of these systems is not practicable, the regulatory body may authorize intermittent operation subject to (f) below.

(f) When the ventilation system has been changed, has failed or has been shut down, workers should be allowed to return to their workplaces only after the ventilation system has been restarted and appropriate monitoring has been performed to ensure that the concentrations of airborne contaminants have been reduced to acceptable levels.

9.17. The employer should take measures to deter unauthorized entry to any underground area within a mine that is not ventilated. In the event that the ventilation system is not in operation, essential maintenance services necessary to ensure the operation of equipment or machinery may be carried out, and all practicable measures should be taken to limit the doses received by the workers engaged in the maintenance operation.

9.18. In some situations, such as in an underground mine or in a building where the dry processing of radioactive minerals is carried out, the local operating instructions should specify the actions to be taken in the event of a failure of the ventilation system.

9.19. The location of fixed workstations in return airways or in areas of high external radiation should be avoided. Where appropriate, operator booths with a filtered air supply should be used in these circumstances to provide the necessary protection.

DUST CONTROL

9.20. In most operations involving the potential for high dust generation, for example mining and mineral processing, the adoption of dust control measures is usually a legal requirement because of the necessity of protecting workers against non-radiological hazards such as inhalation of silica particles. These measures generally provide sufficient restrictions on airborne dust concentrations to ensure adequate protection of workers against the inhalation of any radionuclides of natural origin that may be present in the dust.

9.21. To ensure that adequate methods for the control of dust are in place in underground mines and in buildings where the dry processing of radioactive minerals is carried out, programmes for the air sampling and control of airborne dust should be formalized. The following measures should be taken:

(a) The generation of dust in operations should be reduced to the extent practicable by the use of appropriate techniques for mining and mineral processing, such as the use of proper blasting patterns and timing, the use of water and other means of suppressing dust, and the use of appropriate equipment.
(b) Where dust is generated, it should be suppressed at source. Where necessary and practicable, the source should be enclosed under negative air pressure. Air might have to be filtered before being discharged to the environment.
(c) Dust that has not been suppressed at source may be diluted to acceptable levels by means of frequent changes of air in the working area. Again, the exhaust air might have to be filtered before being discharged to the environment.
(d) Care should be taken to avoid the resuspension of dust as a result of high air velocities.
(e) Where methods of dust control do not achieve acceptable air quality in working areas, enclosed operating booths with filtered air supplies should be provided for the workers.

SPILLAGE OF RADIOACTIVE MATERIAL

9.22. The employer should establish standard operating procedures to be followed in the event of any significant radiation hazard or potential radiation hazard arising from the spillage of radioactive material from a facility or during transport between facilities. Such standard operating procedures should include procedures for the following:

(a) Cleaning up spillages;
(b) Restricting access to the area;
(c) Implementing contingency plans;
(d) Monitoring affected persons;
(e) Obtaining advice from the radiation protection officer or qualified expert;
(f) Managing waste arisings;
(g) Notifying the regulatory body or other relevant authorities as required.

9.23. Any spillage of radioactive material should be cleaned up as soon as practicable in order to minimize the spread of contamination. The area should be decontaminated by the removal of all loose contamination and contaminated materials to the extent practicable.

SURFACE CONTAMINATION

Programme for contamination control

9.24. Work with unsealed radioactive substances creates the potential for contamination of surfaces. A programme for contamination control should be implemented to identify the presence of surface contamination and to prevent the inadvertent transfer of such contamination at levels exceeding specified values under normal operating conditions. A programme for contamination control that makes use of physical design features and that includes additional engineered controls and administrative controls, as appropriate, should be an essential element of a comprehensive radiation protection programme aimed at ensuring that the protection and safety of workers is optimized.

9.25. In implementing a programme for contamination control, physical design features for controlling surface contamination at source should be considered the most important element. The physical design features used in a programme for contamination control may include:

(a) Specific design features aimed at confining radioactive material to prevent it from causing surface contamination;
(b) Ventilation systems aimed at preventing the buildup of surface contamination as a result of the settling of airborne particles.

9.26. Design features such as those mentioned in para. 9.25 may be the primary methods of controlling workers' internal exposures from inhalation of radionuclides in airborne particles, especially for non-routine work such as

equipment maintenance. This is the case irrespective of whether the particles are released to the air directly from the source of dust or are resuspended into the air from contaminated surfaces. The use of such physical design features is illustrated by the following two examples:

(a) A permanently installed ventilation system with HEPA filtration or appropriate adsorbers can be included as a physical design feature to control concentrations of airborne radionuclides during routine operations. A temporary ventilation system, also using HEPA filtration or adsorbers, may be used as an engineered control during certain maintenance activities.
(b) An appropriately designed drainage system should be made available as a physical design feature to transfer contaminated liquid waste to a controlled collection point (hold-up tanks). Temporary drains may be installed as engineered controls to collect the effluent while the line is opened for maintenance, under a special work permit system where necessary. In the case of liquid waste containing fissile material, additional special measures may be required.

9.27. When the use of physical design features (including specific engineered controls) to restrict individual exposures is impractical or not sufficiently effective, administrative controls should be implemented. Such administrative controls might include restrictions on access to a contaminated area or the use of specific work practices designed to minimize the transfer of contamination.

9.28. Work in contaminated areas should be conducted in a manner that minimizes the spread of contamination to adjacent surfaces, to individuals in the area and to the workplace atmosphere. To control the spread of contamination and to restrict individual exposures, provisions, such as the erection of physical barriers (with changing of footwear) and cordoning off affected areas, should be made in and around contaminated areas.

9.29. Control of access to contaminated areas may be necessary to ensure that workers entering the area are informed of the radiological status and potential hazards and, if necessary, are provided with the appropriate personal protective equipment. Visual display of the levels of contamination and caution boards should be prominently displayed. The workers' exit from contaminated areas should be controlled to ensure that radioactive substances are not inadvertently transferred from the area by personnel or on equipment. Efforts should be made to control the degree of contamination and the size and number of contaminated areas within a facility.

9.30. Exits from contaminated areas should include provisions to facilitate the retention of contamination in the area and for monitoring individuals and the area to ensure that control has been maintained. Individuals exiting contaminated areas should be monitored, as appropriate, for the presence of surface contamination. As a minimum, individuals exiting contaminated areas should perform a check for surface contamination, using either portable monitoring devices or automated monitoring devices, as appropriate. Where the only contaminated areas are laboratory bench surfaces or fume hoods, or where the potential for contamination is limited to specific parts of the body, the checking should be concentrated on the areas most likely to be affected. If background radiation levels or other local conditions at the exit point preclude the detection of surface contamination, the exit point should be moved to an alternative location, for instance to an area with lower background radiation levels. If relocation of the exit point is not practicable, individuals should proceed directly from the exit point to an appropriate area to perform the necessary checks. On their removal from contaminated areas, all objects, including tools, materials, equipment and personal items, should be monitored by competent personnel. Workers should be made aware of the necessity for such monitoring.

9.31. Skin contamination by certain radioisotopes, such as tritium, cannot be reliably detected by the hand-held or automated monitoring instruments that are currently available; therefore, individual checking is not an appropriate means of detecting such skin contamination. Where individual exposure due to such contamination hazards is possible, additional emphasis should be placed on bioassay programmes and routine programmes for contamination monitoring and air monitoring.

9.32. Protective clothing should be worn in contaminated areas where levels of removable contamination exceed specified values. The type of protective clothing required should be determined on the basis of considerations of contamination levels, the chemical and physical form of the contaminant, the activities to be performed and the accessibility of the area. Consideration should also be given to other, non-radiological hazards such as heat, flames, hazardous chemicals, physical obstructions, electrical shock and limited visibility.

9.33. The control measures discussed here have proven to be effective in minimizing the generation and spread of removable contamination. However, these measures might not be appropriate in areas having only fixed contamination. When surfaces with fixed contamination are located within a controlled or supervised area, the requirements for area classification and entry control should be such as to provide for adequate control of entry and exit. Additional control

measures may be necessary to prevent inadvertent or unauthorized removal of the fixed contamination by methods that disturb the surface. Although fixative coatings may be used to bind the contamination to the surface, such usage should be minimized, and as much of the contamination as practicable should be removed prior to application of the coating.

Monitoring for surface contamination

9.34. A contamination monitoring programme should be carried out as part of the prior radiological evaluation and ongoing safety assessments, and to verify the effectiveness of the measures for preventing and controlling surface contamination.

9.35. The instruments and techniques used for contamination monitoring should be appropriate for the types, levels and energies of the radiation encountered. Instruments should be regularly maintained and calibrated for the prevailing environmental conditions and should be routinely tested for operability. A suitable surface contamination meter should be available wherever unsealed radioactive substances, such as liquids and powders, are in use. Care should be taken to avoid the instrument coming into contact with potentially contaminated surfaces. Instruments that comprise a rate meter and probe provide versatility both in the range of detectable radionuclides (using different probes) and the ease with which readings can be taken. Surfaces that should be routinely monitored for spillages or contamination include the body, protective clothing, working areas (such as benches and floors), equipment and packages used for the transport of radioactive material.

9.36. Particular care is needed when making surface contamination measurements on items contaminated with naturally occurring radioactive material. For some items, alpha monitoring equipment may be completely unsuitable, even though alpha emitters are usually the radionuclides of greatest concern. The alpha self-absorption in the contaminant layer is usually too high for a reliable measurement to be obtained. The alpha probe of the instrument should be held within 5 mm of the surface. This might be impossible when measuring rough or uneven surfaces. Furthermore, the vulnerability of the surface of the alpha probe could result in it becoming damaged in attempts to measure rough or uneven surfaces. Because of these difficulties, beta monitoring is generally the preferred method for measuring items contaminated with naturally occurring radioactive material. Even with the more highly penetrating beta radiation, however, self-absorption should still be taken into account as necessary. Most beta detectors are sensitive to gamma radiation. If this is not adequately taken

into account, the presence of ambient gamma radiation might be misinterpreted as contamination. Since the radionuclides in the contaminating layer may be out of equilibrium, measurement of beta emissions might not provide sufficient information on the activity concentrations of alpha emitting radionuclides. It may therefore be necessary to establish, in advance, the radionuclide composition of the contaminating layer by detailed analysis in a laboratory.

9.37. Even quite low levels of surface contamination can give rise to a risk of internal exposure. Instruments for monitoring surface contamination have detection efficiencies in the range of 0–30% for different radionuclides. Measurements should be made by using a calibrated instrument with the best available predetermined detection efficiency for the radionuclides of concern. The measurements, in counts per second, should be converted to units of becquerels per square centimetre. Some surface contamination meters are programmable: the user sets the instrument's likely response to the radionuclide in use and obtains a direct measurement of surface contamination (in becquerels per square centimetre).

9.38. Each surface contamination meter is designed and type tested to measure a specific range of contaminants. Its response to contamination will depend on:

(a) The type and energy of the radiation emitted by the radionuclides present in the contamination;
(b) The instrument's intrinsic detection efficiency for each radionuclide, which is determined by the detector's characteristics, the thickness of the window and the dimensions of any protective grille;
(c) The detection geometry, including the detector's dimensions, the extent of the contamination, the nature of the contaminated surface and the detector-to-surface distance;
(d) Inherent electrical noise, ageing or fault conditions in the instrument's components.

9.39. When selecting equipment for monitoring surface contamination, it should be noted that for contamination uniformly distributed across a surface (i.e. as opposed to a single small spot of contamination), the response of the instrument increases with the surface area of the probe. This is illustrated in Table 6 for four types of surface contamination monitor. The management should consult the radiation protection officer or a qualified expert, as appropriate, for advice on the selection of the monitoring equipment.

TABLE 6. SURFACE CONTAMINATION MONITORS: VARIATION OF SENSITIVITY WITH PROBE SURFACE AREA

Type of surface contamination monitor	Surface area of probe (cm^2)	Calibration factor (^{60}Co source) (Bq/cm^2 per count/s)
Geiger–Müller end window	7	10.2
Zinc sulphide + plastic scintillator	50	0.5
Plastic scintillator	65	0.1
Xenon counter	260	0.03

9.40. Specially designed instruments for monitoring surface contamination may be needed in facilities in which surface contamination by naturally occurring radioactive material is generated. In the oil and gas industry, for instance, the risk of fire or explosion might necessitate the use of intrinsically safe instrumentation. In addition, the widespread presence of surface contamination on the inside of pipes will necessitate the use of special cylindrical form beta detectors (see para. 9.36). For monitoring for surface contamination by naturally occurring radioactive material, the monitoring instruments and measurement systems should ideally be calibrated by using natural uranium and natural thorium standard sources, as appropriate.

9.41. Each monitoring instrument should be tested before first use, at regular intervals (typically annually) and after any repair that may have affected the instrument's performance. These tests should be conducted by qualified experts using calibrated, uniformly contaminated plaques with an active area of dimensions similar to those of the detector. The radionuclide used should emit radiation similar to the radiation emitted by the potential contaminant. The objectives are:

(a) To determine the operating voltage for each detector, especially interchangeable probes; other electrical and mechanical features can also be tested.
(b) To determine or to confirm the detection efficiency of the instrument for each relevant radionuclide.

By using the detection efficiency, a calibrated response can then be provided to the user to convert the reading (in counts per second) to a surface activity concentration (in becquerels per square centimetre). The linearity of the response and any inter-range differences can also be investigated. The user of the

instrument should keep a certificate relating to the most recent formal test and should carry out routine checks on the instrument. Sources for these purposes are available and are sometimes attached to the contamination detector cover. The condition of the battery should be checked each time the instrument is used.

Personal hygiene and first aid

9.42. To prevent inadvertent intakes by workers, the employer should provide washing facilities for all workers that are convenient to the place of work, and should allow sufficient time to each worker for the use of the washing facilities before rest breaks and meal breaks, and at the end of the shift. The employer should provide — at locations that are outside of contaminated working areas and that are reasonably accessible to every worker — clean eating areas that are supplied with water, good quality air, hand washing facilities and toilet facilities. These facilities should be designed, monitored and maintained in a manner that is acceptable to the regulatory body. Workers using these facilities should be instructed on how to prevent the spread of contamination.

9.43. No person should be permitted to eat, drink, chew gum or tobacco, smoke, take snuff or apply cosmetics in working areas in which radioactive substances could be ingested.

9.44. Special precautions should be taken in the cleaning of wounds sustained in areas where radioactive contamination is present or wounds involving contaminated equipment. Advice from a medical officer should be sought in such cases (see also para. 9.52).

9.45. Before workers enter working areas in which contamination might be present, any cuts and wounds, in particular wounds to the hands, should be properly dressed with waterproof dressings.

9.46. The employer should ensure that workers are provided with first aid training that is specific to the job.

DECONTAMINATION OF EQUIPMENT AND DECONTAMINATION OF PERSONNEL

Decontamination of equipment and areas of floors and walls

9.47. The employer should provide, as necessary, a facility and decontamination agents for the decontamination of contaminated equipment and tools used for maintenance work, and should provide means for cleaning contaminated areas of floors and walls. In general, water is the preferred decontamination agent. Other cleaning agents should be selected on the basis of: their effectiveness; their hazardous properties; the amount of waste generated; their compatibility with the contaminated surface and with other systems or items that might be contaminated (including protective clothing and waste handling systems); and ease of disposal (see Refs [159, 160] for additional information). The effectiveness of decontamination measures should be periodically reviewed and target levels should be identified in local operating procedures.

Decontamination of personnel

9.48. Personal contamination includes the contamination of personal clothing, skin, hair, eyes, mucous membranes and wounds. In this context, personal clothing includes work clothing provided by the employer, but does not include protective clothing provided solely for the purposes of contamination control.

9.49. When contamination is detected, the radiation protection officer should be informed. The radiation protection officer should ensure adequate characterization of the potential for significant doses by assessing the extent of the contamination, and by retaining samples of the contamination, as necessary, to perform a detailed dose assessment and to initiate decontamination procedures. The levels of contamination that would trigger the need for dose assessments and methods of decontamination should be established for site specific radionuclides.

9.50. Intrusive methods of decontamination, such as removal of tissue, require medical assistance. In the event of contamination of the skin by contaminants such as radioactive iodine, decontamination by washing or by using detergent might not be effective; in the event of serious contamination, medical advice should be sought immediately.

9.51. Contaminated personal clothing should be decontaminated by laundering or other appropriate methods, and then monitored and returned to the owner or, if necessary, disposed of as radioactive waste.

Wounds

9.52. Medical treatment of injuries takes precedence over radiological considerations. Emergency medical care should be administered immediately for wounds involving radionuclides (see para. 10.4(d)). However, decontamination efforts should also be started immediately to prevent uptake of soluble radionuclides into the blood.

PERSONAL PROTECTIVE EQUIPMENT

9.53. Personal protective equipment should be selected with due consideration of the hazards involved. The equipment should not only provide adequate protection but also be convenient and comfortable to use. Consideration should also be given to the possibility of an increase in exposure caused by the additional constraints of the personal protective equipment.

9.54. Examples of personal protective equipment include reinforced clothing, ventilated suits, protective glasses and respiratory protective equipment. Workers who may have to use such equipment should be properly trained in its use, operation, maintenance and limitations. It should be ensured that personal protective equipment correctly fits the wearer.

Respiratory protective equipment

9.55. Employers should not rely on the use of respiratory protective equipment to comply with the dose limits for workers, except in temporary and unforeseen circumstances. Respiratory equipment may, nevertheless, be necessary in emergencies, for repair and maintenance activities, and in special short term circumstances. Respiratory protective equipment should be used for a specified and limited period of time only.

9.56. If levels of airborne contaminants exceed the safe working levels (e.g. derived air concentration) specified by the management of the facility, appropriate respiratory protective equipment should be worn by those persons undertaking actions under those circumstances. While corrective measures are being undertaken, the area should be monitored to estimate possible exposures. Employers should withdraw workers from affected areas if continued exposures are such that the recommended safe working levels, derived air concentration values or dose limits are likely to be exceeded.

9.57. Respiratory protective equipment and its use should be in conformance with the following:

(a) The use of respiratory protective equipment should be carefully supervised to ensure that the expected protection is provided.
(b) The management should ensure that respiratory protective equipment fits properly and is used properly.
(c) The protection factors to be used in assessing the actual intake of the worker should be specified.
(d) The periods of use of respiratory protective equipment should not be so long as to discourage its proper use.
(e) Filter respirators should have a low breathing resistance and should be efficient for the dust size concerned.
(f) When equipment for supplied air is used, the air supplied should be of respirable quality and of sufficient quantity to ensure leak free operation in the conditions of use.
(g) Powered air respirators or helmets with face shields should be preferred to other types of respiratory protective equipment for the comfort of the workers using them, provided that they ensure effective respiratory protection.
(h) In choosing equipment for a particular operation, factors affecting the comfort of workers (e.g. the weight of the equipment, its restriction of vision, and effects on temperature and mobility) should be considered as well as the required protection factor.
(i) Respiratory protective equipment should be cleaned and maintained regularly, and should be inspected at appropriate intervals by properly trained persons in suitably equipped facilities.
(j) Respiratory protective equipment should be examined, fitted and tested, as appropriate, by a competent person before being issued for use and periodically during use; and the results of these examinations and tests and details of any repairs should be recorded.
(k) The frequency of testing of respiratory protective equipment should be determined on the basis of the type of equipment, the environment in which it is used and how it is handled.
(l) Respiratory protective equipment should be checked by users before use and by the safety maintenance staff after cleaning, and should be pressure tested, as appropriate.

Other personal protective equipment

Protective clothing

9.58. Where there is a potential for contamination, the employer should specify appropriate requirements for protective clothing on the basis of the level of risk. The employer should provide the necessary overalls, head coverings, gloves, boiler suits and impermeable footwear and aprons (including lead shielding aprons, where appropriate) in accordance with the risks of external exposure and internal exposure, and, as appropriate, for the working conditions. Work clothes, including gloves and footwear, should be provided to every worker whose personal clothing is likely to become contaminated during the course of work.

9.59. The employer should also specify cases in which individuals are required to shower and change clothes on leaving contaminated workplaces, and should provide suitable storage facilities for clothing and washing facilities.

9.60. Individuals should wear the specified protective clothing. In some cases, it may be appropriate for personal clothing and work clothing to be removed before protective clothing is donned. Personal clothing and work clothing should be changed in separate locker rooms, with a washroom in between, where appropriate, to control the spread of contamination.

9.61. Where contaminated work clothes are stored, laundered or otherwise decontaminated, or disposed of, the employer should put in place measures to prevent the spread of contamination to other persons or workplaces, and to minimize the exposures of individuals and the release of contaminants to the environment. The employer should provide suitable laundry facilities, boot washes, vacuum systems or other means of decontamination, as necessary.

Protective glasses

9.62. Where engineered controls and administrative controls are not sufficient to ensure that protection for the lens of the eye is optimized, consideration should be given to protecting the lens of the eye by means of workers' use of appropriate protective glasses. Glasses made of Perspex may be sufficient when the exposure is predominantly to beta radiation. Account should be taken, however, of any bremsstrahlung generated by high energy beta radiation. When the exposure is predominantly to penetrating radiation (gamma radiation or X rays), consideration should be given to the use of protective glasses with lenses containing lead.

9.63. If conventional industrial safety glasses are to be used to protect against exposure to beta radiation, they should be evaluated for their shielding properties beforehand. Similarly, protective glasses with lenses containing lead should also be evaluated before use. Such glasses may well be adequate for protecting against low energy X rays, but may be inadequate for protecting against higher energy gamma radiation.

9.64. The radiation attenuation factor of the eyeglass lenses is not an adequate descriptor, in itself, of the effectiveness of the eyewear in reducing radiation exposure [161]. The area covered by the lenses should also be considered. Well fitting glasses containing a small percentage of lead (including side shields) should be adequate to give the required protection to the lens of the eye [161]. For maximum effectiveness, protective glasses should intercept as much of any scattered radiation as possible, in particular in image guided interventional procedures. Workers should use such protective glasses in workplaces with a higher potential for exposure of the lens of the eye.

JOB ROTATION

9.65. In workplaces where there are areas with a potential for high levels of radiation exposure, when no other practicable means of control are available, job rotation can be considered as an administrative control to restrict the exposure of individual workers. The use of job rotation should be kept to a minimum, however, and job rotation should never be used as a substitute for the development and use of appropriate methods of control of individual exposure.

CONSIDERATIONS FOR MINERAL PROCESSING OPERATIONS INVOLVING NATURALLY OCCURRING RADIOACTIVE MATERIAL

9.66. Some mineral processing operations involve the presence of naturally occurring radioactive material, either in the form of the mineral itself or in the form of a residue, product or by-product (see para. 3.162). The first consideration in the design of the facilities concerned should be the containment of naturally occurring radioactive material. For instance, the design and operation of crushing and screening plants should be such as to keep the release of contaminants as low as practicable. The design of the concentrator should be such as to minimize the generation of airborne or liquid contaminants.

9.67. It should be recognized that complete containment of process material in such facilities is often impractical. Any naturally occurring radioactive material that cannot be contained effectively within the process and becomes airborne should be controlled by means of an adequate ventilation system to remove airborne contaminants and to minimize occupational exposure (see paras 9.10–9.19).

9.68. In the design of processing plants involving naturally occurring radioactive material, aspects that prevent the buildup of contamination should be considered. The design should facilitate maintenance work for the removal of any contaminants that do accumulate.

9.69. During maintenance operations, special care should be taken to control occupational exposure arising from the accumulation of naturally occurring radioactive material in pipes and vessels in the plant owing to the formation of sediments and the buildup of scale.

9.70. As far as practicable, all hazardous material should be handled with automated equipment in enclosures where negative air pressure is maintained, regardless of whether the hazard is due to radionuclides of natural origin in high concentrations or to chemically toxic constituents.

9.71. To help facilitate cleanliness, the paint colours used for walls, handrails, equipment, furniture and other objects should be different from the colours of the materials and products being processed.

9.72. Solid, liquid and gaseous residues from the processing operation should be managed in accordance with procedures approved by the regulatory body for the protection of workers and the public, and protection of the environment.

10. WORKERS' HEALTH SURVEILLANCE

RESPONSIBILITIES

Management

10.1. In terms of paras 3.76(f), 3.108 and 3.109 of GSR Part 3 [2], the management should ensure that all workers engaged in activities in which they

could be subject to occupational exposure are provided with the necessary workers' health surveillance and health services. For itinerant workers who are subject to exposure due to a source under the control of the facility at which they work, the management of that facility should make special arrangements with the employer of the contracted workers to ensure that they are provided with the necessary workers' health surveillance (see para. 6.34(i)).

10.2. The management should make available, in the vicinity of the workplace, suitable facilities for medical examinations in connection with workers' health surveillance.

Occupational health services

10.3. The occupational health services should have the following responsibilities in relation to workers' health surveillance:

(a) To assess the health of workers.
(b) To help ensure initial and continuing compatibility between the health of workers and the conditions of their work.
(c) To establish a record that provides useful information in the case of:
    (i) Accidental exposure or occupational disease;
    (ii) Statistical evaluation of the incidence of diseases that might relate to the working conditions;
    (iii) An assessment for public health purposes of management in relation to protection and safety in facilities in which occupational exposure can occur;
    (iv) Medical–legal inquiries.
(d) To provide workers with counselling on any radiation risks to which they might be subjected, and to provide an advisory and treatment service in the event of personal contamination or overexposure.

Occupational physician

10.4. The occupational physician in charge of the programme for workers' health surveillance should have the following responsibilities:

(a) To carry out medical examinations of workers;
(b) To advise management periodically on the fitness of workers for their intended tasks, on the basis of a the worker's state of health and the employer's requirements for the job;

(c) To give clearance for the return of workers to their normal working environment after being removed from that environment on medical grounds;
(d) To advise, as appropriate, on the arrangements for hygiene at work and the removal of contamination from wounds, in consultation with the radiation protection officer, as appropriate.

10.5. The occupational physician, including any private occupational physician employed on a part time basis, should be knowledgeable, through training and, when necessary, retraining, about the biological effects of radiation exposure, the means of control of exposure, and the interpretation of exposure data and dosimetric assessments [162]. With the support of specialists, as appropriate, the occupational physician should be in a position to use this knowledge in the programme for workers' health surveillance and also to provide counselling to the following categories of workers with regard to radiological health risks:

(a) Occupationally exposed workers who suspect that they are pregnant or who may become pregnant, or who are breast-feeding (see paras 6.2–6.20);
(b) Individual workers who have received, or who may have received, an exposure substantially in excess of the dose limits;
(c) Workers who may be worried about their radiation exposure;
(d) Workers who otherwise request such counselling.

10.6. In order to be able to make judgements about workers' fitness for work, the occupational physician should be familiar with the tasks in the workplace and the conditions in the working environment. For operations involving unusual working conditions, as might be the case for certain mines and mineral processing facilities, the occupational physician should maintain an awareness of such conditions by visiting the workplaces periodically. The employer should provide appropriate opportunities for the occupational physician to develop the necessary degree of familiarity with the tasks in the workplace and the conditions in the working environment.

10.7. The occupational physician should take responsibility for case management in the event of a suspected overexposure. This should include the submission of details of the case to relevant qualified experts, the counselling of the worker, and the briefing of workers' representatives and the worker's family members if appropriate. Further guidance in this area is given in Ref. [162].

PROGRAMME FOR WORKERS' HEALTH SURVEILLANCE

10.8. In terms of para. 3.108 of GSR Part 3 [2], a programme for workers' health surveillance is required to be based on the general principles of occupational health, as set out in Ref. [163], and is required to be designed to assess the initial fitness and continuing fitness of workers for their intended tasks. Further objectives of a programme for workers' health surveillance are:

(a) To provide a baseline of information that can be used in the case of accidental exposure to a particular hazardous agent or in the case of occupational disease and for specific counselling of workers with respect to any occupational health risks (including radiation risks) to which they are, or might be, subjected;
(b) To support the care of overexposed workers.

10.9. The main elements of the programme for workers' health surveillance should be:

(a) Assessment of the health of workers for the purpose of ensuring that they are fit to undertake the tasks assigned to them;
(b) Establishment and maintenance of confidential medical records;
(c) Arrangements for dealing with accidental exposures, overexposures and subsequent follow-up;
(d) Provision of medical advice to management and workers.

10.10. Detailed guidance for persons responsible for the design, establishment, implementation and management of programmes for workers' health surveillance is provided in Ref. [163].

MEDICAL EXAMINATION OF WORKERS

10.11. Medical examinations of occupationally exposed workers should follow the general principles of occupational medicine. Occupational exposure to radiation may not be the only reason for performing medical examinations of workers. Other reasons include exposure to hazards such as noise, dust and chemicals. For example, a periodic review of pulmonary function for workers in a dusty environment could be highly desirable, and the occupational physician should consider the advisability of special investigations such as tests of pulmonary function and, if appropriate, chest X rays. Special assessments and

tests may be warranted if exposures to radiation or exposures to other hazards exceed relevant limits.

10.12. As in any doctor–patient relationship, the occupational physician should keep the worker fully informed of the reasons for particular examinations, as well as of any significant findings bearing on the worker's health and the particular working environment.

10.13. Medical examinations of workers should be performed before the start of employment, periodically thereafter and at the termination of employment.

10.14. A medical history and assessment should be established for each worker for the following purposes:

(a) To determine fitness for the specific work for which the worker is to be employed;
(b) To provide a baseline for use in the consideration of changes to specific work practices;
(c) To provide a baseline for use in assessing an occupational disease or overexposure.

10.15. The initial medical examination should be aimed at assessing the worker's health and fitness for the intended tasks and identifying whether the worker has a condition that might necessitate special precautions during work. However, it should be rare for the radiation component of the working environment to influence significantly the decision about the fitness of a worker to undertake work with radiation, or to influence the general conditions of service. The medical conditions that the occupational physician should look for include those that would affect the ability to use and wear protective clothing and equipment, the ability to hear alarms and respond to radiation hazards, and the ability to use specialized tools and equipment.

10.16. Fitness for work with radiation depends on the worker's state of health and the type of work involved, as illustrated by the following examples:

(a) If a worker's duties are such that the use of respiratory protective equipment is required, the occupational physician should examine the fitness of the worker for wearing respiratory protective equipment, including checks on the integrity of lung function.
(b) If a worker is engaged in the handling of unsealed sources, fitness for work could be influenced by the presence of skin disease such as eczema or

psoriasis. In such cases, the decision regarding fitness should be based on the nature, extent and evolution of the disease and the nature of the job. Workers with such diseases should not necessarily be excluded from work with unsealed radioactive substances if the levels of activity are low and provided that appropriate precautions, such as covering the affected parts of the body, are taken.

(c) If a worker is required to work with radiation sources, fitness for work could be influenced by a psychological disorder. In such cases, the decision on fitness should take account of the safety implications of symptomatic episodes of such a disorder. The primary concern is whether such workers could represent a danger to themselves or to their co-workers or the public.

10.17. There is no inherent reason why a worker who has previously undergone radiotherapy should be excluded from work with radiation. Each case should be evaluated individually, by taking into account the outcome of the radiotherapy treatment, the general prognosis and other health considerations, the understanding and wishes of the worker, and the nature of the work.

10.18. In the periodic medical examinations, the occupational physician should confirm that no clinical condition that could prejudice the health of the worker has developed during work in areas involving occupational health hazards, including hazards due to radiation. The nature of a periodic medical examination should depend on factors such as the type of work that is undertaken, and the age and health status, and possibly the habits of the worker (e.g. smoking habits). For example:

(a) The skin should be examined where the nature of the work creates a potential for localized skin damage from exposure, particularly of the hands.

(b) A worker who has already received accumulated doses to the lens of the eye of more than 0.5 Gy, or who could, after a few more years, accumulate doses in excess of this level, may need to be subject to regular ophthalmological examination. This relates to the risk of detectable opacities and visual impairment, which might affect the ability of the worker to carry out the intended tasks (e.g. performing image guided interventional procedures).

10.19. The frequency of periodic medical examinations should be based on the state of health of the worker and on the type of work involved. Normally, exposure to radiation should not, in itself, be a reason for carrying out periodic medical examinations more frequently than usual.

10.20. In keeping with good practice for occupational health, the occupational physician should ensure that a worker, on return from absence due to injury or illness, is fit to resume work.

10.21. On completing a medical examination, the occupational physician should communicate the conclusions in writing to both the worker and the employer. These conclusions should not contain information of a medical nature, but should at least categorize the worker as one of the following:

(a) Fit for work in a specific job or trade;
(b) Fit for such work with certain restrictions (e.g. no work necessitating respiratory protective equipment);
(c) Unfit for the work in question.

With regard to (c) above, the occupational physician should have the authority to declare workers temporarily or permanently unfit on medical grounds for their regular work or to recommend the transfer of a worker to other work. The occupational physician should also have the authority to advise the employer on reinstating such workers in their normal duties on medical grounds.

10.22. In the case of an observed ailment likely to have been caused by prevailing working conditions, the occupational physician should advise the management of the need to investigate the working conditions. Where appropriate, the management should take corrective actions in consultation with the occupational physician.

10.23. In a medical examination at the termination of employment, any work related impairment should be identified and, if necessary, arrangements should be made for further periodic and follow-up examinations by the worker's physician after employment has ceased (see Ref. [163] for additional guidance). This is line with ILO Recommendation No. 147 [164], which states that:

> "12. The competent authority should ensure that provision is made for appropriate medical examinations or biological or other tests or investigations to continue to be available to the worker after cessation of the assignment".

10.24. The data compiled from the medical assessments may be useful for epidemiological studies.

NOTIFICATION OF AILMENTS AND OF OVEREXPOSURE

10.25. Workers should be encouraged to report any significant ailment promptly to the occupational physician.

10.26. Workers should report any suspected accidental intake of radioactive substances to their supervisor and to the radiation protection officer. The occupational physician should be informed if it is suspected that an accidental intake exceeds a limit specified by the regulatory body and should be advised of the outcome of any investigation to establish whether such an intake has actually occurred. The occupational physician may be made part of the proceedings for the investigation of the overexposure.

10.27. Where a worker has received a dose in excess of an investigation level (see paras 3.122–3.127), the regulatory body may require notification and investigation of the circumstances of the exposure.

MEDICAL RECORDS

10.28. Medical records should include records of all medical assessments — pre-employment, periodic, special and post-illness assessments, and assessments at the termination of employment — laboratory reports, sickness reports and medical history reports. Information on radiation exposures should also be recorded, where appropriate, especially in cases of overexposure. Medical records should be kept confidential and should be preserved in a manner approved by the regulatory body. Medical records should be retained for at least the lifetimes of the workers concerned. However, because of the possibility of litigation, a longer period of retention of records should be considered.

MANAGEMENT FOR OVEREXPOSED WORKERS

10.29. In accordance with the conditions of authorization, the management should draw up formal plans to deal with situations in which workers might be overexposed. These plans should address management for overexposed workers and the health consequences that might be encountered. The plans should specify the necessary actions to be taken, and the management should allocate resources for carrying out those actions.

10.30. In the case of accidental exposure or overexposure, the occupational physician should cooperate with the management to ensure that all suitable arrangements are made to evaluate the severity of the exposure.

10.31. If an overexposure is suspected to have occurred, the management should promptly undertake an investigation to assess the doses received by the worker(s) concerned. The investigation should include the reading of personal dosimeters and of any monitoring instruments and, in the case of internal exposure, in vivo or in vitro monitoring, as appropriate.

10.32. Assessed doses that are close to dose limits are unlikely to call for anything more than an investigation of the causes, so that appropriate lessons can be identified. They do not necessitate any special medical investigations or treatment. Only at doses much higher than the dose limits (i.e. doses of 0.1 Sv or higher) would special dose investigations involving biological dosimetry (e.g. chromosomal aberration analysis in somatic cells, mainly lymphocytes) and further extended diagnosis or medical treatment be necessary (see paras 4.30–4.32). The medical treatment of those persons with external exposure to radiation at high levels should address any adverse health effects, particularly deterministic effects.

10.33. Measures to reduce the committed dose may be warranted in the event of a worker having suffered a significant intake of radionuclides. Such workers should be forewarned of the possibility of medical intervention to reduce the uptake in certain situations. The action to be taken will depend on the radionuclides involved, the magnitude of the committed equivalent dose to relevant organs, and the efficiency of, and risk associated with, the protective measure. The action should be implemented only when the dose reduction would outweigh the side effects. Examples of such therapies include increasing the rate of excretion from the body of incorporated actinides by using Ca-DTPA treatment, forced diuresis after an intake of tritium, and surgical excision of contaminated wounds.

10.34. Detailed investigations of accidents, their circumstances and consequences should involve specialists in different fields, in particular the occupational physician and a radiation specialist. There should be close liaison between these specialists in order to ensure that all actions undertaken to provide medical treatment are correctly coordinated. If it is suspected that the doses received are close to or above the thresholds for deterministic effects, the investigation should determine, as accurately as possible, the absorbed doses and their distribution over the body, and should include appropriate medical examinations of the affected worker(s).

Appendix I

EXPOSURE OF WORKERS TO NATURALLY OCCURRING RADIOACTIVE MATERIAL

I.1. As with other occupational exposures, the only reliable way of assessing the effective dose received by a worker exposed to naturally occurring radioactive material is through a properly developed monitoring programme conducted in the relevant workplace. However, for exposure to gamma radiation and exposure due to airborne dust, it is possible to establish, in advance, a broad indication of the expected dose if there is a reasonable knowledge of the characteristics of the material and the work situation in which the material is used. This is because the dose is quite strongly influenced by the activity concentrations of radionuclides in the material, reflecting the underlying linear relationship between these two parameters. A broad indication of the dose from exposure to gamma radiation and exposure due to airborne dust can be used during the prior radiological evaluation as a prioritization tool to identify, on the basis of activity concentrations in process materials, the types of industrial process and scenarios of exposure in which the need for measures for protection and safety is likely to be greatest.

I.2. A description is given in Ref. [24] of the derivation of indicative relationships between dose and activity concentration for a range of process materials and associated exposure scenarios likely to be encountered in industrial activities involving naturally occurring radioactive material. Three basic categories of process material are considered:

(a) Large quantities of material (e.g. an ore body or a large stockpile);
(b) Small quantities of material (e.g. mineral concentrates, scale and sludge);
(c) Material that has been volatilized in a high temperature process (i.e. precipitator dust and furnace fume).

The results are summarized in Table 7. In actual situations, the doses are likely to be considerably lower because of the conservative nature of the assumptions made in the dose modelling process.

TABLE 7. INDICATIVE RELATIONSHIP BETWEEN DOSE AND ACTIVITY CONCENTRATION FOR OCCUPATIONAL EXPOSURE TO GAMMA RADIATION AND DUE TO AIRBORNE DUST

Category of material	Examples	Annual dose per unit activity conc. of the radionuclide with the highest activity conc. (mSv/a per Bq/g)	
		Min.	Max.
Bulk quantity	Ore body Large stockpile	0.02	0.4
Small quantity	Mineral concentrate Scale Sludge	0.008	0.04
Volatilized material in which only ^{210}Pb and ^{210}Po are of concern	Furnace fume Precipitator dust	0.000 6	0.003

Source: Table 2 of Ref. [24].

I.3. The implications of the results in Table 7 can be illustrated by the following two examples:

(a) A worker's job involves, on a routine basis, close proximity to a 500 000 t stockpile of material in which the highest mean activity concentration of an individual radionuclide in the ^{238}U decay series or the ^{232}Th decay series is 5 Bq/g. Depending on the type of material, the annual effective dose expected to be received by the worker would range from 0.1 mSv/a (5 Bq/g × 0.02 mSv/a per Bq/g) to 2 mSv/a (5 Bq/g × 0.4 mSv/a per Bq/g). This would suggest that, in terms of the graded approach, the exposure scenario would be of only minimal concern with regard to protection and safety.

(b) A worker's job involves, on a routine basis, close proximity to 100 kg of process residue in which the highest mean activity concentration of an individual radionuclide in the ^{238}U decay series or the ^{232}Th decay series is 250 Bq/g. Depending on the type of material, the annual effective dose expected to be received by the worker would range from 2 mSv/a (250 Bq/g × 0.008 mSv/a per Bq/g) to 10 mSv/a (250 Bq/g × 0.04 mSv/a per Bq/g). This would suggest that, in terms of the graded approach, the exposure scenario would be of fairly significant concern for protection and safety.

Appendix II

METHODS AND SYSTEMS FOR INDIVIDUAL MONITORING FOR ASSESSMENT OF EXTERNAL EXPOSURE

II.1. Only a few dosimetric methods are widely used for individual monitoring purposes. They differ in the technology used to detect radiation. As a consequence, they also differ with regard to such characteristics as the ability to measure radiation of various types and energies, the detector size, sensitivity, technological complexity, ease and degree of automation, and robustness with respect to climatic conditions. In selecting a dosimetry system, these characteristics should be carefully considered in the light of the local circumstances.

PHOTON RADIATION AND BETA RADIATION

Photographic film dosimetry

II.2. Photographic film dosimeters commonly consist of a photographic film placed inside a suitable holder containing appropriate filters. Such assemblies are often referred to as film badges.

II.3. The emulsion of the film is made of silver bromide crystals suspended in a gelatinous medium. A thin layer of this emulsion is coated uniformly onto a plastic base. The action of ionizing radiation on the grains in the emulsion produces a latent image. In subsequent developing, the silver ions in the latent image produce permanent blackening. The optical density is measured with a densitometer, and is a function of the type and energy of the radiation being measured. Photographic film dosimeters are used most widely for monitoring photon and beta radiation. They may also be used for indirect measurement of thermal neutron dose, through the capture of thermal neutrons with a cadmium filter (by n–γ reaction) and the assessment of the blackening of the film produced by the resulting gamma radiation as an indication of the neutron dose.

II.4. The sensitivity of the film as a function of photon energy is quite different from that of human tissue. For instance, the optical density at 50 keV can be 25 times higher than that at 1.25 MeV (the average photon energy of ^{60}Co) for the same dose to tissue. Several methods have been developed to compensate for this energy dependence. Most of them use filters made of various metals (such as aluminium, copper, lead and tin) and of various thicknesses, mounted in the film holder in front of the film. These filters attenuate the radiation in a manner

dependent on its photon energy, which results in areas of different optical density from which information on the radiation spectrum can be drawn. 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. In practice, empirically developed algorithms are used to combine the 'apparent gamma doses' of the different areas, resulting in a reasonably accurate estimation of the quantities $H_p(10)$ and $H_p(0.07)$.

II.5. Even with appropriate filters and algorithms, it is difficult to determine $H_p(10)$ for photon energies less than about 20 keV or greater than 3 MeV without considerable expertise and some knowledge of the energy spectrum of exposure. The type of incident radiation and the dose can be estimated from the responses behind different filters.

II.6. The optical densities of the film depend not only on the radiation energy, the filters used and the dose, but also on the type of film, the temperature of the developer, the developing time and the climatic conditions (temperature and humidity) in which the film was exposed before being processed. Film dosimeters are susceptible to temperature and humidity, resulting in fading of the latent image [165].

II.7. A further complication arises from the fact that the dose–density relationship is not linear but sigmoid in shape. This implies that, together with the customer's films, a set of calibration films (that are exposed to the entire range of radiation doses, commonly using ^{137}Cs or ^{60}Co gamma radiation) should be developed. From the optical densities of these films, a calibration curve can be drawn that is used to express all optical densities in terms of apparent gamma dose. The calibration curve can, by applying curve fitting procedures, easily be expressed as a mathematical function that is used to convert the measured optical densities into an apparent gamma dose. This should be done with every batch of newly bought films. There is no way to take into account sensitivity differences within a batch of films because a film can be irradiated only once.

II.8. Most densitometers are capable of measuring optical densities[25] of between 0.02 and 4.0 (corresponding to the transmittance of light through the film of

[25] For a given wavelength of light, the optical density of a material (also referred to as the absorbance A) is the ratio of the intensity I of light passing through a material to the intensity I_0 of light falling on the material, expressed logarithmically according to the expression $A = -\lg (I/I_0)$.

between 95% and 0.01%). The corresponding dose range is rather limited and most films used for individual monitoring, therefore, have two layers of sensitive emulsion, one on each side of the carrier, which differ in sensitivity by a factor of about a hundred. In the case of a severe overexposure, the sensitive layer (which will be saturated) can be removed and the remaining insensitive layer can be used for quantitative dose measurements of doses of about 2–10 Sv. A separate calibration curve is necessary for this emulsion.

II.9. 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 of 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 'control films' simultaneously with the dosimeters.

Thermoluminescence dosimetry

II.10. Thermoluminescence dosimetry is based on the excitation (followed by subsequent trapping) of electrons by ionizing radiation and the subsequent release of the trapped electrons by heating. This results in the emission of light, the amount of which is directly related to the radiation dose initially received by the material. The relationship between the amount of light emitted during readout and the quantity to be measured is determined by means of calibration. After readout, the detector can be reused, normally following an annealing procedure.

II.11. Quantitative measurement of the light output from a thermoluminescent dosimeter during readout is usually done by using a photomultiplier tube. The photomultiplier output plotted as a function of 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 area under the glow curve is used as a measure of dose.

II.12. In principle, the use of thermoluminescent dosimeters is simple and straightforward. However, care should be taken to always apply the correct readout and annealing procedures, otherwise significant variations in the sensitivity of thermoluminescent dosimeters may occur. Although the amount of fading is less than for film dosimeters, this phenomenon is complicated. Care and experience are therefore required for making measurements of adequate accuracy and precision.

II.13. The use of thermoluminescent dosimeters has several characteristics that are beneficial for personal dosimetry applications. This has resulted in their wide application over the years because of the progress made in the development of materials for thermoluminescent dosimeters and the current sophistication of reader instrumentation for such dosimeters. The successful use of thermoluminescent dosimeters as a routine means of measuring radiation dose has been demonstrated many times (e.g. in international intercomparison studies [166, 167]).

II.14. Many materials for thermoluminescent dosimeters have been manufactured and investigated, but only a few are routinely applied for individual monitoring purposes. The most widely used materials are lithium fluoride (LiF:Mg, Ti or LiF:Mg, Cu, P) and lithium borate ($Li_2B_4O_7$:Mn). The material LiF:Mg, Cu, P is increasingly being used because of its higher sensitivity and lesser susceptibility to fading compared with LiF:Mg, Ti. On account of their low effective atomic number (7.3–8.3), these materials exhibit a response versus energy curve that is within 20% of that for soft tissue. This obviates the need for using compensating filters and, hence, allows for a simple design of dosimeter for the measurement of $H_p(10)$ and $H_p(0.07)$. However, thermoluminescent dosimeters may not have a good energy response if they are to be used for measuring photons with energies less than about 20 keV [168].

II.15. The quantity $H_p(0.07)$, which becomes important when photons below 12 keV or beta radiation are to be measured, requires the application of a very thin (~4 mg/cm^2) detector covered by a very thin (~4 mg/cm^2) protective layer. Such thin detectors are available commercially in two versions: (i) a thin radiation sensitive layer on top of a more robust radiation insensitive carrier; and (ii) regular thermoluminescent dosimeter material loaded with a small amount of carbon (the latter preventing the luminescence from layers deeper than 4–10 mg/cm^2 from reaching the detector during the readout process). Because of the very small amount of detector material available for dose measurements, the sensitivity of thin thermoluminescent dosimeters is low. However, owing to the use of LiF:Mg, Cu, P, these detectors now have a suitable detection threshold, and they have the most appropriate material for extremity dosimetry when beta radiation is involved [169].

II.16. Thermoluminescent dosimeter materials with higher effective atomic numbers (10.2–16.3) are also used because of their greater sensitivity. Examples include calcium fluoride (CaF_2), calcium sulphate ($CaSO_4$:Dy or $CaSO_4$:Tm) and aluminium oxide (Al_2O_3). Thermoluminescent dosimeters incorporating these materials are used in badges with several filters. These dosimeters essentially

mimic the characteristics of the film dosimeter, giving a broad indication of the energy of the radiation that gave rise to the dose received by the wearer.

II.17. In contrast to the response of photographic film, the response of thermoluminescent dosimeters (i.e. the luminescent light output) varies almost linearly with dose over a very wide dose range, up to at least 2 Sv or even higher: up to 5 Sv for LiF and even higher for $CaSO_4$:Dy. The behaviour of LiF is supralinear above a few sieverts, up to saturation at several thousand sieverts. Modern thermoluminescent dosimeter systems (i.e. combinations of thermoluminescent dosimeters and related readout equipment) are capable of measuring doses down to 10 μSv with a satisfactory accuracy and precision.

II.18. Manual, semi-automatic and also very sophisticated and highly automated thermoluminescent dosimeter systems are available commercially (see Ref. [170]). For smaller services, manual or semi-automatic systems are usually adequate.

Photoluminescence dosimetry

II.19. 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 induced luminescent 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 [171].

II.20. Phosphate glasses can be produced on a large scale with good reproducibility and constant sensitivity. 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. This eliminates some of the drawbacks of the older, conventional readout technique, which needed cleaning of the glass and subtraction of the pre-dose in order to measure doses below 100 μSv.

II.21. Because of the high atomic number of some glass materials, energy compensating filters are used. An energy dependence within ±15% is achievable for photon energies above 15 keV.

II.22. 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. A complete phosphate glass dosimetry system with an automatic readout using ultraviolet laser excitation is suitable for use in large scale systems for individual monitoring [172, 173]. Photoluminescence dosimetry systems are available commercially and are already widely used, with excellent results having been achieved in intercomparisons.

Optically stimulated luminescence dosimetry

II.23. Optically stimulated luminescence dosimetry is similar in principle to thermoluminescent dosimetry and photoluminescence dosimetry. Optically stimulated luminescence techniques use optical methods to release the energy of electrons trapped in luminescent materials following their exposure to ionizing radiation [174–178]. The detection system is based on the use of aluminium oxide (Al_2O_3:C) luminescent material. The source of light used to excite the material is typically provided by a laser or light emitting diode. Optically stimulated luminescence can be performed in pulsed or continuous mode. In the latter mode, the stimulating light is separated from the emitted light by a series of filters. Technology for optically stimulated luminescence provides the option of reprocessing the dosimeter at a later time, if desired, since only a small portion of the optically stimulated luminescence signal is erased during a single readout. The detection level is low because of the high sensitivity of the phosphor. A disadvantage is that the Al_2O_3:C material is not tissue equivalent, requiring the use of filters and a suitable calculation algorithm for the determination of $H_p(10)$. The relationship between the amount of light emitted during readout and the quantity of radiation to be measured is determined by calibration.

II.24. Widespread use is now being made of optically stimulated luminescence dosimetry based on Al_2O_3:C, as a result of the development of material with the required degree of sensitivity and of practical readout systems. A second commercial dosimetry system based on optically stimulated luminescence has been introduced in recent years. It works with beryllium oxide (BeO) material, which has the advantage of being nearly tissue equivalent, thus avoiding the need for filters or a calculation algorithm for determining $H_p(10)$.

Direct ion storage dosimetry

II.25. The direct ion storage dosimeter is based on the combination of an ion chamber and a non-volatile electronic charge storage element. The direct ion

storage integrates the doses received, and allows repeated readouts in a small on-site reader. The readout takes only a few seconds and can be performed by the worker at their convenience. The dosimeter does not need to be returned to the dosimetry service, except for resetting (e.g. once a year). The results recorded in the reader can be sent to the service automatically at every readout. The direct ion storage is designed to measure the personal dose equivalent $H_p(10)$ and $H_p(0.07)$ to the required accuracy [179, 180]. It has a high sensitivity, similar to that of an active personal dosimeter, it exhibits no fading and it is not influenced by climatic conditions. The direct ion storage dosimeter is a passive device by nature, although it can be used in a special holder as an alarm dosimeter with a direct reading. The direct ion storage dosimeter is finding more and more applications and is now approved in some States as an official or legal dosimeter.

Active personal dosimetry

II.26. Many types of active personal dosimeter are commercially available. They are usually based on an energy compensated Geiger–Müller counter or a silicon detector.

II.27. Although the majority of these dosimeters are useful as alarm dosimeters for use in controlled areas and for short term radiation control of workers' exposures, they are not all suitable for use as official or legal dosimeters. This is mainly because some dosimeters do not measure beta radiation in addition to photons, some do not record both $H_p(10)$ and $H_p(0.07)$, and some have too high an energy threshold for photons. Other important factors that should be considered are reliability and the risk of data loss [181]. Furthermore, for most devices, difficulties are encountered in measuring pulsed radiation. Some active personal dosimeters do not record both $H_p(10)$ and $H_p(0.07)$, and dosimeters of two different types may, therefore, need to be worn.

II.28. The development of improved dosimeters is continuing and more and more devices are now technically equivalent to, and as reliable as, passive devices. Recently, active personal dosimeters have been accepted as legal dosimeters for routine dosimetry in some States [182]. When used for such purposes, only one dosimeter, serving both alarm and monitoring purposes, should be worn by the worker. An overview of available active personal dosimeters has been compiled and several such dosimeters have been assessed against applicable standards [183, 184]. On the basis of the findings of these investigations, it is evident that the energy and the directional response characteristics of recently developed active personal dosimeters are, in most cases, as good as those of passive dosimeters. The data transfer characteristics and reliability levels are

comparable to those of passive dosimeters and the technical characteristics are similar or better. Care should be taken when using active personal dosimeters in certain types of radiation field such as low energy X ray radiation fields and pulsed fields [185].

NEUTRON RADIATION

II.29. Information on individual neutron monitoring can be found in Ref. [186]. An evaluation of a wide range of neutron personal dosimeters was carried out, in which the dosimeters were compared with reference values in a range of real and simulated workplace radiation fields (see Ref. [61]).

Nuclear track emulsions

II.30. Nuclear track emulsions can be used 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 [187].

II.31. 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.32. Neutrons with energies below 10 eV can be detected through interactions with the nitrogen nuclei of the gelatine, resulting in the production of recoil protons from $^{14}N(n, p)^{14}C$ reactions.

II.33. A microscope 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 dose measured depends on the skill of the operator in recognizing the tracks in the emulsion.

II.34. One disadvantage of nuclear track emulsions is their high rate of fading before being processed. 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.35. 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. Owing to these disadvantages, including the high neutron energy threshold, nuclear track emulsions are increasingly being replaced in personal dosimetry by other methods. In general, the method using nuclear track emulsions is to be avoided.

Solid state nuclear track detectors

II.36. 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 various plastics [188]. By etching the surface of the detector with suitable reagents, the damage zone along the particle track can be removed and the etch pits can be enlarged to become visible under an optical microscope. The application of electrochemical etching greatly enlarges the track size, and track densities can easily be measured (i.e. by counting).

II.37. 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 concentration and temperature of the etchant and the etching time). These parameters should be optimized for each material and for each particular application.

II.38. For neutron dosimetry, three detector types have commonly been used: fission track detectors, recoil track detectors and (n, α) track detectors. These are described briefly in paras II.39–II.42. A more comprehensive discussion of measurement techniques for track detection can be found in Ref. [189].

Fission track detectors

II.39. 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 and 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 some States because of their radioactivity.

Recoil track detectors

II.40. The elastic scattering of neutrons with the nuclei of plastic detectors, such as poly allyl diglycol carbonate or CR-39 (allyl diglycol carbonate) [188, 190–192], can produce charged recoil particles such as protons or ions of carbon, oxygen and nitrogen. These recoils produce latent tracks that 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. Because of the linear energy transfer 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. In addition to poly allyl diglycol carbonate, the most common detector materials are polycarbonate and cellulose nitrate.

II.41. A dosimeter based only on poly allyl diglycol carbonate has an energy threshold of about 100–150 keV, but its low energy response can be improved by the use of a plastic filter that contains nitrogen. Low energy neutrons react with nitrogen by the capture process to produce protons with an energy of about 0.5 MeV. Its angular response is not good but if the mean response is averaged over angles of 0°, 20°, 40° and 60°, a response that is flat to within ±30% is obtained in the 0.15–14 MeV region. The use of the nitrogenous plastic filter also produces a satisfactory response from neutrons in the energy range from thermal to 10 keV. This type of detector is not sensitive to photons, it does not suffer much from fading and the dose threshold is as low as 0.1 mSv. Depending on the required sensitivity, no workplace correction factor may be needed. Automatic readers for use with this type of detector have also been developed and are commercially available. However, to operate a track etch dosimetry service requires a high level of technical expertise; the precision of the etching procedure and the interpretation of the tracks produced are both important factors for obtaining good results.

Track detectors based on (n, α) reactions

II.42. 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 kiloelectronvolts. 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.

Thermoluminescent albedo dosimeters

II.43. 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 can therefore serve as an albedo detector.

II.44. Albedo dosimeters are usually thermoluminescent dosimeters, such as ^{6}LiF in boron loaded plastic encapsulations, which separate the albedo neutrons from incident thermal neutrons. Owing to the photon sensitivity of thermoluminescent dosimeters, the neutron dose reading is given by the difference between the ^{6}LiF and the ^{7}LiF detector readings.

II.45. 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 [193, 194]. 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.46. The neutron response depends on the neutron energy spectrum, which can 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. Albedo dosimeters are also very sensitive to the position of the dosimeter on the worker's body, since they mainly detect the neutrons emerging from the body.

II.47. 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 thermoluminescent dosimeter reader. The track detector then only needs to be processed if a significant exposure is indicated by the thermoluminescent dosimeter.

Bubble detectors

II.48. A bubble detector consists of a tube in which superheated liquid drops are dispersed in a polymer gel. Neutrons passing through the detector create protons that can deposit sufficient energy in the droplet for the threshold energy to be surpassed and for the droplets to become visible vapour bubbles, which are trapped at the sites of formation [195]. The number of bubbles gives a measure of the neutron dose. Bubble detectors are not sensitive to photons, have a very high sensitivity (down to the microsievert range) and have a good response to the dose equivalent rate above a certain neutron energy threshold, usually around 100 keV. Thermal neutrons can be detected by a special bubble detector with ^{6}Li dispersed in it. The disadvantages of the bubble detector are its limited range of energies and doses, its sensitivity to shock and its temperature dependence, although detectors compensated for temperature are available. A bubble detector does not require a workplace correction factor, but counting the bubbles is a labour intensive process. The bubble detector is a completely passive device that can be stored until needed for use. It does not require any electronic apparatus for measurement or reading. However, an automatic reader that is computer controlled can be used to perform the reading if a large number of detectors are used routinely.

Electronic neutron dosimeters

II.49. Active personal neutron dosimeters have been developed [196]. Their principle of operation is the same as that for active personal photon dosimeters, except that a plastic material is positioned in front of the diodes to convert the neutrons to protons which are then measured. The introduction of ^{6}Li or ^{10}B can make the dosimeter sensitive to thermal neutrons. Gamma energies can be discriminated electronically by means of an energy deposition threshold. Active personal neutron dosimeters have the advantages of being direct reading and easy to use. At present, however, their energy response is not ideal, their sensitivity to fast neutrons is low and they often require a workplace correction factor.

Criticality dosimeters

II.50. Criticality dosimeters are a separate class of neutron dosimeter. Their function is to estimate the neutron doses received in the event of a criticality accident. The operating principles of criticality dosimeters need to be different from those of routine neutron dosimeters because in a criticality accident high neutron dose rates in short pulses are expected. Criticality dosimeters normally contain activation detectors, for example elements such as gold, cadmium, indium and sulphur. More information can be found in Ref. [148].

Appendix III

WORKPLACE MONITORING INSTRUMENTS FOR ASSESSMENT OF EXTERNAL EXPOSURE

III.1. Workplace monitoring instruments are primarily intended to provide information on the dose rates within the workplace to permit decisions to be made on its occupancy. It is necessary to know the dose 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 or installed workplace monitoring instruments are often equipped with remote displays or audible alarms. Apart from some engineering differences, their detectors and operating methods are similar to those of portable workplace monitoring instruments. A comprehensive discussion of monitoring methods can be found in Refs [197, 198].

PHOTONS (GAMMA RADIATION AND X RAYS)

Ionization (ion) chambers

III.3. Ionization chambers are the simplest form of radiation detector and can be used for the detection of radiation in various circumstances. The ionization chamber is a gas filled detector. The detection principle is based on the measurement of the charge from the number of ion pairs created within the gas by the incident radiation. The charge is collected by applying a voltage across two electrodes and can be measured as a current (in the 'current mode') or as a voltage (in the 'pulse mode').

III.4. To ensure that the output signal is proportional to the amount of energy released in the chamber, the correct voltage should be applied.

III.5. Hand-held monitoring instruments 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 $H'(0.07, \Omega)$.

III.6. Hand-held instruments for use at normal occupational exposure levels (e.g. dose rates of a few microsieverts per hour) generally have chamber volumes in the range of 300–700 cm^3. Installed instruments designed for use where beta radiation and low energy photon radiation are not expected often have large (of the order of 5000 cm^3) steel walled chambers filled with argon at high pressure. These have a large dynamic range, from about 0.1 μSv/h to as much as 1 Sv/h.

Proportional counters

III.7. Proportional counters are based on the same principle as that for ionization chambers, but use gas multiplication of electrons to enhance the sensitivity by applying a higher voltage between the electrodes. To optimize detection, noble gases are generally used in order to avoid the creation of negative ions.

III.8. Proportional counters can be used as pulse detectors, allowing the measurement of photon dose rates from 0.1 μSv/h to 10 Sv/h. The main advantages of commercial proportional counters are their high sensitivity, wide range of dose rates and low energy dependence. However, to achieve a stable multiplication factor, a stable high voltage supply is required. This makes the instrument considerably more expensive than a ionization chamber or Geiger–Müller counter. Proportional counters can be used as continuous current detectors, but are almost never used like this because the signal of the proportional counter drops very rapidly.

Geiger–Müller counters

III.9. The strong electric field in Geiger–Müller counters causes a Townsend avalanche (cascade of electrons) over the complete anode every time an ionizing particle hits the detector. This means that every single event in the Geiger–Müller counter, regardless of the energy of the incoming particle, causes a signal in the detector with the same magnitude, meaning that all information about the energy of the incoming particle is lost. To be able to measure ambient dose equivalent, the Geiger–Müller counter has to be calibrated in terms of the pulse frequency of the counter as a function of the energy of the incoming particles.

III.10. Geiger–Müller 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.11. Geiger–Müller counters are popular for use in X ray and gamma radiation 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. Quenching, either external or internal, restores the Geiger–Müller counter to a working condition. Care should be taken to ensure that the dose rate indication does not fall back on the scale at high count rates; this is a fundamental test that should be performed during type testing. Geiger–Müller counters are best used to monitor low levels of contamination or radiation dose rates.

III.12. The use of Geiger–Müller counters in pulsed radiation fields, such as with some X ray generating equipment, may lead to a serious underestimation of the radiation quantity measured. For this reason, extreme caution is needed when Geiger–Müller counters — or indeed any pulse counting detectors — are used in such situations.

Semiconductor detectors

III.13. In semiconductor materials, such as silicon, ionization after interaction with ionization radiation causes electrons to jump from the valence band to the conduction band where they are free to move through the entire crystal. To be able to detect the freed electrons, the crystal is surrounded by two electrodes.

III.14. Since the energy gap between the valence band and the conduction band is only a few electronvolts, the output signal is greater, with a higher signal to noise ratio than for gas filled detectors, for which the ionization energy is typically about 30 eV. The small energy gap also gives the advantage of a better resolution. However, measures should be taken to avoid thermal agitation of charge carriers.

III.15. Dose rates can be measured with silicon diodes used as pulse generators (at low dose rates) or as photocurrent generators (at high dose 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.

Scintillation detectors

III.16. In scintillation detectors, excitation of electrons occurs on interaction with ionizing radiation, and visible light is emitted. There are two types of scintillator:

(a) Inorganic scintillators are crystals of alkali halides or oxides grown in high temperature furnaces. The scintillation properties are a consequence of the crystalline structure and are thus only present in the solid state of the material.
(b) Organic scintillators comprise aromatic hydrocarbons and take the form of plastics or liquids. The scintillation process can be traced back to the molecule itself, meaning that the process takes place irrespective of the physical state of the material.

III.17. Scintillators are used in combination with a photomultiplier tube to convert and enhance the light signal to an electronic signal that can be measured with relative ease.

III.18. Inorganic scintillators, such as NaI(Tl) crystals, are widely used in gamma spectroscopy and make very sensitive detectors. However, their response is highly 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 also very sensitive.

III.19. When used to measure exposure rate or air kerma rate, organic scintillators are sufficiently similar to air in terms of their effective atomic number that they 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 a high atomic number in front of the crystal can partially offset this drop, and commercially available monitoring instruments allow the measurement of photons with energies as low as 20 keV.

III.20. Scintillation instruments can be used for measurement of all types of X rays and gamma radiation [199]. 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 dose rates at natural background levels.

Ionization chambers

III.21. It is possible to measure the dose equivalent rates both from beta radiation (or low energy X rays) and from strongly penetrating photon radiation using a single detector. In this case, the detector (ion chamber) is fitted with a 'window' that can be opened or closed (see para. III.22). When it is closed, 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 rate is estimated by subtraction.

III.22. Most survey measurements for beta radiation and low energy photons are made with small, portable ion chambers that can also be used for X ray and gamma radiation surveys. One side of the chamber comprises a thin conducting plastic sheet that is covered when measuring photons ('window closed') with a piece of material equivalent to 1 cm of tissue. This cover is removed for measuring beta radiation [200]. Another type of beta survey meter is constructed entirely with thin walls. Such a chamber might not be appropriate for the measurement of directional dose equivalent.

III.23. The walls of an ion chamber to be used for the measurement of beta radiation 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 absorption and backscattering by the body.

Geiger–Müller counters

III.24. Thin walled or thin windowed Geiger–Müller counters are also used for the detection of beta radiation. If the counter is provided with a cover that is sufficiently thick to stop beta radiation, the difference between readings with and without the cover can be used to distinguish between beta radiation and gamma (including bremsstrahlung) radiation. Geiger–Müller detectors with thin end windows, in particular, may have acceptable energy dependence for beta dose rate monitoring in the workplace, and have the additional advantage of having a small size that is capable of detecting a relatively low dose rate.

Semiconductor detectors

III.25. 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 measurements of beta radiation and low energy photon radiation, thin, sensitive layer silicon diodes are suitable for the evaluation of $H'(0.07, \Omega)$. However, 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 measuring dose rates from beta radiation and low energy photons in operational radiation protection.

Scintillation detectors

III.26. A good dose rate monitor for $H'(0.07, \Omega)$ for beta radiation 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 Geiger–Müller detector, or in the current mode at high dose rates.

NEUTRON RADIATION

Moderator based survey instruments

III.27. Moderator based survey instruments are the most common instruments for monitoring neutron fields [186, 201]. They consist of a hydrogenous moderator that moderates the neutrons and detects the thermalized neutrons using detectors such as proportional counters filled with boron trifluoride (BF_3) or ^{3}He gas, or ^{6}LiI scintillators. The neutrons are detected by $^{10}B(n, \alpha)^7Li$, $^3He(n, p)^3H$ or $^6Li(n, \alpha)^3He$ reactions, the characteristics of which allow the achievement of good discrimination from gamma radiation. By choosing an appropriate thickness for a moderating shield, or by varying the wall thickness and the gas mixture and pressure, the response to neutron radiation can be adjusted to give an output that is roughly proportional to the dose equivalent rate. Crude neutron spectrometry can be achieved by mathematically analysing the responses of a set of moderated spheres of different diameters (see Ref. [202] for a description). The responses for several moderated neutron instruments to operational neutron radiation fields can be found in Ref. [203].

III.28. By thermalizing the neutrons in a hydrogenous moderator, an instrument with an approximately energy independent response to the dose equivalent rate

for neutrons up to 10 MeV was developed in the 1960s [204]. The instrument uses a BF_3 proportional counter surrounded by a perforated cadmium shield in a cylindrical moderator, and it suffers from some anisotropy in response (by a factor of two or more). This anisotropy was largely overcome by the use of a spherical moderator of polyethylene 20–30 cm in diameter, 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 BF_3 proportional counters. The main characteristic of all these instruments is an over-response to intermediate energy neutrons.

III.29. Another instrument [205] uses two moderating spheres (107 mm and 64 mm in diameter) in a single case. It weighs 3 kg and covers a dose equivalent rate in the range of 30 μSv/h to 100 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.30. Ionization chambers were first developed to measure exposure to X rays and gamma radiation. If hydrogen is introduced into the walls and the gas, ionization chambers can be made more sensitive to neutron radiation. However, they are also sensitive to photons, and so it is necessary to provide a second chamber that is relatively insensitive to neutrons (e.g. with graphite walls and a carbon dioxide gas mixture, or aluminium walls and argon gas) to correct for the gamma radiation that is always associated with neutron radiation. Such ionization chambers measure the neutron absorbed dose, not the dose equivalent rate.

III.31. Since the response of an ionization chamber to gamma radiation per unit dose is similar to the response to neutron radiation, it is not possible to discriminate efficiently between the two types of radiation with a single chamber. However, ionization chambers can be used where it is not necessary to discriminate between the contributions from gamma radiation and from neutron radiation in a radiation field. It is possible to make the wall of the ionization chamber in near tissue equivalent material, and to fill the detector with near tissue equivalent gas. The energy deposition in the detector then mimics the energy deposition in tissue, regardless of the type of radiation. These ionization chambers are operated in proportional counter mode. Such tissue equivalent proportional counters are mostly used at low gas pressure, and can thus be used for microdosimetric purposes, but they are also useful as ambient monitors.

Other neutron instruments

Recoil proton proportional counters

III.32. 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 calculations of the energy and range relationship, 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 from about 10 keV to 1.5 MeV.

Rossi proportional counters

III.33. Tissue equivalent proportional counters can be used to measure, in addition to dose, the linear energy transfer of the deposited energy. The linear energy transfer (LET) can then be used to determine the mean radiation quality factor Q using the Q–LET relationship established by the ICRP (see para. 2.34 and Ref. [3]). This can then be incorporated into the electronics of the instrument. Thus, dose can be converted to personal dose equivalent. These instruments can also be used for measurements in mixed radiation fields.

Scintillation detectors

III.34. 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 in the mathematical analysis. These limitations restrict the energy range of the detector to about 0.2–20 MeV.

Semiconductor detectors

III.35. Semiconductor detectors are normally based on silicon or 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 a thousand times that of the gas in a chamber.

Appendix IV

BIOKINETIC MODELS FOR ASSESSMENT OF INTERNAL EXPOSURE

IV.1. Intakes of radionuclides can occur via various pathways, namely inhalation, ingestion, injection and dermal absorption (through intact skin or via a wound). For occupational exposure, the main route of intake is by inhalation. However, a small fraction of material deposited in the respiratory tract is transferred to the throat by ciliary action and is swallowed, giving the opportunity for absorption in the gastrointestinal tract. A fraction of the ingested radionuclides is absorbed in the blood. Intakes can also occur by direct ingestion or, for some radionuclides, by absorption through intact skin. Damage to the skin in the form of cuts or other wounds can also result in intakes of radionuclides. Routes of intake of radionuclides into the body, subsequent transfers within the body and excretion from the body are shown schematically in Fig. 6.

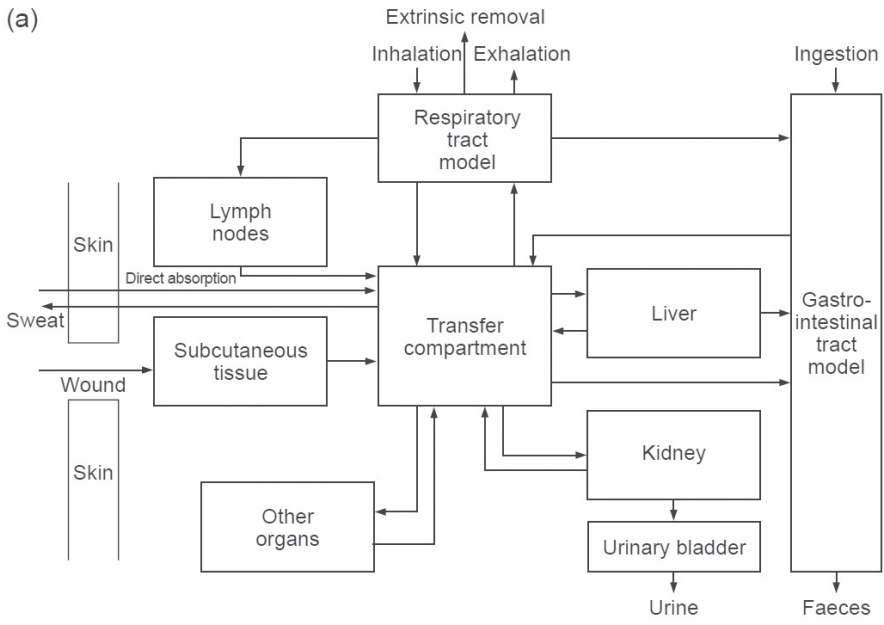

FIG. 6. Routes of intake of radionuclides into the body, transfers within the body and excretion from the body [13].

IV.2. Intake, uptake, internal transfer and excretion of radionuclides can be described by means of compartmental models. The ICRP has developed specific models for workers who are subject to occupational exposure.

IV.3. Biokinetic models of the alimentary tract and respiratory tract are used to define the entry of radionuclides into the body and their movement within the body, resulting in absorption to blood and/or removal of radionuclides from the body. The behaviour of radionuclides absorbed to blood is described by means of element specific systemic models that range in complexity.

IV.4. The models are intended both for the derivation of dose coefficients and for the interpretation of bioassay data, and they can be applied for regulatory control of the workplace. A general overview of the models used in the generation of dose coefficients for intakes of radionuclides is given below. Further details and information can be found in the references provided.

MODELS FOR DIFFERENT ROUTES OF ENTRY

Inhalation

IV.5. The behaviour of radionuclides inhaled by workers is described in Ref. [133] in a human respiratory tract model. Guidance on the use of the human respiratory tract model can be found in Ref. [206].

IV.6. The human respiratory tract model treats deposition and clearance of inhaled radionuclides separately. It calculates doses to specific tissues of the respiratory tract and takes account of differences in the radiosensitivity of tissues.

IV.7. The human respiratory tract is represented as two tissues: the extrathoracic airways and the thoracic airways. The extrathoracic airways are divided into two regions, one corresponding to the anterior nasal passage and the other corresponding to the posterior nasal passage, the pharynx and the larynx. The thoracic regions are bronchial, bronchiolar and alveolar–interstitial, the gas exchange region. Lymphatic tissue is associated with both the extrathoracic airways and the thoracic airways. Reference values of dimensions and scaling factors are specified in the model.

IV.8. Deposition of inhaled particulates is calculated for each region of the respiratory tract, with account taken of both inhalation and exhalation. This is done as a function of particle size, breathing parameters or workload, and is

assumed to be independent of chemical form. Age dependent default deposition parameters are given for a range of particle sizes from an activity median aerodynamic diameter (AMAD) of 0.6 nm to 100 μm.

IV.9. For inhalation of radionuclides by workers, the reference subjects are taken to be normal nose breathing persons undertaking light work. For simplicity, deposition in, and clearance from, the respiratory tract are calculated for the reference adult male only. An AMAD of 5 μm is considered to be the most appropriate default particle size for radionuclides in the workplace [135], whereas an AMAD of 1 μm is used as a default for members of the public.

IV.10. Clearance from the respiratory tract is treated as two competing processes: particle transport (by mucociliary clearance or translocation to lymph nodes); and absorption to blood. Most of the deposited material that is not absorbed to blood is cleared to the gastrointestinal tract by particle transport. The small amounts transferred to lymph nodes continue to be absorbed into body fluids at the same rate as in the respiratory tract.

IV.11. The human respiratory tract model assigns gases and vapours to three default solubility and reactivity classes, on the basis of the initial pattern of deposition in the respiratory tract. Subsequent retention in the respiratory tract and absorption to body fluids are determined by the chemical properties of the gas or vapour.

IV.12. The human respiratory tract model has been used to calculate the dose coefficients for inhalation of radionuclides by workers that were presented in Ref. [130] and table III.2A of GSR Part 3 [2], and also to calculate the bioassay functions presented in Ref. [13].

IV.13. The ICRP has recently developed a revised version of the human respiratory tract model [16]. The revised version has some simplifications and modifications both to the structure of the model and to the values of its parameters, but the basic features of the model remain unchanged. The modifications are based mainly on experience gained in the application of the model and on new evaluations of the available sets of experimental data. New dose coefficients and bioassay functions for workers, based on this updated model, are given in Ref. [16].

Ingestion

IV.14. The behaviour of radionuclides ingested by workers is described in Ref. [207] in a model based on four gastrointestinal tract compartments representing the stomach, the small intestine, the upper large intestine and the lower large intestine. The mean residence times in the four compartments are 1, 4, 13 and 24 h, respectively. The uptake to blood takes place from the small intestine and is specified by fractional uptake (f_1) values.

IV.15. This model forms the calculation basis for the dose coefficients for ingestion of radionuclides by workers presented in Ref. [130] and table III.2A of GSR Part 3 [2], and also for the interpretation of bioassay data in Ref. [13].

IV.16. A new model of the behaviour of ingested radionuclides, the human alimentary tract model, has been developed and is described in Ref. [208]. This model includes a larger number of regions, namely the oral cavity, oesophagus, stomach, small intestine, right colon, left colon and rectosigmoid, and allows for absorption of an element and its radioisotopes to blood from several sections of the tract. The total fractional uptake of ingested radionuclides is indicated by the symbol f_A. However, the general assumption, which is valid for nearly all radionuclides, is that absorption occurs exclusively in the small intestine (i.e. the value of f_A equals the fractional absorption of ingested radionuclides from the small intestine). In addition, the model structure allows for retention of ingested radionuclides in the mucosal tissues of the walls of alimentary tract regions and on teeth.

IV.17. New dose coefficients and recommendations for the interpretation of bioassay data, on the basis of this new human alimentary tract model, have recently been published by the ICRP [16].

Entry through wounds

IV.18. Although much of the radioactive material can be retained at the wound site, soluble material can be transferred to the blood and, hence, to other parts of the body. Insoluble material will be slowly translocated to regional lymphatic tissue, where it will gradually dissolve and eventually enter the blood. A variable fraction of insoluble material can be retained at the wound site or in lymphatic tissue for the remainder of the individual's life. If particulate material enters the blood directly, it deposits principally in phagocytic cells in the liver, spleen and bone marrow.

IV.19. For insoluble radionuclides retained at a wound site, the most exposed tissues will be those around the wound. Consideration should be given, in consultation with the occupational physician, to the excision of contaminated local tissues. For this, the variation with depth of contamination at the wound site should be accurately determined. The absorbed dose at the wound site and in the regional lymph nodes can be assessed from the activity remaining after excision, the characteristics of the radionuclides involved, the mass of tissue irradiated and the time since exposure. If the materials are soluble, they can translocate from the wound site to the blood at a rate that depends on their solubility. The distribution of this soluble component will, in most instances, be similar to that of material entering the blood from the lungs or from the gastrointestinal tract, but there may be exceptions for some chemical forms of radionuclides that enter the blood directly.

IV.20. Dose coefficients for incorporation of radionuclides through wounds have been calculated for 38 radionuclides [209] using a wound model [210] combined with systemic models used to calculate dose coefficients for workers [130].

Entry through intact skin

IV.21. Certain radioactive substances, such as tritium labelled compounds, organic carbon compounds and compounds of iodine, can penetrate intact skin. A fraction of the activity will enter the blood, but there is no general model for the assessment of doses, and specific models have to be developed [211]. For example, the behaviour of tritiated organic compounds following direct absorption through the skin will be significantly different from their behaviour after inhalation or ingestion. For skin contamination, both the equivalent dose to the area of skin contaminated and the effective dose should be considered.

MODELS FOR SYSTEMIC RADIONUCLIDES

IV.22. A systemic biokinetic model describes the time dependent distribution and retention of a radionuclide in the body after it reaches the systemic circulation, and its excretion from the body. The systemic biokinetic models used in Ref. [207] had a relatively simple structure that included the passage of material from the systemic circulation to selected tissues and organs, and then directly to excretion. In Refs [212–215], physiologically based, age specific models were developed for selected radionuclides. These models included the possibility of

recycling of the deposited radionuclides and a more realistic description of the excretion pathways.

IV.23. These systemic biokinetic models, together with the human respiratory tract model, form the calculation basis for the dose coefficients for ingestion of radionuclides by workers that are presented in Ref. [130] and table III.2A of GSR Part 3 [2], and for the interpretation of bioassay data in Ref. [13].

IV.24. Further development of the systemic biokinetic models has since been carried out [16], leading to the definition of model structures with an increased physiological realism compared with those described in previous publications of the ICRP. The physiologically descriptive modelling scheme has been applied more broadly and, in some cases, the model structure has been slightly modified. In addition, the approach to the modelling of radioactive progeny has been revised. The general assumption until now has been that the progeny follow the same biokinetic behaviour as that of the parent, except in the case of progeny that are isotopes of lead, radium or thorium, for iodine progeny of tellurium and for noble gas isotopes arising in various decay series. In the revised models, separate systemic biokinetics have been applied to the parent and its progeny. These revised systemic biokinetic models have been used in the development of revised dose coefficients and recommendations for the interpretation of bioassay data, and have recently been published by the ICRP [16].

Appendix V

METHODS FOR INDIVIDUAL MONITORING OF INTERNAL CONTAMINATION

DIRECT METHODS

V.1. Direct measurement of the distribution and total body content of one or more incorporated radionuclides is possible when the radionuclide emits penetrating radiation (normally X rays or gamma radiation and, in special cases, bremsstrahlung) of sufficient energy and yield to be detectable outside the body. For most in vivo counting applications, photon detectors are positioned at specified locations around the body, usually with at least partial shielding of the detector or the subject to reduce interference from ambient external sources. Low level whole body counters are located in shielded counting chambers.

V.2. Generally, interpretation of direct measurements in terms of intake and assessment of committed effective dose relies on biokinetic modelling of the distribution and retention of the incorporated radionuclides, and on biophysical modelling of energy deposition. Both of these aspects can vary markedly over time and between individuals.

Measurement geometries

V.3. A variety of physical arrangements of detectors have been developed to serve specific purposes. For radionuclides that are distributed throughout the body, counting of the whole body, or of a large fraction of the body, provides the greatest sensitivity. Whole body counting is carried out either using a static geometry, with one or more detectors, or by scanning — moving the subject with respect to static detectors or moving detectors around a static subject. Static geometries commonly comprise an array of detectors distributed along a standing or supine subject, or a single detector directed towards the centre of a subject on a tilted chair or curved frame. Some examples of counting geometries are shown in Fig. 7.

V.4. For other radionuclides that are at least temporarily concentrated in particular tissues or organs of the body, the monitoring of specific sites should be conducted. Examples are radioiodine, which is taken up by the thyroid, and inhaled radioactive particles, which are retained in the lungs. For bone seeking radionuclides that emit low energy photons, such as ^{241}Am and isotopes of

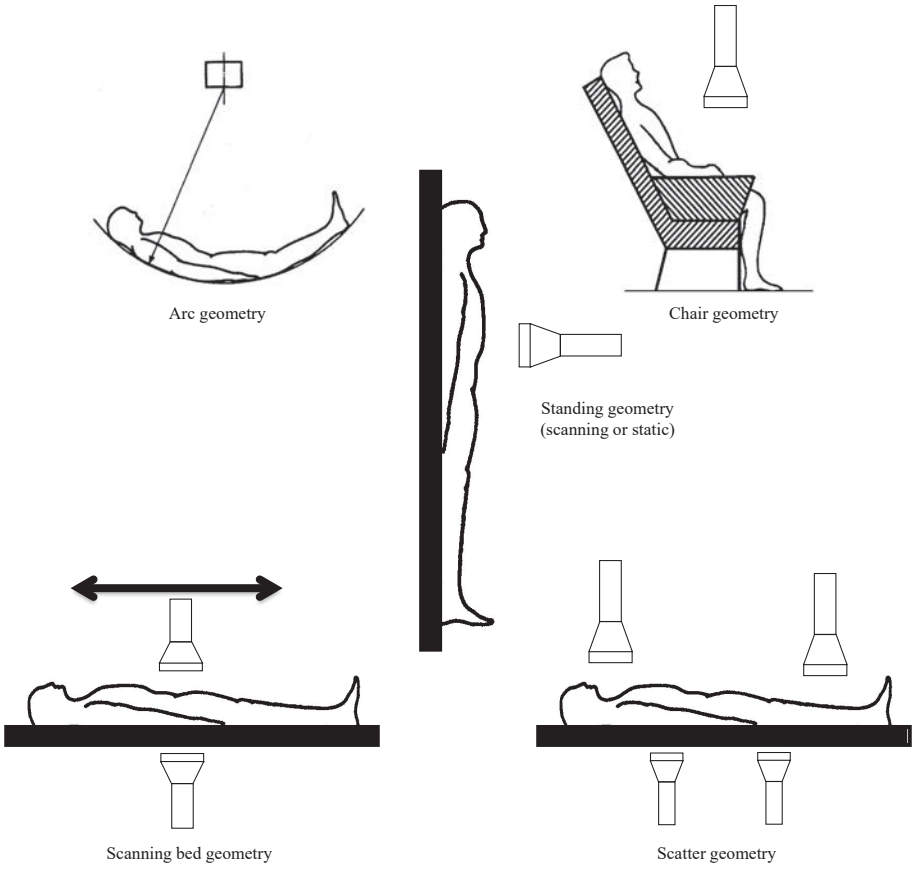

FIG. 7. Various geometries used for whole body counting.

plutonium and uranium, measurements should be conducted on bones surrounded by a thin layer of tissues, such as the knee or the skull [216, 217].

V.5. Localized monitoring should also be conducted when intake is through a wound or when there are other reasons for determining the distribution of the radionuclides within the body. Whole body counting is unlikely to fail completely to detect a significant amount of localized activity, but might not provide an accurate estimate of the amount or give good information on its spatial distribution. The applications of phantoms and their limitations are described in Ref. [218].

V.6. In all cases, the method should be to compare the signal measured from the subject with that obtained under the same conditions from an anthropomorphic phantom, or other surrogate, containing known quantities of the radionuclide in question. The distribution of the radionuclide in the calibration phantom should match that expected in the human subject as far as possible, although some measurement techniques are more sensitive than others to this distribution. In vivo monitoring systems may also be characterized and calibrated by means of Monte Carlo techniques that have been specifically developed for this purpose [219, 220].

Methods of detection

V.7. Various detection systems are used for different purposes. Inorganic crystals of high atomic number materials, usually thallium activated sodium iodide (NaI(Tl)), are commonly used to detect energetic photons (above 100 keV), such as those emitted by many fission and activation products. Scintillations produced by the crystal's interaction with high energy photons are detected by photomultiplier tubes. These generate electronic pulses that are processed to produce a spectrum reflecting that of the radiation absorbed by the crystal.

V.8. This type of measurement system provides the most sensitive method of quantifying radioactive content in the body. However, the energy resolution of the detectors is limited, so that even deconvolution techniques might be unable to determine the radionuclides giving rise to a complex spectrum, such as that from a fresh fission product mixture, or in the presence of a varying background such as that due to radon and its progeny.

V.9. Semiconductor detectors have major advantages in energy resolution and so allow almost unambiguous identification of the radionuclides in a mixture, but they are inconvenient in that they need cooling to liquid nitrogen temperatures. High purity germanium detectors can tolerate cycles to room temperature and need cooling only during operation. Electrically cooled cryostats or mechanical coolers allow the operation of germanium detectors without the need for liquid nitrogen. The lower efficiency of these detectors, in comparison to that of inorganic crystals and other scintillators, is more than compensated for by the lower background signal and improved energy resolution.

V.10. Low energy photons, such as those emitted by ^{239}Pu (13–20 keV) and by ^{241}Am (60 keV), can be detected with thin NaI(Tl) crystals, which have a similar detection efficiency to larger crystals but a much lower background. The addition of a second crystal, usually of thallium activated caesium iodide (CsI(Tl)), as

an anticoincidence guard improves the detection sensitivity by eliminating the contribution of high energy photons. Such a device, which is commonly known as a 'phoswich' (phosphor sandwich) detector, can lower the detection limit for these photons by more than an order of magnitude. Multiple high purity germanium planar detectors are increasingly used for the detection of low energy photons because of their high resolution and low background. For low energy photon counting (using, for example, phoswich or high purity germanium detectors), account should be taken of the thickness of the overlying tissue in determining the detection efficiency.

V.11. Miniature semiconductor detectors, in particular those using cadmium telluride (CdTe) operating at room temperatures, are becoming increasingly available. CdTe detectors offer high sensitivity for detection of low energy photons. Their small size (approximately 10 mm in diameter and 2 mm thick) make them ideal for monitoring localized wounds. Their additional advantages are that there is no need to confine a worker in a shielded enclosure and that quick assessment of the success of a surgical excision procedure is possible. However, these small detectors are not suitable for the identification and quantification of radionuclides by spectrometry.

V.12. In setting up an advanced in vivo monitoring facility, a variety of detection systems, appropriate for the specific radionuclides likely to be of concern, should generally be installed.

Measurement procedures

V.13. Subjects for direct measurements should be free of external surface contamination and in fresh clothing, such as disposable paper garments. Personal belongings, such as jewellery, watches and spectacles, should be removed. Such precautions help to avoid false identifications of internal activity, and also help to prevent the transfer of contamination to the counting equipment. Individuals should, to the extent practicable, be in a defined counting position to ensure reproducibility in serial measurements and to improve comparison with calibration results. In some cases, the subject will need to remain stationary for periods of up to an hour for satisfactory precision in the measurement. Some means of communication should be provided for subjects in enclosed shielding, particularly when extended counting periods are necessary.

V.14. Background counts arising in the detectors are normally attributed to four sources:

(a) Ambient background radiation from natural sources, such as cosmic radiation or radon and its decay products;
(b) Background radiation from activity in the shielding and other equipment;
(c) Radiation from natural radioactivity in the subject;
(d) Radiation scattered into the detector by interactions of the subject with ambient radiation.

V.15. For counting systems based on scintillation counting (NaI(Tl) crystals or phoswich detectors), background counts for the detector system should be determined using an appropriate phantom that is as similar as possible to the subject to be counted and is placed in a defined counting position. The background level can be considerably reduced by proper design and adequate shielding of the enclosure (i.e. a steel room facility) in which the subject is counted for internal contamination. For whole body counting, background counts determined using uncontaminated subjects, matched with respect to gender, height and weight, will improve the results. However, exact matching will not be possible and factors such as ^{40}K content cannot be controlled. Better results can therefore be obtained from matched control groups or from measurements on the specific individual made before starting work. Measurements of background in the counter should be made as close as possible in time to the measurement of the subject, ideally just before and just after the measurement. When using semiconductor detectors, background counting with matching phantoms is not necessary.

INDIRECT METHODS

V.16. Indirect monitoring is based on the determination of activity concentrations in biological materials separated from the body — usually urine, faeces, breath or blood — or in physical samples taken from the work environment, such as samples of air or samples of contamination from surfaces.

V.17. Indirect methods should be used for those radionuclides that do not emit strongly penetrating radiation to any significant extent. For some other radionuclides, such as those that emit only low energy photons or that are preferentially eliminated in excretions, the insensitivity of, and uncertainties in, the direct monitoring measurement may be such that an indirect method can provide a more reliable estimate of intake. In other cases, indirect methods may be more practicable than direct monitoring and may be sufficiently accurate.

V.18. Information on the most suitable bioassay measurement techniques for all radionuclides of common interest in occupational exposure is given in Ref. [13]. This information has recently been updated [16].

Biological samples

V.19. The biological samples most commonly used for the estimation of intakes are urine and faeces, but breath, blood and other samples are used in special cases. For example, the analysis of activity in a nose blow or nasal swab provides an early estimate of the identities and relative levels of radionuclides in an inhaled mixture. In this case, however, the relationship between the activity concentration in the sample and the intake is so uncertain that such data can provide only a crude indication of the size of the intake.

V.20. The choice of bioassay sample will depend not only on the major route of excretion, as determined from the physicochemical form of the intake and the biokinetic model for the elements involved, but also on such factors as ease of collection, analysis and interpretation. Urine samples are relatively easy to obtain and analyse. They generally provide information on the intake of radionuclides in chemical forms that are readily transferred to the blood. In contrast, intakes of insoluble material are usually assessed from faecal samples.

Urine

V.21. Following the entry of radionuclides into the blood and systemic circulation, clearance from the body will generally be via the urine. Urine contains waste and other materials, including water, that is filtered from the blood by the kidneys and collected for up to several hours or more in the bladder before voiding. Because of this mixing in the bladder, radionuclide levels in samples of urine obtained soon after an acute intake should be interpreted with caution. The bladder should be cleared soon after the intake, and then a second sample and subsequent samples obtained. All samples should be analysed.

V.22. After the first few days, 24 h samples of urine normally provide the best basis for assessing intake. In circumstances in which 24 h samples have not been obtained, the total excretion can be estimated by means of normalization relative to creatinine content, collection time (i.e. length of actual sampling interval), volume and specific gravity [221, 222]. Reference [223] reports that methods based on creatinine and normalization of specific gravity do not provide improved confidence over normalization by time or volume, and require additional measurements (and costs) for the laboratories involved.

V.23. In routine monitoring for radionuclides with prompt components of excretion, consideration should be given to the day on which samples are taken, since there can be significant differences between samples taken before and after even short periods free from exposure.

V.24. For intakes of tritiated water, the concentration of tritium in urine is the same as in body water, and it can be used to assess body content and dose rate without reference to an excretion model. A dose assessment method for intakes of tritiated water is provided in annex VI of Ref. [224].

Faeces

V.25. Faecal samples contain water, cellular debris lost from the wall of the gastrointestinal tract, unabsorbed waste products transported through the gastrointestinal tract, including insoluble materials cleared from the lung, and metabolic products cleared from the liver in bile. The mass and composition of individual faecal voidings can be quite variable and depend strongly on diet. For this reason, reliable estimates of daily faecal excretion rates of radionuclides can usually be based only on total collections over 3–4 d. Single samples should, in most cases, only be used for screening purposes.

V.26. In the monitoring of workers chronically exposed to long lived radionuclides, faecal samples should ideally be collected after a vacation (at least a 10 d absence from work) and prior to returning to the working environment. Such post-vacation measurements allow for differentiation between the fraction of inhaled radionuclides cleared rapidly through the gastrointestinal tract and the delayed clearance of systemic activity and long term deposits of insoluble forms of radionuclides in the lung.

Breath

V.27. Breath is a significant route of excretion only for those few materials that are exhaled directly or metabolized to gases or volatile liquids. However, for these cases, breath samples can provide a convenient way of measuring the activity of excretions, free from most other sources of contamination.

V.28. The measurement of ^{220}Rn in breath has been used in various States to determine thorium intakes by workers involved in the mining and processing of thorium containing minerals [225–230]. The ^{220}Rn contained in the exhaled breath is used as a measure of the ^{224}Ra, and hence ^{232}Th, present in the lung. The exhaled ^{220}Rn activity is expressed as the activity of the freely emanating

^{224}Ra parent that would support the ^{220}Rn concentration measured at the subject's mouth. The method provides a relatively inexpensive and portable means of detecting moderate levels of inhaled thorium in the body. Two basic methods for measuring ^{220}Rn in breath are reported:

(a) The first method, as described for example in Refs [225, 229], is based on the 'double filter' system. Air from the lung is exhaled into a cylinder fitted with filters at both ends. The exhaled ^{220}Rn decays during its transit and the progeny are collected on the exit filter. After a delay of 5 h to allow the progeny to decay, the alpha activity on the filter is measured by alpha counting.
(b) The second method, as described for example in Refs [227, 231], is based on electrostatic collection of the ^{220}Rn progeny, ^{212}Pb, 85–88% of which is positively charged, onto a negatively charged Mylar disc. After the collection period, the alpha decays can be measured by low level alpha spectrometry [228].

V.29. One disadvantage of the ^{220}Rn in breath technique is that the measurements have to be taken after a break from any work involving exposure to thorium following the intake, to take account of the clearance of activity in the upper airways and the possible presence of short lived ^{220}Rn progeny. The break period should be at least 12 h, but preferably 72 h, to allow for seven half-lives of ^{212}Pb.

V.30. Another more serious disadvantage is that the measurements require knowledge of the relationship between exhaled ^{220}Rn, expressed as the emanating ^{224}Ra equivalent activity at the mouth, and the lung burden of thorium. This relationship, referred to as the ^{220}Rn emanation rate, appears to depend on the nature of the thorium contamination. The breath measurement should be calibrated against in vivo measurements of thorium lung burden [232]. The calibration procedure requires there to be workers with thorium lung burdens that are high enough to be detected by the in vivo gamma counting technique. Estimates of ^{220}Rn emanation rate vary widely, from 3.7% to 20% [225, 229, 230, 233–237]. Because of this wide variation and the associated uncertainty, the use of the ^{220}Rn in breath technique is of limited value for routine dose assessment.

Blood

V.31. Blood samples provide the most direct source for estimating radionuclides present in the systemic circulation, but are not often used because of medical constraints on the sampling process. Investigations of the concentration of thorium in the blood of heavy mineral sands workers in Western Australia and

thorium plant workers in India are reported in Refs [238, 239], respectively. However, with only a few exceptions (e.g. in the detection of dilute tritiated water, ^{59}Fe and ^{51}Cr in labelled erythrocytes), blood samples provide very limited information on the total systemic activity following an intake. This is because of rapid clearance from the blood stream and deposition in tissues and organs.

Nose blows

V.32. Nose blows and nose swabs should not be used to estimate an intake, but they can be useful in task related and special monitoring to identify the components in a mixture of radionuclides. Nose blows and nose swabs can also be used to indicate the need for additional sampling and analysis, especially when exposure due to alpha emitters, such as actinides, may have occurred.

Tissue samples

V.33. For localized deposits of radionuclides with high radiotoxicity (e.g. transuranic elements) in a wound, it is usually advisable, subject to medical advice, to excise the contamination soon after the intake. Radiochemical analysis of excised tissue by destructive and non-destructive methods can provide information on the radionuclides and their relative concentrations, and may assist in assessing the uptake to blood and in determining the course of further actions.

V.34. Other biological samples, such as hair and teeth, can be used to assess intakes, although, in general, they cannot be used for quantitative dose assessments.

Air samples

V.35. For compounds that disperse readily in air, such as radioactive gases and vapours (e.g. $^{14}CO_2$ and tritiated water), samples from stationary air samplers can provide a reasonable representation of inhaled radionuclides, especially in small rooms. Stationary air samplers can be deployed at fixed locations in the workplace and have relatively high sampling rates, typically about 20 L/min. For other sources, however, such as resuspended particulates, such samples may lead to estimates of the activity of the material inhaled that are wrong by an order of magnitude or more, depending on the relative locations of the source, the sampler and the worker.

V.36. More representative samples are obtainable from personal air samplers, which are battery powered systems carried by the worker that draw air samples

from the immediate breathing zone at a relatively low sampling rate, typically 2 L/min. Even these samples may, however, lead to an overestimation or underestimation of intakes, depending on the assumptions made about the particle size of the aerosol and breathing rates.

V.37. Both forms of sampling rely on the collection of radionuclides from the passing air on a filter medium. To some extent, this medium will be specific to the material to be collected. For example, particulate material can be captured on coarse fibre filters, while activated charcoal beds are used to sample radon gas and iodine vapour. Tritiated water can be collected in a water trap.

Airborne dust

V.38. The sampling efficiency of an air sampler should be taken into account in the assessment of internal exposure. Air samplers are designed to follow a sampling convention for a specific particle size that is based on sampling criteria for industrial hygiene and relates to the fraction of the total airborne particles sampled. In terms of this sampling convention, there are three dust fractions that can be sampled:

(a) The inhalable dust fraction is the fraction of total airborne particles that enters the body through the nose or the mouth during breathing. It includes particles with aerodynamic diameters less than about 100 μm.
(b) The thoracic dust fraction is the subfraction of the inhalable fraction that can penetrate into the tracheo-alveolar region of the lung. It includes particles with aerodynamic diameters less than about 30 μm.
(c) The respirable dust fraction is the subfraction of the inhalable fraction that penetrates into the alveolar region of the lung, including the respiratory bronchioles and the alveolar ducts and sacs. It includes particles with aerodynamic diameters less than about 10 μm.

V.39. In workplaces involving exposure due to ^{238}U and ^{232}Th series radionuclides in airborne dust, the following considerations apply to air sampling equipment and techniques:

(a) Air samplers typically underestimate the airborne activity concentration and thus the activity inhaled. The degree of underestimation depends on the AMAD and geometric standard deviation of the ambient aerosol, on the dust load in the air and on the type of sampler used [103]. A correction factor can be applied to minimize the degree of underestimation. For an AMAD of 5 μm and a geometric standard deviation of 2.5 (the default

values recommended in Ref. [133] for workplaces in which the actual values are unknown), this correction factor is 1.18 for inhalable samplers, 1.41 for thoracic samplers and 2.5 for respirable samplers [103]. The use of the appropriate correction factor does not remove all of the uncertainty, however. This is because the AMAD and geometric standard deviation vary with the location, time and circumstances of dust production, and can therefore never be precisely known.

(b) The size distribution of aerosol particles also has a significant effect on the dose coefficient, leading to an additional source of uncertainty when assessing the effective dose due to the inhalation of particles. The dependence of the dose coefficient on the AMAD is particularly strong for particles of lung absorption type S. When assessing the effective dose, a sampler should be selected with a sampling efficiency that follows as closely as possible the dependency on the AMAD of the relevant dose coefficients [103].

(c) Knowledge of the lung absorption type should be used in determining not only the most appropriate dose coefficient but also the type of sampler that best minimizes the possible bias arising from an incomplete knowledge of the particle size distribution [103].

(d) The preferred type of sampling for minimizing possible bias in the assessment of effective dose is inhalable sampling for particles of lung absorption type F and thoracic sampling for particles of lung absorption types M and S [103]. Particles of lung absorption types M or S are likely to be encountered in many naturally occurring radioactive material industries. However, thoracic samplers are, at present, not as widely available as inhalable samplers, and often they are not suitable for alpha counting owing to the dust particles being collected on foam rather than on flat filters.

(e) The alpha activity of material inhaled by workers may be underestimated if there is significant alpha particle self-absorption in large particles or in multilayers or agglomerates of smaller particles deposited on the filter. Dust loadings on filters may, in such cases, need to be restricted accordingly. Various types of filter medium and sampling cassette are available. Where the dust concentration is relatively low (about 1–2 mg/m^3) and sampling is undertaken over a 4–6 h period, the choice of filter medium and cassette is not likely to be critical. However, when the dust concentration is relatively high (more than about 3 mg/m^3) and the sampling is undertaken for a period of 8 h or more, the selection of equipment requires more careful consideration. For some types of filter medium, such as polyvinyl chloride, part of the sample can be lost as a result of dust not fully adhering to the surface. For some types of monitoring cassette, the dust can adhere to the

inside wall, requiring it to be removed by washing and added to the material collected on the filter prior to analysis [25].

(f) For alpha emitting radionuclides, a delay between sample collection and counting may be necessary to enable the decay of short lived ^{222}Rn and ^{220}Rn progeny that would contribute to sample counts.

Radon

V.40. Personal monitoring devices for radon and its progeny are of either the passive or the active type. Passive devices take the form of solid state nuclear track detectors that are worn by a worker for an appropriate time period. After exposure, the track detectors are processed by chemical or electrochemical etching. The etching procedure reveals the nuclear tracks caused by the alpha particles from decay of ^{222}Rn. The density of the tracks is proportional to the cumulative exposure due to ^{222}Rn over the deployment period. Active devices involve the drawing of air through a sampling filter by a battery powered pump. The alpha emissions from the ^{222}Rn progeny deposited on the filter are recorded by:

(a) A detector disc of a thermoluminescent dosimeter, which provides information on gross alpha activity;
(b) A silicon solid state detector with associated electronics, which again provides information on gross alpha activity or provides nuclide specific information;
(c) A solid state nuclear track detector, which provides information on individual ^{222}Rn progeny.

V.41. For workplace monitoring of ^{222}Rn in air, the concentration is determined either as an instantaneous measurement based on a single air sample (known as a grab sample) or as a time integrated measurement. Instantaneous measurements have traditionally been made using an alpha scintillation cell (commonly referred to as a Lucas cell). In this method, a sample of the air is collected in a detector chamber. The inside surface of the chamber has a scintillation coating comprising a layer of silver activated zinc sulphide. The air sample is filtered to remove the ^{222}Rn progeny, leaving only the parent radionuclide ^{222}Rn inside the chamber. As the ^{222}Rn and ingrowing progeny decay by emitting alpha particles, the scintillations from the alpha decay are counted at a known equilibrium by a photomultiplier mounted on top of the chamber.

V.42. Other techniques are available for instantaneous measurement of ^{222}Rn. These include the pulse counting ionization chamber technique and the double

filter sampler technique, the latter of which can be used for measuring both ^{222}Rn and ^{220}Rn. Air is passed through a chamber after removal of ^{222}Rn progeny and ^{220}Rn progeny by an inlet filter. The decay of ^{222}Rn and ^{220}Rn during passage through the chamber generates progeny, which are collected on an outlet filter. The alpha emissions from the progeny on the outlet filter are counted. The results of this are used to back-calculate the ^{222}Rn and ^{220}Rn concentrations. Time integrated measurements can be made by using nuclear track detectors known as radon cups, by using thermoluminescent dosimeters, or by using devices known as electret passive environmental radon monitors.

V.43. So called 'continuous' monitoring techniques are available. They do not provide truly continuous measurements, but are either based on frequent instantaneous sampling, by using adaptations of the instantaneous sampling methods described above, or are based on other specific techniques. Active pumping or diffusion of radon gas into the sensitive volume of a high voltage chamber allow deposition of ingrowing positively charged radon progeny on the surface of a silicon surface barrier detector for subsequent alpha spectroscopy. This method allows separation of ^{222}Rn and ^{220}Rn. Portable instruments are available that are relatively rugged and lightweight. They have been used quite extensively in mining environments, including underground mines. Portable instruments can be equipped with alarms that are triggered when a specified ^{222}Rn concentration is exceeded.

V.44. Workplace monitoring of the short lived progeny of ^{222}Rn is carried out by drawing air through a filter to capture the progeny radionuclides. Owing to the short half-lives of the ^{222}Rn progeny, counting of the alpha or beta activity on the filter should be performed during, or shortly after, sampling.

V.45. As with the monitoring of ^{222}Rn gas concentrations, the monitoring of ^{222}Rn progeny can be carried out either by instantaneous measurements or by measurements over a given time period. Through the development of automated sampling and analytical techniques, instruments have become available for semi-continuous monitoring using integrated measurements and for continuous monitoring. In some instruments that perform alpha or beta spectroscopy, raw data can be stored on a continuous basis within the instrument and downloaded later for processing to determine the individual radionuclide concentrations over time.

V.46. The instruments and counting methods used for measuring the concentrations of ^{222}Rn and its progeny can, in principle at least, be adapted for measuring the concentrations of ^{220}Rn and its progeny, with certain limitations.

Some continuous monitoring instruments can measure ^{220}Rn and its progeny. For personal monitoring, integrating nuclear track detectors can be used. One type of personal alpha dosimeter records alpha emissions from ^{212}Po separately, allowing the direct measurement of ^{220}Rn progeny.

Surface samples

V.47. Because modelling of the transfer of radionuclides from surfaces into the body is particularly uncertain, samples of radionuclide concentrations on surfaces are used primarily to indicate the potential for significant intakes and the need for individual monitoring. Such samples can also indicate the relative amounts of various radionuclides in a mixture and the presence of any radionuclides not detected in a bioassay sample.

V.48. Surface samples are usually obtained by wiping a specified area of the surface with materials such as filter papers or cotton swabs. These materials are chosen for their ability to transfer the expected contaminants from the surface for analysis. The efficiency of collection should be determined for the particular combination of surface and wiping material, but is typically assumed to be around 10% for a moist swab on a moderately porous surface.

Handling of samples

V.49. Special care should be taken in the handling of samples to be used for the assessment of internal exposure, firstly, to avoid the transfer of radioactive or biological contamination in handling and, secondly, to ensure a traceable link between the analytical result and the original sample, as required by the quality assurance programme.

V.50. With respect to the potential hazard from contamination, both radioactive and biological contaminants should be considered. Biological samples can contain pathogens such as bacteria and viruses. These pathogens will be potentially active until the complete sample has been turned into ash or otherwise sterilized. All such samples should therefore be stored at a low temperature, preferably frozen, until analysis. This treatment will also reduce unwanted biological degradation of certain materials, such as organically bound tritium, for which the molecular form is an important factor that should be considered in the subsequent analysis. Another way to prevent degradation is to treat the sample with acid.

V.51. To establish traceability, a chain of custody should be maintained such that at each step in the collection, transport and analysis of the samples, documentation is created to describe and verify the transfers that have occurred.

V.52. In order to ensure that the activity measured in the sample is representative of body clearance only, urine, faeces and other biological samples should not be collected in radioactively contaminated areas. The sample should be clearly marked to show the worker's identity and the date and time of sample collection.

V.53. Those responsible for decisions concerning the type(s) of analysis to be performed on the sample should be informed about the work areas in which the worker may have been exposed, especially if the sample is likely to have high levels of activity, as may be the case for special monitoring. Those responsible should also be aware of the use of any medication or treatment that could interfere with the analysis of the sample or its interpretation.

Methods of analysis

V.54. The analysis of biological or physical samples involves the detection and quantification of emissions from the radionuclides present by means of appropriate instrumentation. In many cases, the radionuclides first have to be separated from the sample matrix to allow sensitive and reproducible detection. In cases in which the detectors cannot discriminate between radionuclides that have similar emissions (e.g. some actinides), the samples should be subjected to chemical separation of the elements before counting.

Detection

V.55. Instrumentation used for radiometric assessment can be divided into three classes based on what is being measured: alpha particles, beta particles or photon emissions.

V.56. Alpha particles can be detected by various techniques, each having advantages and disadvantages. The simplest gross count of total alpha activity can be made using a zinc sulphide detector or a gas flow proportional counter. These methods are efficient, but they do not discriminate between alpha particles of different energies and they cannot identify or quantify individual radionuclides in a mixture. Radiochemical separation of individual radionuclides (see para. V.63) followed by alpha spectroscopy using silicon detectors can be used to quantify individual radionuclides, provided that their energies are sufficiently different. Long counting times are generally needed to achieve adequate sensitivity.

Radiometric analysis of individual radionuclides is unlikely to be cost effective for routine analysis of individual air sampling filters because it is time consuming and expensive. On a non-routine basis, however, filters can be retained, bulked over a longer period, and the activity can be determined by these more sensitive analytical techniques to obtain the aggregated intake of individual radionuclides over a longer period.

V.57. Industrial activities involving naturally occurring radioactive material can give rise to dust particles containing alpha emitting radionuclides in the ^{238}U decay series and/or the ^{232}Th decay series. The detection of this alpha activity on air sampling filters involves the following considerations:

(a) For naturally occurring radioactive material that has not been chemically or thermally processed, equilibrium of the ^{238}U decay series and the ^{232}Th decay series is unlikely to be significantly disturbed in freshly generated dust particles. Apart from any subsequent escape of ^{222}Rn or ^{220}Rn from the captured dust particles (see (c)), equilibrium conditions can generally be assumed when analysing air sampling filters by gross alpha counting.

(b) For naturally occurring radioactive material that has been subject to chemical or thermal processing, equilibrium conditions in the airborne dust particles cannot be assumed and the radionuclide composition should therefore be determined before analysing air sampling filters by gross alpha counting.

(c) It is possible that some ^{222}Rn or ^{220}Rn might escape from the captured dust particles between the time of sampling and the time of analysis. Investigations carried out for ore dust particles suggest that the loss of ^{222}Rn and ^{220}Rn is in the range of 0–50% [240]. Zero loss of ^{222}Rn or ^{220}Rn should be assumed for dust particles associated with minerals having very low ^{222}Rn or ^{220}Rn emanation coefficients, such as zircon and monazite. For dust particles associated with other minerals, such as uranium ore or uranium–thorium ore, some loss of ^{222}Rn or ^{220}Rn should be expected. For dust particles with the ^{238}U decay series and the ^{232}Th decay series in equilibrium at the time of sampling, the measured gross alpha activity should be multiplied by a correction factor in the range of 1–1.23 to account for the loss of ^{222}Rn or ^{220}Rn and their associated short lived progeny. For a typical loss of 25% ^{222}Rn or ^{220}Rn [241], a correction factor of about 1.10 should be used.

V.58. Beta particles are most commonly detected by liquid scintillation counting, especially for low energy beta emitters. In some cases, separation of two or more beta emitters in a mixture, such as tritium, ^{14}C and ^{32}P, can be achieved by setting

energy windows on the detector response. Gross measurements of high energy beta emitters deposited on planchettes or filters can be obtained by using gas flow proportional detectors. High energy beta particles can be detected by means of Cherenkov counting with a liquid scintillation spectrometer.

V.59. Alpha and/or beta spectroscopy is commonly used for determining concentrations of individual ^{222}Rn progeny on a filter. One alpha–beta spectroscopic technique uses a passivated implanted planar silicon detector. Various counting methods are employed, depending on the amount of information on individual ^{222}Rn progeny that is required. Counting can be performed just once (i.e. single count methods) or as a sequence of counts at specified intervals after sampling (i.e. two count and three count methods). By solving the relevant equations for radionuclide decay and in-growth, either the gross activity of the ^{222}Rn progeny or the activities of individual progeny can be determined.

V.60. Photon emissions from physical samples or biological samples are usually detected by means of conventional gamma spectrometry.

V.61. Non-radiometric techniques are also available. For example, luminescence techniques, such as ultraviolet fluorimetry and kinetic phosphorescence analysis, can be used for the assay of uranium, irrespective of the degree of enrichment. For bioassay measurements at low detection limits, inductively coupled plasma mass spectroscopy offers significant advantages in accuracy, speed and sample preparation for the determination of uranium and thorium in urine [242], as does thermal ionization mass spectrometry for ^{239}Pu [243]. Other techniques such as fission track analysis and neutron activation analysis can be used to measure specific radionuclides, but they are time consuming and expensive and they are necessary only in special circumstances.

V.62. Counting times for all of the methods in paras V.55–V.61 will depend on the activity in the sample, the measurement equipment used and the precision needed.

Radiochemical separation

V.63. In many cases, radionuclides should be separated from the sample matrix, or from radioisotopes of other elements, before counting, to enable the activity to be reliably quantified. This process is, to a large extent, specific to the elements being separated, but it generally includes sample preparation and pre-concentration, purification, source preparation and determination of yield. In general, various approaches can be taken to isolate a specific radionuclide from sources of interference in order to improve detection. An essential element

of the process is to trace the recovery of the radionuclide through each step, so that the final result can be used to reliably determine the concentration in the initial sample. Appropriate blank samples should be prepared for the purpose of measuring the background.

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Annex

TECHNIQUES FOR RETROSPECTIVE DOSIMETRY

A–1. This Annex is a shortened version of a 2011 review [A–1].

HAEMATOLOGICAL TECHNIQUES

A–2. A differential blood cell count is the first quantitative bioindicator that can be applied after radiation exposure. The assay is readily available, automated and inexpensive because it is a standard diagnostic tool for investigating many clinical conditions. Measurements take only a fraction of an hour for multiple samples.

A–3. For radiation exposures, the assay is quantified with respect to detecting acute exposures and whole body exposures (or nearly whole body exposures) that might lead to the haematological component of acute radiation syndrome.

A–4. Normal inter-individual and intra-individual variations in counts impose a background 'noise' such that it requires a dose of 1 Gy or higher before values depart from the normal ranges. The most informative early responses are the counts of lymphocytes and granulocytes. The platelet count is slower to respond because the lifespan of platelets in the circulating blood is longer.

A–5. Frequently repeated sampling is performed throughout the time course of clinical management, and the variation of the differential count with respect to the first sample, taken to be close to the pre-exposure background values, is plotted. The first blood sample should therefore be taken as soon as possible after exposure.

CYTOGENETIC TECHNIQUES

A–6. Cytogenetic damage in peripheral blood lymphocytes includes dicentric chromosomes, chromosomal aberrations, micronuclei and translocations. The dicentric chromosome assay, the premature chromosomal condensation technique and the micronucleus assay are best applied to the assessment of dose from more recent exposures, whereas fluorescence in situ hybridization (FISH) is the assay of choice to detect stable translocations for exposures that have taken place years or decades before, or that are chronic.

Dicentric chromosome assay

A–7. Dicentric chromosomes are almost exclusively induced by ionizing radiation. The spontaneous frequency of dicentrics is very low in the healthy general population (about 1 dicentric per 1000 cells). Dicentric frequencies in peripheral blood lymphocytes can show a clear linear quadratic dose–effect relationship up to around 5 Gy for acute photon exposures. Owing to these characteristics, the dicentric assay is able to detect whole body doses down to about 0.1 Gy from the analysis of 500–1000 metaphase spreads. Ideally, the dicentric assay is performed on blood samples within a few days of the exposure.

A–8. The duration of this assay depends on the number of cells analysed, on the level of automation and on the experience of the personnel. An assay takes 3 d or longer, including at least 51 h for sample preparation. Dose estimates based on an analysis of 20–50 cells (1–2.5 h) are sufficient for estimating the order of magnitude of the exposure, even if with large uncertainties (±0.5 Gy). Mathematical procedures are available to take partial body exposure or dose protraction into account [A–2, A–3].

Premature chromosomal condensation technique

A–9. The premature chromosomal condensation technique enables the visualization of chromosomal aberrations during interphase in both cycling and non-cycling cells. The frequency of spontaneously occurring premature chromosomal condensation fragments is in the range of 1–3 in 1000 cells. In general, 4–5 excess fragments per cell per gray are observed for low linear energy transfer radiation. For the premature chromosomal condensation assay, unstimulated lymphocytes should be immediately isolated following exposure in order to perform fusion with mitotic ovary cells of the Chinese hamster. If sampling is delayed, the repair kinetics for premature chromosomal condensation fragments should be taken into account.

A–10. The whole process from collecting blood to slide preparation takes 3 h at most. Microscope scoring of Giemsa stained preparations is time consuming. However, utilization of automated systems for scoring premature chromosomal condensation fragments, currently under development, could speed up the analysis.

A–11. The chemically induced premature chromosomal condensation assay uses the phosphatase inhibitors calyculin A and okadaic acid, which induce chromosome condensation in S and G2 phase cells but not in unstimulated

lymphocytes. This assay therefore takes at least 40 h. It has been found to be suitable for the analysis of ring chromosomes, especially at higher doses [A–4].

Micronucleus assay

A–12. Micronuclei arise from acentric fragments or whole chromosomes that are not incorporated into the daughter nuclei during cell division. Micronuclei are not radiation specific; they can be caused by exposure to many clastogenic and aneugenic agents. Like dicentrics, micronuclei represent unstable chromosomal aberrations, which disappear with time after exposure, and thus their use is restricted to rather recent exposures.

A–13. Compared to the dicentric assay, scoring of micronuclei is simple and quick, and does not require extensive experience in cytogenetics. Together with the fact that micronuclei scoring can be automated, this technique is attractive for high throughput analysis and has been validated as a good dosimetric tool in a limited number of radiation accidents [A–2]. However, it does not allow the assessment of partial body exposure, as micronuclei are inherently overdispersed.

A–14. The greatest limitations of the micronucleus assay technique are the time needed to obtain a first dose estimate (at least 75 h owing to the fact that lymphocytes require 3 d to enter cytokinesis following stimulation) and the relatively high and variable spontaneous yield of micronuclei, which tends to increase with age and is more pronounced in females [A–5]. The detection limit can be lowered to 0.05–0.1 Gy by restricting scoring to centromere negative micronuclei, since their frequency is not affected by the age dependent increase [A–6].

Fluorescence in situ hybridization

A–15. The technique most commonly used is single colour FISH, which enables the detection of interexchanges, such as dicentrics and translocations. In order to assess induced translocations among different labelled chromosomes, multicolour FISH and, for whole genome analysis, multiplex FISH have been developed. Multiplex FISH is the method of choice for studying complex interchromosomal rearrangements. It is a 24 colour technique for identifying and evaluating the size, shape and number of chromosomes in a sample of body cells.

A–16. Translocation frequencies have been shown to persist for many years in circulating lymphocytes [A–7 to A–10], making this technique very advantageous in cases of protracted exposure or for assessment of old exposures. FISH

techniques have been most widely used for individuals exposed to low linear energy transfer radiation, but they have also been used for individuals exposed to high linear energy transfer radiation.

A–17. Processing times are about 5 d after receipt of a blood sample, owing to the lengthy hybridization protocols. Background frequencies increase significantly with age [A–11, A–12], and they can vary greatly between individuals of similar age and dose history. Smoking habits have been suggested to be a significant additional confounding factor [A–12].

GENETIC TECHNIQUES

Somatic mutation assays

A–18. Two somatic mutation assays that have been suggested for use as alternative biodosimeters to chromosomal aberration analysis are the glycophorin A and hypoxanthine guanine phosphoribosyl transferase mutation assays. Several studies have compared one or both of these assays with chromosomal aberration analysis and all have concluded the latter to be the technique of choice for retrospective biodosimetry [A–13 to A–15].

Gene expression assays

A–19. Expression levels of many genes are modulated in response to exposure to ionizing radiation. Gene expression profiles have been assessed in radiation workers and radiotherapy patients [A–16 to A–19]. The key steps in the application of the assay in array format are RNA extraction, labelling and hybridization. About 2 d might be necessary before a dose estimate for fewer than ten samples can be obtained.

PROTEIN BIOMARKERS

A–20. Numerous changes in protein abundance and localization as well as enzymatic modifications occur as a consequence of biological responses to radiation exposure at the cellular, tissue or systemic level. Such changes can be identified in urine samples or blood samples by using a range of proteomic approaches. The time between the receipt of a sample and the result is typically of the order of a few hours for these assays.

Gamma-H2AX

A–21. The immunofluorescence microscopic detection of foci of the phosphorylated histone gamma-H2AX — which form at the sites of DNA double strand breaks — has been tested in multiple clinical settings, showing that it is a sensitive biomarker for radiation exposure. Gamma-H2AX foci form within minutes after exposure in a dose dependent manner. Foci levels peak within less than an hour and then decay rapidly, returning to baseline levels within one to several days, depending on the dose received.

A–22. The sensitivity of this array is reduced by considerable inter-individual variation of baseline levels and by the rapid loss of foci over time. It can therefore be reliably applied only to very recent exposures (within less than 1 d). Automated foci scoring techniques ensure more reproducible scoring criteria [A–20].

C-reactive protein

A–23. A high level of radiation exposure induces an inflammatory response, which, through cytokines, triggers the induction of C-reactive protein for a few days after the exposure. Given that C-reactive protein is increased in a large number of acute or chronic medical conditions, it is not specific to radiation exposure and, therefore, it is unsuitable as a stand-alone biodosimetry tool.

A–24. The advantage of the C-reactive protein assay is that it is already fully automated and can be performed rapidly (within a few hours) at any modern hospital with a clinical biochemistry department. Furthermore, hand-held deployable C-reactive protein assay systems are in routine use, and it can therefore be used as a rapid screening tool.

Serum amylase

A–25. Increased serum amylase activity (hyperamylasaemia) is observed after exposure of the salivary tissue, as a consequence of the induction of acute inflammatory and degenerative changes. In a similar fashion to that for C-reactive protein, serum amylase levels increase in a dose dependent manner, peaking at 18–30 h after exposure and returning to baseline levels within a few days [A–21]. One obvious limitation of the technique is that it is restricted to the dose received by the salivary gland, since exposure of other tissues would not significantly change amylase levels. Furthermore, as with C-reactive protein, it is not specific to radiation exposure and is, therefore, likewise unsuitable as a stand-alone biodosimetry tool.

A–26. Various other protein markers for human radiation exposure have been suggested [A–22, A–23].

PHYSICAL TECHNIQUES

A–27. Physical techniques are techniques that involve the investigation of physical effects produced by radiation, rather than biological effects, even when performed in biological tissues such as hair, fingernails, tooth enamel and bone. In general, the time from receipt of a sample to the dose estimate is between 1 h and 48 h, depending on the accuracy required.

Electron paramagnetic resonance dosimetry

A–28. The electron paramagnetic resonance (EPR) technique gives an estimate of the absorbed dose by detection of the paramagnetic centres, such as radicals or point defects that are generated specifically by ionizing radiation. Typical applications are EPR spectroscopy with tooth enamel [A–24, A–25] or, when bone biopsies are available, with bone tissue [A–26]. Both applications require invasive sample collection, however. Other suitable materials that can be collected with non-invasive procedures include sugar, plastics, glass, wool, cotton, hair and fingernails.

A–29. The time stability of the EPR signal varies widely between materials, ranging from 5 d to 7 d for plastics [A–27] to around 106 years for tooth enamel [A–28]. The presence of background non-radiation induced signals in EPR affects the detection limits of the technique, which vary widely between around 100 mGy for tooth enamel and 10 Gy for cotton.

A–30. The preparation of samples for EPR dosimetry is usually relatively simple. Depending on the material, a single measurement can take between some minutes up to a few hours. The readout is non-destructive, allowing for repeated measurements of the same sample. A drawback is that spectrometers for EPR are expensive and highly qualified personnel are required for their operation.

A–31. Techniques for in vivo measurements of teeth by EPR use microwave frequencies of 1 GHz [A–29], which are lower than the frequencies used for conventional in vitro measurements (~10 GHz). With low frequency microwaves, a loss in sensitivity by a factor of five to ten compared with X band spectrometry is expected from calculations; hence, the detection limit is expected to be in the range of 0.5–1 Gy.

Luminescence dosimetry

A–32. The basis for luminescence techniques in retrospective dosimetry is the same as that described in Appendix II for luminescence techniques in prospective dosimetry. Quartz extracted from bricks and other fired building materials is currently the main mineral used for the purposes of retrospective luminescence dosimetry. In addition to quartz, other phosphors have recently been studied, which can be found either in the urban environment or in materials carried on, or close to, the body by the general population [A–30].

A–33. Examples of such materials include memory chip modules from telephone cards, identity cards, health insurance cards, cash cards and credit cards [A–30 to A–34]; ceramic resistors of portable electronic devices such as mobile telephones [A–34, A–35]; materials used for dental restoration [A–31, A–36]; tooth enamel [A–37, A–38]; household and workplace chemicals [A–39, A–40]; and glass [A–41].

A–34. Procedures for sample preparation and measurement protocols vary but they are comparatively quick and easy for most materials: a sample from a personal object can be processed within less than an hour. Most of these items show a linear dose response over a wide dose range, and detection limits of the order of 10 mGy can be achieved for most materials.

Activation techniques

A–35. Neutron activation techniques are based on the measurement of activity induced by neutron interaction with biological tissues, such as blood, hair or fingernails, or metallic elements worn by the victims, such as coins, jewellery or belt buckles.

A–36. Activation techniques can be used in the management of the emergency response for a criticality accident and in dose reconstructions many years after exposure to neutrons. In the early phase of the management of a criticality accident, rapid and efficient triage can be performed by using the measurement of sodium activation in humans. At the site of an accident, it is possible to perform very rapid measurements of gamma radiation emitted by ^{24}Na (produced by activation of ^{24}Na in the body and emitting gamma peaks at 1.36 MeV and 2.75 MeV, with a half-life of 14.96 h) with a simple direct gamma survey instrument positioned against the abdominal area of a victim. A more precise estimation of sodium activity in the victim can be performed at a later time by using a whole body counter or by gamma spectrometry of blood samples.

A–37. Measurements of activated sulphur in hair and fingernails have also been used for the purposes of dose reconstruction following accidents. In this case, the beta particles emitted by ^{32}P, produced by activation of ^{32}S in the body, can be measured directly by using a Geiger–Müller counter or by liquid scintillation techniques, following simple chemical procedures.

A–38. Another possibility is to determine doses by measuring long lived activated nuclei in environmental samples (e.g. ^{63}Ni in copper samples and ^{152}Eu, ^{60}Co, ^{59}Ni, ^{41}Ca, ^{39}Ar, ^{36}Cl, ^{14}C or ^{10}Be in granite) or in biological materials (e.g. ^{41}Ca in tooth enamel), as was done for atomic bomb survivors in Japan.

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ABBREVIATIONS

AMAD	activity median aerodynamic diameter
DTPA	diethylenetriaminepentaacetic acid
EPR	electron paramagnetic resonance
FISH	fluorescence in situ hybridization
HEPA	high efficiency particulate air
ICRP	International Commission on Radiological Protection
ICRU	International Commission on Radiation Units and Measurements
IEC	International Electrotechnical Commission
RBE	relative biological effectiveness

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