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ICRP *Publication 103* and beyond

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Abstract

This paper focuses on ICRP *Publication 103*, the 2007 Recommendations of the International Commission on Radiological Protection, which lays out the system of radiological protection for all exposure situations and exposure types. In addition, subsequent ICRP publications which delve more deeply into specific aspects of this system are reviewed to some extent, including: ICRP *Publication 104* Scope of Radiological Protection Control Measures; ICRP *Publication 105*, Radiological Protection in Medicine; ICRP *Publication 108*, Environmental Protection — the Concept and Use of Reference Animals and Plants; ICRP *Publication 109*, Application of the Commission's Recommendations for the Protection of People in Emergency Exposure Situations; and an ICRP publication in press titled Application of the Commission's Recommendations to the Protection of People Living in Long Term Contaminated Areas After a Nuclear Accident or a Radiation Emergency.

Introduction

The International Commission on Radiological Protection (ICRP) is an independent, international organization that advances for the public benefit the science of radiological protection, in particular by providing recommendations and guidance on all aspects of protection against ionizing radiation.

ICRP was established in 1928 by the International Society of Radiology (ISR) to respond to growing concerns about the effects of ionizing radiation being observed in the medical community.

In preparing its recommendations, ICRP considers the fundamental principles and quantitative bases upon which appropriate radiation protection measures can be established, while leaving to the various national protection bodies the responsibility of formulating the specific advice, codes of practice, or regulations that are best suited to the needs of their individual countries.

ICRP offers its recommendations to regulatory and advisory agencies and provides advice the intended to be of help to management and professional staff with responsibilities for radiological protection. Although ICRP itself has no formal power to impose its recommendations, in fact legislation in most countries adheres closely to ICRP recommendations. In addition, the International Atomic Energy Agency (IAEA) International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources (commonly referred to as “the BSS”) is based heavily

on ICRP recommendations, and the International Labour Organisation (ILO) Convention 115, Radiation Protection Convention, General Observation 1992, refers specifically to the recommendations of ICRP. Effectively, ICRP recommendations form the basis of radiological protection practice, programmes, regulations, and international standards and guidance worldwide.

ICRP has published well over one hundred publications on all aspects of radiological protection. Most address a particular area within radiological protection, but a handful of publications, the so-called fundamental recommendations, each describe the overall system of radiological protection. The system of radiological protection has been developed by ICRP based on (i) the current understanding of the science of radiation exposures and effects and (ii) value judgements. These value judgements take into account societal expectations, ethics, and experience gained in application of the system.

The 1990 Recommendations of ICRP (ICRP *Publication 60*) form the basis of the current IAEA BSS, and are also the foundation of most radiological protection practices, programmes and regulations worldwide. The 2007 Recommendations of ICRP (ICRP *Publication 103*) recently replaced the 1990 Recommendations. They are the result of nearly a decade of development and several major worldwide public consultations.

The 2007 Recommendations replace the 1990 Recommendations and update, consolidate, and develop additional guidance on the control of exposure from radiation sources issued since 1990. They reflect a more up-to-date understanding of the science behind radiological protection and evolving societal expectations.

The System of Radiological Protection

The system of radiological protection described in the 2007 Recommendations is an evolutionary change from that described in the 1990 Recommendations. This evolution is necessary in order for the system to remain current with our evolving understanding of the relevant scientific findings, and also to continue to reflect current societal norms. For example, the revised radiation and tissue weighting factors reflect updated scientific knowledge, while a greater emphasis on environmental protection reflects a heightened social awareness of the importance of this area. As well, practical application of the system can point out areas for improvement.

What follows is a brief review of the system of radiological protection as described in the 2007 Recommendations and subsequent publications. Much of this is, necessarily, taken almost directly from ICRP *Publication 103*.

Scope

The system of radiological protection applies to all exposures to ionising radiation from any source, regardless of its size and origin. However, the system can apply in its entirety only to situations in which either the source of exposure or the pathways leading to the doses received by individuals can be controlled by some reasonable means. Some exposure situations are excluded from radiological protection legislation, usually on the basis that they are unamenable to control with regulatory instruments, and some exposure situations are exempted from some or all regulatory requirements

where such controls are regarded as unwarranted. ICRP *Publication 104* elaborates on the scope of radiological protection control measures.

Health effects of ionising radiation

An understanding of the health effects of ionising radiation is central to the system of radiological protection. Following a review of the biological and epidemiological information on the health risks attributable to ionising radiation ICRP concluded that the distribution of risks to different organs/tissues has changed somewhat since 1990. However, assuming a linear response at low doses, the combined detriment due to excess cancer and heritable effects remains essentially unchanged at around 5% per Sv (see Table 1). Embodied in this current estimate is the use of a dose and dose-rate effectiveness factor for solid cancers which is unchanged at a value of 2. In addition, following prenatal exposure, the cancer risk will be similar to that following irradiation in early childhood, and a threshold dose exists for the induction of malformations and for the expression of severe mental retardation.

The assumption of a linear dose–response relationship for the induction of cancer and heritable effects, according to which an increment in dose induces a proportional increment in risk even at low doses, continues to provide the basis for the summation of doses.

Table 1. Detriment-adjusted nominal risk coefficients (10^{-2} Sv^{-1}) for stochastic effects after exposure to radiation at low dose rate.

Exposed population	Cancer	Heritable effects	Total
Whole	5.5	0.2	5.7
Adult	4.1	0.1	4.2

From ICRP *Publication 103*, Table 1.

Dose limits

The dose limits (see Table 2) remain unchanged from those recommended in 1990. However, it is recognized that further information is needed and revised judgements may be required particularly in respect of the lens of the eye. In addition, emerging data on possible excess risk in non-cancer diseases (e.g., cardiovascular disorders) are being watched closely.

Table 2. Dose limits in planned exposure situations.

Type of Dose Limit	Occupational	Public
Effective dose	20 mSv per year, averaged over defined periods of 5 years, and 50 mSv in any single year[a]	1 mSv in a year
Annual eququivalent dose in: Lens of the eye Skin Hands and feet	 150 mSv 500 mSv 500 mSv	 15 mSv 50 mSv –

[a] Additional restrictions apply to the occupational exposure of pregnant women.
From ICRP *Publication 103*, Table 6.

The use and calculation of equivalent and effective dose

Effective dose is intended for use as a protection quantity. The main uses of effective dose are the prospective dose assessment for planning and optimisation in radiological protection, and demonstration of compliance with dose limits for regulatory purposes.

Although the use of equivalent and effective dose remains unchanged, a number of revisions have been made to the methods used in their calculation. Reviews of the range of available data on the relative biological effectiveness of different radiations, together with biophysical considerations, have led to changes to some of the values of radiation weighting factors (see Table 3 and Figure 1). In addition, the distribution of risks to different organs/tissues has changed somewhat since 1990, particularly in respect of the risks of breast cancer and heritable disease, resulting in revised tissue weighting factors (see Table 4) intended to apply as rounded values to a population of both sexes and all ages.

Another change is that doses from external and internal sources will be calculated using reference computational phantoms of the human body based on CT images (the first appearing in ICRP *Publication 110*), replacing the use of various mathematical models. For adults, equivalent doses will be calculated by sex-averaging of values obtained using male and female phantoms.

Note that calculation of equivalent and effective dose use models and parameters selected from a range of experimental investigations and human studies through judgements, and meant to apply to a population rather than individuals. For individual retrospective dose and risk assessments, individual parameters and uncertainties have to be taken into account.

Collective effective dose is used for optimisation, predominantly in the context of occupational exposure. It is not intended as a tool for epidemiological risk assessment, and it is inappropriate to use it in risk projections. The aggregation of very low individual doses over extended time periods is inappropriate, and in particular, the calculation of the number of cancer deaths based on collective effective doses from trivial individual doses should be avoided.

Table 3. Radiation weighting factors.

Radiation type	Radiation Weighting Factor, w_R
Photons	1
Electrons and muons	1
Protons and charged pions	2
Alpha particles, fission fragments, heavy ions	20
Neutrons	A continuous function of neutron energy (See Figure 1)

From ICRP *Publication 103*, Table 2.

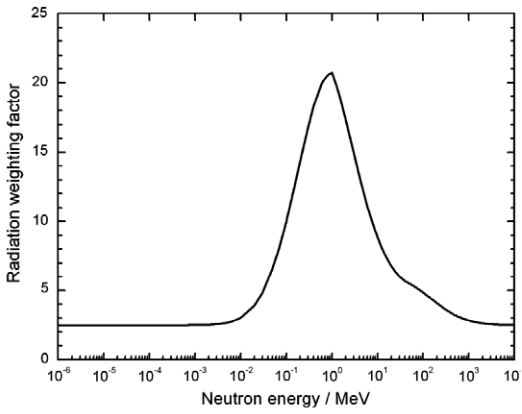

Fig. 1. Radiation weighting factor, w_R, for neutrons versus neutron energy (from ICRP *Publication 103*, Fig.1).

Table 4. Tissue weighting factors.

Tissue	Tissue weigting factor, w_T	$\sum w_T$
Bone-marrow (red), Colon, Lung, Stomach, Breast, Remainder tissues*	0.12	0.72
Gonads	0.08	0.08
Bladder, Oesophagus, Liver, Thyroid	0.04	0.16
Bone surface, Brain, Salivary glands, Skin	0.01	0.04
	TOTAL	1.00

* Remainder tissues: Adrenals, Extrathoracic (ET) region, Gall bladder, Heart, Kidneys, Lymphatic nodes, Muscle, Oral mucosa, Pancreas, Prostate (male), Small intestine, Spleen, Thymus, Uterus/cervix (female). From ICRP *Publication 103*, Table 3.

Planned, existing and emergency exposure situations

The system of radiological protection described in the 2007 Recommendations recognises planned, existing and emergency exposure situations. These are intended to cover the entire range of exposure situations, and replace the previous categorisation into practices and interventions.

Planned exposure situations (which include situations previously categorised as practices) encompass sources and situations that have been appropriately managed within the system of radiological protection. Protection during medical uses of radiation is also included in this type of exposure situation. Recommendations for planned exposure situations are substantially unchanged from these provided previously.

Existing exposure situations include naturally occurring exposures as well as exposures from past events and accidents, and practices conducted outside the system of radiological protection. In this type of situation, protection strategies will often be implemented in an interactive, progressive manner over a number of years.

Indoor radon in dwellings and workplaces is an important existing exposure situation, and is the subject of current high-priority work for ICRP. In November 2009 ICRP issued a Statement on Radon. The intention is to publish this Statement in the Annals of the ICRP with an accompanying report on assessment of lung cancer risk from radon, after undertaking a public consultation on the two documents together.

The statement reaffirms that, for planned exposure situations, any workers' exposure to radon incurred as a result of their work, however small, shall be considered as occupational exposure. However, because of the ubiquity of radiation, the direct application of this definition to radiation would mean that all workers should be subject to a regime of radiological protection. Therefore the use of 'occupational exposures' is limited to radiation exposures incurred at work as a result of situations that can reasonably be regarded as being the responsibility of the operating management.

Emergency exposure situations include consideration of emergency preparedness and emergency response. Emergency preparedness should include planning for the implementation of optimised protection strategies which have the purpose of reducing exposures, should the emergency occur, to below the selected value of the reference level. During emergency response, the reference level would act as a benchmark for

evaluating the effectiveness of protective actions and as one input into the need for establishing further actions.

Occupational, public and medical exposures

The system of radiological protection continues to distinguish among three categories of exposure: occupational, public, and medical. Table 5 shows these three categories of exposure, the three exposure situations, and the dose constraints and reference levels applicable in each circumstance.

Table 5. Dose constraints and reference levels used in the system of radiological protection.

Type of situation	Occupational exposure	Public Exposure	Medical Exposure
Planned exposure	Dose limit Dose constraint	Dose limit Dose constraint	Diagnostic reference level (Dose constraint[b])
Emergency exposure	Refrerence level	Reference level	n/a
Existing exposure	n/a[a]	Reference level	n/a

[a] Exposures resulting from long-term remediation operations or from protracted employment in affected areas should be treated as part of planned occupational exposure, even though the source of radiation is 'existing'.
[b] Comforters, carers, and volunteers in research only.
From ICRP *Publication 103*, Table 4.

Fundamental principles of radiological protection

The three key principles of radiological protection have remained unchanged in the 2007 Recommendations. The principles of justification and optimisation apply in all exposure situations. The principle of application of dose limits applies only for doses expected to be incurred with certainty as a result of planned exposure situations. These principles are defined as follows:

- The Principle of Justification: Any decision that alters the radiation exposure situation should do more good than harm.
- The Principle of Optimisation of Protection: The likelihood of incurring exposure the number of people exposed, and the magnitude of their individual doses should all be kept as low as reasonably achievable, taking into account economic and societal factors.
- The Principle of Application of Dose Limits: The total dose to any individual from regulated sources in planned exposure situations other than medical exposure of patients should not exceed the appropriate limits specified by the Commission.

Emphasis on optimisation

The 2007 Recommendations emphasise the key role of the principle of optimisation. This principle should be applied in the same manner in all exposure situations. Restrictions are applied to doses to a nominal individual (the Reference Person), namely dose constraints for planned exposure situations and reference levels for emergency and existing exposure situations. Options resulting in doses greater in magnitude than such restrictions should be rejected at the planning stage. Importantly,

these restrictions on doses are applied prospectively, as with optimisation as a whole. If, following the implementation of an optimised protection strategy, it is subsequently shown that the value of the constraint or reference level is exceeded, the reasons should be investigated but this fact alone should not necessarily prompt regulatory action. This emphasis on a common approach to radiological protection in all exposure situations should aid application of the system of radiological protection in the various circumstances of radiation exposure.

Table 6. Framework for source-related dose constraints and reference levels with examples of constraints for workers and the public from single dominant sources for all exposure situations that can be controlled.

Bands of constraints and reference levels[a] (mSv)	Characteristics of the exposure situation	Radiological protection requirements	Examples
Greater than 20 to 100	Individuals exposed by sources that are not controllable, or where actions to reduce doses would be disproportionately disruptive. Exposures are usually controlled by action on the exposure pathways.	Consideration should be given to reducing doses. Increasing efforts should be made to reduce doses as they approach 100 mSv. Individuals should receive information on radiation risk and on the actions to reduce doses. Assessment of individual doses should be undertaken.	Reference level set for the highest planned residual dose from a radiological emergency.
Greater than 1 to 20	Individuals will usually receive benefit from the exposure situation but not necessarily from the exposure itself. Exposures may be controlled at source or, alternatively, by action in the exposure pathways.	Where possible, general information should be made available to enable individuals to reduce their doses. For planned situations, individual assessment of exposure and training should take place.	Constraints set for occupational exposure in planned situations. Constraints set for comforters and carers of patients treated with radiopharmaceuticals. Reference level for the Highest planned residual dose from radon in dwellings.
1 or less	Individuals are exposed to a source that gives them little or no individual benefit but benefits to society in general. Exposures are usually controlled by action taken directly on the source for which radiological protection requirements can be planned in advance.	General information on the level of exposure should be made available. Periodic checks should be made on the exposure pathways as to the level of exposure.	Constraints set for public exposure in planned situations.

[a] Acute or annual dose.
From ICRP *Publication 103*, Table 5.

The relevant national authorities will often play a major role in selecting values for dose constraints and reference levels. Guidance on the selection process is provided by recommending bands of constraints and reference levels (see Table 6).

Emphasis on optimisation using reference levels in emergency and existing exposure situations focuses attention on the residual level of dose remaining after implementation of protection strategies. This residual dose should be below the reference level, which represents the total residual dose as a result of an emergency, or in an existing situation, that the regulator would plan not to exceed. These exposure situations often involve multiple exposure pathways so protection strategies involving a number of different protective actions will have to be considered. However, the process of optimisation will continue to use the dose averted by specific countermeasures as an important input into the development of optimised strategies.

Radiological protection of the environment

In the past, ICRP concerned itself with the environment primarily with regard to the transfer of radionuclides through it because this directly affects the radiological protection of human beings. In the 1990 Recommendations ICRP stated that "the standard of environmental control needed to protect man ... will ensure that other species are not put at risk. Occasionally, individual members of non-human species might be harmed, but not to the extent of endangering whole species or creating imbalance between species."

However, in its 2007 Recommendations, ICRP acknowledges that that it is now necessary to demonstrate, directly and explicitly, that the environment is being protected. Therefore, it is necessary to develop a clearer framework to assess the relationships between exposure and dose, and between dose and effect, and the consequences of such effects, for non-human species, on a common scientific basis. This is further developed in ICRP *Publication 108*.

Recent ICRP Publications which Further Elaborate the System of Radiological Protection

ICRP *Publication 104*: Scope of Radiological Protection Control Measures

In principle the system of radiological protection applies to all exposures to ionizing radiation. However, in practice the measures undertaken to control these exposures must be limited based on practical considerations. ICRP *Publication 104* discusses the scope of radiological protection control measures, and describes certain tools (e.g. exemption, exclusion and clearance) that can be used to manage the scope.

ICRP *Publication 108*: Environmental Protection – the Concept and Use of Reference Animals and Plants

The 2007 Recommendations acknowledge that that it is now necessary to demonstrate, directly and explicitly, that the environment is being protected, rather than simply assume that adequate protection of humans is sufficient to protect the environment. Therefore, it is necessary to develop a clearer framework to assess the relationships between exposure and dose, and between dose and effect, and the consequences of such effects, for non-human species, on a common scientific basis.

To this end, ICRP *Publication 108* sets out some high-level ambitions with regard to environmental protection. To aid in demonstrating whether these ambitions are being achieved, and help optimise the level of effort that might be expended on environmental protection, and ICRP has developed a set of Reference Animals and Plants (RAPs) and derived consideration reference levels (DCRLs). ICRP does not propose the application of dose limits to Reference Animals and Plants.

Recent ICRP Publications Elaborating on Specific Exposure Categories and Situations

One of the first publications following the 2007 Recommendations was ICRP *Publication 105* on Radiological Protection in Medicine. This publication describes the application of the system of radiological protection for medical exposures, an exposure category quite different from occupational and public exposures by virtue of the fact that most of the time, most of the benefit and the detriment apply to a single individual: the patient. In the other two exposure categories, typically one group receives most of the detriment (e.g. the workers in a facility), while another group receives most of the benefit (e.g. the public at large receiving electrical power).

Two publications each examine one of the three exposure situations: ICRP *Publication 109* examines emergency exposure situations, and the ICRP publication in press titled Application of the Commission's Recommendations to the Protection of People Living in Long Term Contaminated Areas after a Nuclear Accident or a Radiation Emergency examines the system of radiological protection as applied to existing exposure situations.

Other Current Initiatives and Future Work

For more information on current and future work of ICRP the reader is referred to another paper prepared for this congress by the same author titled “International Commission on Radiological Protection – recent publications, current initiatives and future work”.

Acknowledgements

Preparation of this paper drew extensively on the hard work of ICRP members through several ICRP publications shown in the references section. The author would like to acknowledge the dedication of those ICRP members who contributed to these publications, and thus recognize their important contribution to this paper.

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Radiation protection metrology and measurements

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Abstract
Quantitative assessments of external and internal exposure need reasonable state-of-the-art physical / biological / ecological models and reliable quantities and measurement methods and instruments. Fundamental basis for dose and activity measurements are solid instruments and adequate measurement methods.

In this paper, the appropriate quantities and units as far as necessary measurement chains to assure traceability from international primary standards to end-user measurement facilities are given. The basic concepts and practical implementation of uncertainty assessments in activity and dose measurements methods are presented. Finally main aspects of quality assurance in the field of radiation protection measurements are discussed.

The target group for this information are newcomers as far as experienced radiation protection experts, radionuclide and dosimetry metrologists and end-user of measurement instruments in radiation protection.

Introduction
Accurate and high-quality measurements of ionizing radiation and radionuclide activities are required in a wide range of industrial, scientific and medical applications where they are critical relating to human health and safety. Dose and activity measurements for radiation protection are stringent in its accuracy and reliability requirements. Generally this means that the uncertainty of measurements for radiation protection around the internationally agreed dose limits should not exceed a few percent.

To provide an international metrological basis for measurements in radiation protection and other sectors (e.g. medical diagnostics and therapy), the 11th *General Conference of the Meter Convention* decided to establish the Ionizing Radiation section at the *Bureau International des Poids et Mesures*, BIPM, in 1960.

The main activities of the BIPM in the field of ionizing radiation are to maintain the international reference standards for dose and activity measurements [1]. These standards are used in the BIPM key comparisons and their development and

improvement is a major part of the international metrological research and development programme. The ionising radiation section of the BIPM also undertakes calibrations for national laboratories, and participates in international comparisons.

At the European level *European Association of National Metrology Institutes* EURAMET is acting as the regional metrolo¬gical organisation. It is coordinating the metrological activities of the European National Metrology Institutes (NMI's) and Designated Institutes (DI's) of the European Union including the Joint Research Centres of the European Commission, EFTA and EU Accession States. The objective of EURAMET is to promote the coordination and development of metrological activities and services and support the European Metrology Research Programme EMRP. EURAMET is working in Technical Committees; on of them is the TC Ionising Radiation. This TC is divided the Sub-Fields photon dosimetry, radioactivity, and neutron measurements. The basic activities are jointly done by co-operative scientific and technical projects e.g. joint research projects and comparisons.

In Fig. 1 the international, European and national (for example the Austrian) organisations supporting the quality of measurements for radiation protection are schematically shown. In addition to the metrological infrastructure there is a support in legal metrology (International Organization of Legal Metrology OIML; European Cooperation in Legal Metrology WELMEC; national/Austria: Federal Office for Metrology and Surveying BEV) and for accreditation (International Laboratory Accreditation Cooperation ILAC; European co-operation for Accreditation EA; National/Austria: Federal Ministry of Economy, Family and Youth, BMWFJ) established.

	International	Europe	National e.g. Austria
Metrology	BIPM	EURAMET European Association of National Metrology Institutes	Willkommen im BEV
Legal Metrology	OIML	European Cooperation in Legal Metrology WELMEC	Willkommen im BEV
Accreditation	ilac	European co-operation for Accreditation	bmwfj Federal Ministry of Economy, Family and Youth

Fig. 1. International, European and national metrological infrastructure.

Fig. 2 shows the international organisations and network between society's and economics' needs together with science and technological support for the provision of quality in measurements. Additional to ILAC the *International Accreditation Forum, Inc. (IAF)* acts as the world association of Conformity Assessment Accreditation Bodies and other bodies interested in conformity. In the field of technological standardisation the *International Electrotechnical Commission* IEC (Technical Committee 45 *Nuclear*

instrumentation, Sub Committee 45B *Radiation protection instrumentation*), the *International Organisation for Standardisation* ISO (Technical Committee 85 *Nuclear Energy*, Sub Committee 2 *Radiation Protection*) and the *Telecommunication Standardization Sector of the International Telecommunication Union* ITU-T (for non-ionising radiation) support the development and implementation of measurement radiation protection instruments and methods.

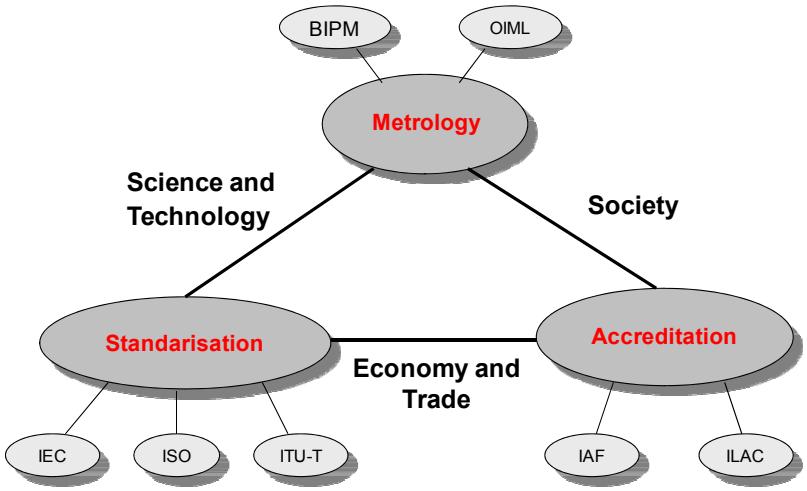

Fig. 2. International organisations and network to ensure quality in measurements.

Basic metrological items and its definitions

In this section the most significant basic definitions of items for metrology and measurement are given. The items / definitions are largely compatible to them given in the International Vocabulary in Metrology (ISO/IEC, 2007; VIM, 2008).

Quantity

… Property of a phenomenon, body, or substance, where the property has a magnitude that can be expressed as a number and a reference.

A reference can be a measurement unit, a measurement procedure, a reference material, or a combination of such. Symbols for quantities are given in ISO 80000 and IEC 80000 series Quantities and Units.

Measurement unit; unit

… Real scalar quantity, defined and adopted by convention, with which any other quantity of the same kind can be compared to express the ratio of the two quantities as a number.

Measurement units are designated by conventionally assigned names and symbols. In some cases special measurement unit names are restricted to be used with quantities of a specific kind only. For example, the measurement unit ‘second to the power minus one’ (1/s) is called hertz (Hz) when used for frequencies and becquerel (Bq) when used for activities of radionuclides.

Metrological traceability; traceability

… Property of a measurement result whereby the result can be related to a reference through a documented unbroken chain of calibrations, each contributing to the measurement uncertainty.

For this definition, a 'reference' can be a definition of a measurement unit through its practical realization, or a measurement procedure including the measurement unit for a non-ordinal quantity, or a measurement standard. The abbreviated term "traceability" is sometimes used to mean 'metrological traceability'. The full term of "metrological traceability" is preferred if there is any risk of confusion.

Metrological traceability chain; traceability chain

… Sequence of measurement standards and calibrations that is used to relate a measurement result to a reference.

A metrological traceability chain is defined through a calibration hierarchy.

Metrological traceability to a measurement unit; metrological traceability to a unit

… Metrological traceability where the reference is the definition of a measurement unit through its practical realization.

The expression "traceability to the SI" means 'metrological traceability to a measurement unit of the International System of Units'.

Calibration

… Operation that, under specified conditions, in a first step, establishes a relation 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 relation for obtaining a measurement result from an indication.

A calibration may be expressed by a statement, calibration function, calibration diagram, calibration curve, or calibration table. In some cases, it may consist of an additive or multiplicative correction of the indication with associated measurement uncertainty. Calibration should not be confused with adjustment of a measuring system, often mistakenly called "self-calibration", nor with verification of calibration. Often, the first step alone in the above definition is perceived as being calibration.

Calibration hierarchy

… Sequence of calibrations from a reference to the final measuring system, where the outcome of each calibration depends on the outcome of the previous calibration.

Measurement uncertainty necessarily increases along the sequence of calibrations. The elements of a calibration hierarchy are one or more measurement standards and measuring systems operated according to measurement procedures. For this definition, the 'reference' can be a definition of a measurement unit through its practical realization, or a measurement procedure, or a measurement standard.

Measurement standard; etalon

… Realization of the definition of a given quantity, with stated quantity value and associated measurement uncertainty, used as a reference

A "realization of the definition of a given quantity" can be provided by a measuring system, a material measure, or a reference material. A measurement standard is frequently used as a reference in establishing measured quantity values and associated measurement uncertainties for other quantities of the same kind, thereby establishing metrological traceability through calibration of other measurement standards, measuring instruments, or measuring systems.

A primary measurement standard is a measurement standard established using a primary reference measurement procedure, or created as an artifact, chosen by convention (eg. Kilogramm artefact) than again a secondary measurement standard is established through calibration with respect to a primary measurement standard for a quantity of the same kind.

Measurement uncertainty; uncertainty

… Non-negative parameter characterizing the dispersion of the quantity values being attributed to a measurand, based on the information used.

Notes:

Measurement uncertainty includes components arising from systematic effects, such as components associated with corrections and the assigned quantity values of measurement standards, as well as the definitional uncertainty. Sometimes estimated systematic effects are not corrected for but, instead, associated measurement uncertainty components are incorporated. The parameter may be, for example, a standard deviation called standard measurement uncertainty (or a specified multiple of it), or the half-width of an interval, having a stated coverage probability.

Measurement uncertainty comprises, in general, many components. Some of these may be evaluated by Type A evaluation of measurement uncertainty from the statistical distribution of the quantity values from series of measurements and can be characterized by standard deviations. The other components, which may be evaluated by Type B evaluation of measurement uncertainty, can also be characterized by standard deviations, evaluated from probability density functions based on experience or other information (ISO/IEC Guide 98-3:2008).

Does quantities and their units

In this section the most significant quantities and units used for radiation protection purposes are assembled. The given items and their definitions are compiled from current international standards and documents (ICRU, 1985; 1998; IEC 60050-393, 2003; IAEA, 2007; ICRP 103, 2007; ISO 12749-1, 2010).

Absorbed dose, *D*

… Quotient of $\mathrm{d}\bar{\varepsilon}$ by d*m*, where $\mathrm{d}\bar{\varepsilon}$ is the mean energy imparted to matter of mass d*m* thus

$$D = \frac{\mathrm{d}\bar{\varepsilon}}{\mathrm{d}m} \qquad (1)$$

The unit of absorbed dose is $J \cdot kg^{-1}$. The special name for the unit of absorbed dose is gray (Gy). The mean energy imparted $d\bar{\varepsilon}$ is the expectation value of the energy imparted by ionizing radiation to the matter in a given volume. The unit of the mean energy imparted is J.

dose equivalent, *H*

… Product of *D* and *Q* at a point in tissue, where *D* is the absorbed dose and *Q* is the quality factor for the specific radiation at this point, thus

$$H = D \cdot Q \tag{2}$$

The unit of dose equivalent is joule per kilogram ($J\ kg^{-1}$), and its special name is sievert (Sv).

Ambient dose equivalent, *H*(d)*

… 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.

This quantity is defined at a point in a radiation field and is used as directly measurable proxy (i.e. substitute) for effective dose for use in monitoring of external exposure. The recommended value of d for strongly penetrating radiation is 10 mm.

The ICRU sphere is a sphere with 30 cm diameter consists of ICRU tissue. This is a material with a density of 1 g cm^{-3} and a mass composition of 76,2 % oxygen, 10,1 % hydrogen, 11,1 % carbon, and 2,6 % nitrogen (ICRU 39, 1985)

Directional dose equivalent, *H'(d, Ω)*

… 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 $\vec{\Omega}$.

This quantity is defined at a point in a radiation field and is used as directly measurable proxy (i.e. substitute) for equivalent dose in the skin for use in monitoring of external exposure. The recommended value of d for weakly penetrating radiation is 0,07 mm. In practice $\vec{\Omega}$ is chosen in direction of maximum *H'*, indicated as *H'(d)*.

Personal dose equivalent, H_p (*d*)

… Dose equivalent in the 'ICRU sphere' at an appropriate depth, d, below a specified point on the human body.

The unit of personal dose equivalent is joule per kilogram ($J \cdot kg^{-1}$) and its special name is sievert (Sv). The specified point is usually given by the position where the individual's dosimeter is worn. In most cases of penetrating radiation d = 10 mm is chosen: $H_p(10)$. For weakly penetrating radiation d = 0,07 mm is chosen: $H_p(0,07)$. In a unidirectional field, the direction can be specified in terms of the angle, α, between the direction opposing the incident field and a specified normal on the phantom surface.

Effective dose

… Sum of the weighted equivalent doses in all tissues and organs of the body.

The effective dose is expressed in units of joules per kilogram ($J \cdot kg^{-1}$) with the special name sievert (Sv). The effective dose is not directly measurable. It can be approximated by operational dose equivalent quantities for external exposure and by dose conversion models (dose conversion factors) in the case of intake (inhalation, ingestion) of radionuclides. The effective dose is used in dose limitation / restriction concepts in radiation protection. Effective dose is a measure of dose designed to reflect the amount of radiation detriment likely to result from the dose. Values of effective dose from any type(s) of radiation and mode(s) of exposure can be compared directly.

Activity quantities and their units

In this section the most significant quantities and units used for radiation protection purposes are assembled. The given items and their definitions are compiled from current international standards and documents.

Activity

… Quotient, for an amount of radionuclide in a particular energy state at a given time, of $\langle dN \rangle$ by d*t*, where $\langle dN \rangle$ is the expectation value of the number of spontaneous nuclear transitions from this energy state in the time interval of duration d*t*:

$$A = -\frac{\langle dN \rangle}{dt} \tag{3}$$

The unit of activity is s^{-1} with the special name *becquerel* (Bq).

Activity concentration;

… Quotient of activity by either the total mass or the total volume of the sample.

This quantity is expressed either in becquerels per kilogram (Bq/kg) or in becquerels per cubic-meter (Bq/m^3). For a gas, in some cases it is necessary to indicate the temperature and pressure conditions for which the activity concentration, expressed in becquerel per cubic metre, is measured, for example standard temperature and pressure (STP). For solid material, sometimes it is necessary to indicate the sample moisture or dryness conditions for which the activity concentration, expressed in becquerel per kilogram, is measured, for example Bq/kg material dried at 105 °C.

Specific activity; massic activity

… Quotient of activity by the total mass of the sample.

This quantity is expressed either in becquerels per kilogram (Bq/kg).

The distinction in usage between specific activity and activity concentration is controversial. A common distinction is that specific activity is used with reference to a pure radionuclide (e.g. 1 g ^{226}Ra has the specific activity of $37 \cdot 10^9$ Bq). Activity concentration (which may be activity per unit mass or per unit volume) is used for any other situation (e.g. when the activity is in the form of contamination in or on a material).

Volumic activity; volumetric activity

… Quotient of the activity by the total volume of the sample.

This quantity is expressed in becquerels per cubic metre (Bq/m³).

Surface activity

… Quotient of the activity by the total area of surface of the sample.

This quantity is expressed in becquerels per square metre (Bq/m²).

Measurement uncertainties in radiation protection

In most cases, a measurand Y is not measured directly, but is determined from N other quantities X_1, X_2, ..., X_N through a functional relationship f:

$$Y = f(X_1, X_2, ..., X_N) \tag{4}$$

An estimate of the measurand Y, denoted by $\bar{y}$, is obtained from Y using input estimates $\bar{x}_1, \bar{x}_2, ..., \bar{x}_N$ for the values of the N quantities X_1, X_2, ..., X_N. Thus the output estimate $\bar{y}$, which is the result of the measurement, is given by

$$\bar{y} = f(\bar{x}_1, \bar{x}_2, ..., \bar{x}_N) \tag{5}$$

When the input quantities are not correlated, the combined standard uncertainty $u_c(\bar{y})$ is the positive square root of the combined variance $u_c^2(\bar{y})$, which is given by

$$u_C(\bar{y}) = \sqrt{\sum_i \left(\frac{\partial f(\bar{x}_1, \bar{x}_2, ... \bar{x}_N)}{\partial X_i} \right)^2 \cdot u^2(\bar{x}_i)} \tag{6}$$

When the input quantities are correlated, the appropriate expression for the combined variance $u_c^2(\bar{y})$ associated with the result of a measurement is

$$u_c^2(\bar{y}) = \sum_i \sum_j \frac{\partial f}{\partial X_i} \frac{\partial f}{\partial X_j} \cdot u(\bar{x}_i, \bar{x}_j) = \sum_i \left(\frac{\partial f}{\partial X_i} \right)^2 u(\bar{x}_i) + 2 \sum_{i=1}^{N-1} \sum_{j=i+1}^{N} \frac{\partial f}{\partial X_i} \frac{\partial f}{\partial X_j} \cdot u(\bar{x}_i, \bar{x}_j) \tag{7}$$

where $u(\bar{x}_i, \bar{x}_j) = u(\bar{x}_j, \bar{x}_i)$ is the estimated covariance associated with $\bar{x}_i$ and $\bar{x}_j$. The degree of correlation between $\bar{x}_i$ and $\bar{x}_j$ is characterized by the estimated correlation coefficient

$$r(\bar{x}_i, \bar{x}_j) = \frac{u(\bar{x}_i, \bar{x}_j)}{u(\bar{x}_i) u(\bar{x}_j)}$$

where $r(\bar{x}_i, \bar{x}_j) = r(\bar{x}_j, \bar{x}_i)$, and $-1 \le r(\bar{x}_i, \bar{x}_j) \le +1$. If the estimates $\bar{x}_i$ and $\bar{x}_j$ are independent, $r(\bar{x}_i, \bar{x}_j) = 0$, and a change in one does not imply an expected change in the other.

Equation (6) and its counterpart for correlated input quantities, Equation (7), both of which are based on a first-order Taylor series approximation of the measurand Y, express what is termed the law of propagation of uncertainty.

In most cases, the best available estimate of the expectation value $\langle X_i \rangle$ of an input quantity X_i that varies randomly, and for which independent observations have been obtained under the same conditions of measurement, is the arithmetic mean or average $\langle x_i \rangle$ of the observations. Thus, for an input quantity X_i determined from independent repeated observations, the standard uncertainty $u(\bar{x}_i)$ of its best estimate $\langle x_i \rangle$ is $u(\bar{x}_i) = s(\bar{x}_i)$, with $s^2\langle \bar{x}_i \rangle$ is the experimental variance or the estimate of the variance $\sigma^2\langle X_i \rangle$. For convenience, $u^2(\bar{x}_i) = s^2\langle \bar{x}_i \rangle$ and $u(\bar{x}_i) = s\langle \bar{x}_i \rangle$ are sometimes called a Type A variance and a Type A standard uncertainty, respectively.

In Tab. 1 a worksheet for the practical implementation of the measurand's uncertainty estimation from uncorrelated (independent) input quantities based on formula (4) to (6) is given.

Table 1. Worksheet for the estimation of the expanded uncertainty *U* of the output quantity (measurand) *Y* from estimates of independent uncorrelated input quantities X_i.

input quantity	estimate of input quantity	evalua-tion type	sensitivity coefficient c_i	standard uncertainty	square products
X_i	$\bar{x}_i$	A or B	$\frac{\partial f(\bar{x}_1, \bar{x}_2, \bar{x}_3, ...)}{\partial X_i}$	$u(\bar{x}_i)$	$\left(\frac{\partial f(\bar{x}_1, \bar{x}_2, \bar{x}_3, ...)}{\partial X_i}\right)^2 \cdot u^2(\bar{x}_i)$
output quantity / measurand	output estimate of measurand		sum of square products: $u_C^2(\bar{y})$		
			square root of the sum: $u_C(\bar{y})$		
Y	$\bar{y} = f(\bar{x}_1, \bar{x}_2, \bar{x}_3, ...)$		coverage factor k:		
			expanded uncertainty: $U(\bar{y}) = k \cdot u_C(\bar{y})$		

Quality management of radiation protection measurements

The state-of-the-art main source for laboratory quality management is the European standard EN ISO/IEC 17025:2005. In this standard the main conformity requirements of an appropriate QM system for testing and calibration laboratories are given:

- **Management requirements:** organisation, management system, document control, review of requests, tenders and contracts, subcontracting of tests and calibrations, purchasing services and suppliers, services to the customer, complaints, control of nonconforming testing and/or calibration work, improvements, corrective action, preventive action, control of records, internal audits, and management reviews.

- **Technical requirements:** general, personnel, accommodation and environmental conditions, test and calibration method validation, equipment, measurement traceability, sampling, handling of tests and calibration items, assuring the quality of tests and calibration results, reporting the results.

Especially for measurements in radiation protection and radiation safety the required measurement traceability is explained in IAEA (2008):

"7.28. To be sure that the measurement results will comply with international standards, each measurement device that has an influence on the results should be calibrated before being put into service and at defined intervals afterwards. The standards used for these calibrations should be traceable to the International System of Units (SI). In some cases — e.g. in connection with ^{222}Rn — the only means of providing confidence in measurements is through participation in suitable international intercomparison exercises.

7.29. Calibration services have to trace their standards and measuring instruments to the SI System by means of an unbroken chain of calibrations or comparisons linking them to relevant primary standards for the SI units of measurement. For measurement services, this traceability can be achieved by using a calibration service."

This metrological requirement must be strictly implemented in the QM system and assured in the day-to-day laboratory practice and documentation.

Conclusions

Measurements in radiation protection and radiation safety for the assessment of external and internal exposure need reliable measurement instruments and methods. To ensure high-quality measurements, an appropriate quality management system has to be implemented and followed in the day-to-day practice. Careful attention has to be paid to the use of appropriate quantities, the reasonable estimation of measurement uncertainties and the metrological traceability of the measurement instruments.

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External dosimetry and individual monitoring

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Abstract
Individual monitoring is required by international regulations to demonstrate the compliance of dose limits. This paper gives an introduction which types of personal dosemeters are in use and how these dosemeters are calibrated in the operational dose quantities $H_p(10)$ and $H_p(0.07)$. In addition type test requirements for different dosemeters stated in recent international standards (IEC) are summarised and compared.

General principles and requirements
The main principles for the radiation protection system are the following three issues:

- **principle of justification**
- **principle of optimization of protection**
- **principle of application of dose limits**

Based on the last principle monitoring of the individual exposure of persons constitutes an integral part of any radiation protection programme. Especially the principle of dose limitation requires reliable assessment of individual doses.

Legal requirements and international recommendations
The fundamental principles for the operational protection of occupationally exposed persons, apprentices and students are laid down in EU Council Directive 96/29/Euratom Basic Safety Standards (BSS) [(2)]. The requirements for dose assessments and the corresponding dose limits are mandatory specified in these BSS based on international recommendations by the International Commission on Radiological Protection (ICRP) and reports of the International Commission on Radiation Units and Measurements (ICRU).

Dose limits and dose quantities
In these regulations dose limits, dose constraints, and reference levels for different situations are stated. In the BSS, the requirements for dose assessment and for dose limits are given in terms of the protection dose quantities, effective dose, equivalent dose to the lens of the eye, equivalent dose to the extremities and equivalent dose to local skin. These protection quantities are difficult to assess and impossible to measure directly. The BSS state that operational dose quantities for external radiation, personal

dose equivalent for personal monitoring, and ambient dose equivalent and directional dose equivalent for area monitoring, are to be used for monitoring for operational protection purposes. All these measurements are required to be traceable to international recognise metrological institutions maintaining primary standards for the relevant basic dose quantities. This traceability is normally ensured by proper calibration of the dosemeter by an accredited calibration laboratory.

The relationship between the different quantities is implicitly given by the exact definition of those (Table 3). For practical reasons conversion factors are published (in most cases determined by MC calculation) for special well defined conditions or procedures (e.g. for the purpose of calibration in a well-defined set up).

One important difference between all these quantities is the fact that the definition considers different concepts: the basic physical quantities are defined in any point of the radiation field. The operational quantities are defined in a point in a special phantom (ICRU sphere or the human body of a person wearing a personal dosemeter) in an aligned and expanded radiation field (Figure 1). The protection quantities are doses averaged over an organ/tissue or weighted over the whole body of a person.

Table 1. Annual dose limits given in the BSS [2].

Limiting quantity (protection dose quantity)	**Exposed workers** (aged over 18)	**Apprentices and students** (aged between16 and 18)	**Public**
Effective dose	100 mSv / 5a and 50 mSv on a single year;	6 mSv	1 mSv
Equivalent dose for the lens of the eye	150 mSv	50 mSv	15 mSv
Equivalent dose for the skin and extremities	500 mSv	150 mSv	50 mSv

Table 2. Summary of the system of different types of dose quantities.

Dose quantities		
basic physical quantities	**operational quantities**	**protection quantities**
Fluence (m^{-2}) Kerma (Gy) Absorbed dose (Gy)	Ambient dose equivalent (Sv) Directional dose equivalent (Sv) Personal dose equivalent (Sv)	Equivalent dose for organs and tissues (Sv) Effective dose for the whole body (Sv)
Realised by primary standards	Measured with a calibrated routine dosemeter	Quantity for which dose limits are stated
Defined by ICRU		Defined by ICRP

The detailed definitions and explanations of the system of basic quantities, protection quantities and operational quantities are summarised in Table 3.

Table 3. Definition and SI units of dose quantities (definitions of ICRU and ICRP).

Definition	Equation	SI unit
The **fluence, Φ** is the quotient of d*N* by d*a*, where d*N* is the number of particles incident on a sphere of cross-sectional area d*a*	$\Phi = \frac{\mathrm{d}N}{\mathrm{d}a}$	m^{-1}
The **Kerma, *K*** is the quotient dE_{tr} by d*m,* where dE_{tr} is the sum of the initial kinetic energies of all charged ionizing particles liberated by uncharged ionizing particles in a volume element of mass d*m*	$K = \frac{\mathrm{d}E_{\mathrm{tr}}}{\mathrm{d}m}$	$J.kg^{-1}$ special name: gray (Gy)
The **absorbed dose, D** is the quotient of $\mathrm{d}\bar{\varepsilon}$ by d*m,* where $\mathrm{d}\bar{\varepsilon}$ is the mean energy imparted by ionizing radiation to matter of mass d*m*	$D = \frac{\mathrm{d}\bar{\varepsilon}}{\mathrm{d}m}$	$J.kg^{-1}$ special name: gray (Gy)
The **equivalent dose, $H_{T,R}$**, is the absorbed dose in an organ or tissue multiplied by the relevant radiation weighting factor, where $D_{T,R}$ is the absorbed dose averaged over the tissue or organ T, due to radiation R and w_R is the radiation weighting factor for radiation type R	$H_{T,R} = w_R \cdot D_{T,R}$	$J.kg^{-1}$ special name: sievert (Sy)
The **effective dose, E**, is a summation of the equivalent doses in tissue, each multiplied by the appropriate tissue weighting factor, where H_T is the equivalent dose in tissue T and w_T is the tissue weighting factor for tissue T.	$E = \sum_T w_T \cdot H_T$	$J.kg^{-1}$ special name: sievert (Sy)
The **dose equivalent, *H*** is the product of Q and D at a point in tissue, where D is the absorbed dose and Q is the quality factor at that point,	$H = Q \cdot D$	$J.kg^{-1}$ special name: sievert (Sy)
The **ambient dose equivalent, *H*(d)*** at a point in a radiation field is the dose equivalent that would be produced by the corresponding expanded and aligned field in the ICRU sphere at a depth, *d*, on a radius opposing the direction of the aligned field	$H^*(d)$	$J.kg^{-1}$ special name: sievert (Sy)
The **directional dose equivalent, *H´(d, Ω)*** 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 specific direction, Ω.	$H'(d,\Omega)$	$J.kg^{-1}$ special name: sievert (Sy)
The **personal dose equivalent, $H_p(d)$** at a point in a radiation field is the dose equivalent in soft tissue at an appropriate depth d, below a specified point on the body.	$H_p(d)$	$J.kg^{-1}$ special name: sievert (Sy)

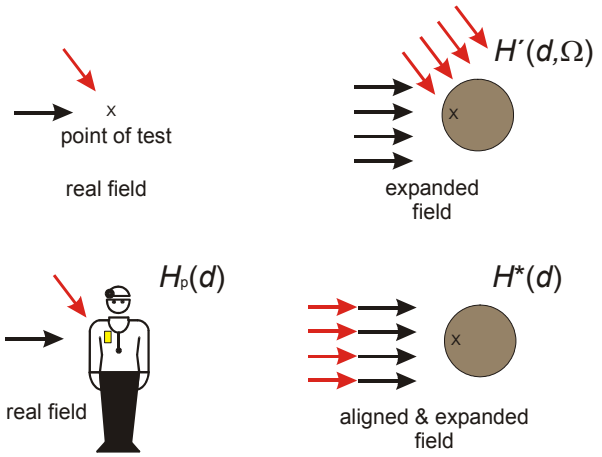

Figure 1. Schematic example of expanded and aligned radiation fields used in the definition of the operational quantities H*(d) and H'(d,Ω), defined in the ICRU-sphere. The quantity $H_p(d)$ is defined in the real field in a point in the human body wearing a dosemeter.

Quality factor Q and Radiation weighting factors w_R

The radiation weighting factor is defined as a factor w_R by which the tissue or organ absorbed dose $D_{T,R}$ is multiplied to reflect the higher biological effects of neutron, proton and alpha radiation compared to low LET-radiation. Considerable changes of the previous values were recently introduced by ICRP 103 [11]. Especially the weighting factor for neutrons was decreased by a factor of two for neutron energies below 1 MeV.

In the early 1960s, radiation weighting in the definition of radiological protection quantities was related to the radiation quality factor Q, as a function of LET. In ICRP 60 [8] the method of radiation weighting was changed in the definition of the protection quantities equivalent dose and effective dose. ICRP selected a general set of radiation weighting factors (w_R) that were considered to be appropriate for application in radiological protection. Today the quality factor Q however is still used for defining the operational quantities H*, H' and H_p.

Tissue weighting factors

The tissue weighting factor w_T is defined as a factor by which the equivalent dose to a tissue or organ H_T is multiplied in order to account for the relative stochastic detriment resulting from the exposure of different tissues or organs. Recent ICRP publication 103 [10] recommended in 2007 new tissue weighting factors for 14 organs and tissues and for a remainder (Figure 1).

New factors in ICRP 103

ICRP 110 [9] compares conversion coefficients for effective dose to air kerma for ICRP 60 and ICRP 103 values. For photons no significant differences especially in the lower energy region were stated. For neutrons mainly the change of the quality factor influences the conversion coefficient for energies below one MeV by a factor of 2.

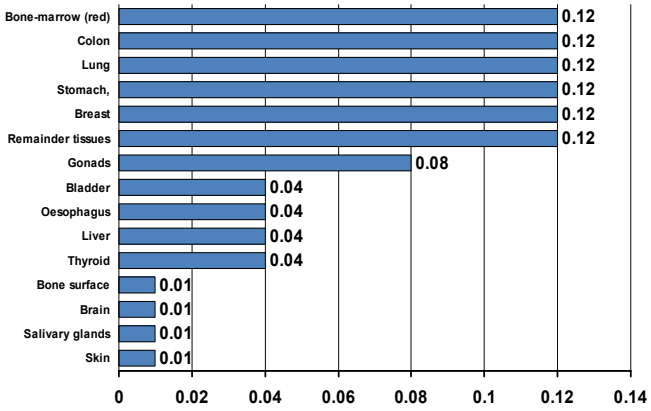

Figure 2. Tissue weighting factors (ICRP-103).

Assessment of the limiting quantities by the corresponding operational quantities

The *protection dose quantities* - defined by the ICRP - form the basis for dose limitation. These quantities are not directly measurable. The exposure can only be assessed by calculations when both the irradiation geometry and the radiation field parameters (type of radiation, intensity, spectral energy distribution etc.) are known. For routine monitoring, measurements with personal dosemeters calibrated in the operational quantities are performed. The suitability of this procedure to give a reasonable estimation of the appropriate limiting quantity was investigated by a joint task group of the ICRP and the ICRU. The results are published both as ICRP publication 74 [10] and ICRU report 57 [7]: in these publications the dose measured by the operational quantities were compared with the dose of the corresponding protection quantities (effective dose, equivalent dose, all weighting factors from ICRP 60). Some results are given in Figure 3. The influence of the changed neutron radiation weighting factor from ICRP 103 is added in addition. These diagrams show that the operational quantities for photons for irradiations from the front side (AP) always give a conservative estimate of the effective dose. This is valid for neutrons below 10 MeV too, when the new neutron radiation weighting factors from ICRP 103 are applied.

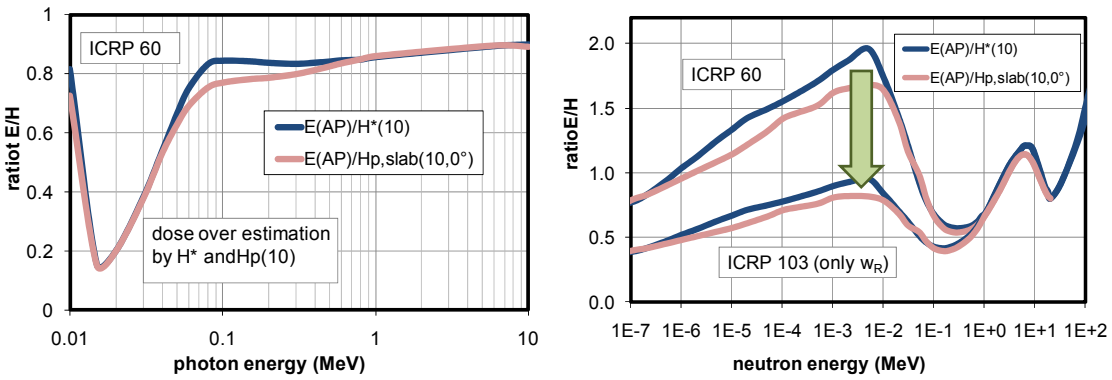

Figure 3. Ratio of the effective dose E and ambient dose equivalent H*(10) and personal dose equivalent $H_{p,slab}(10,0°)$ for photons and neutrons in AP irradiation geometry (ICRP60 weighting factors, except right figure where the new ICPR radiation factors for neutrons are used for the two lower lines) from [10] and [7].

Detection techniques

For routine individual monitoring active and passive dosimetry systems are used. Active systems give the worker an instant indication of both accumulated dose and sometiomes dose rate. If an alarm function is available these devices can be used simultaneously as an integrating dosemeter and as an alarm dosemeter. Active dosemeters are suitable also for short monitoring periods and low dose values. Recent investigations show that some active dosemeters are not capable to measure reliably doses in pulsed radiation fields. Significant under responses were reported [4]. These systems are therefore not recommended/allowed for specific tasks. Passive integrating systems are used normally for longer monitoring periods (weeks and months). These dosemeters are commonly issued by individual monitoring services (IMS). Implementation of a quality management system and sometimes formal accreditation are often national requirements for these services. Pulsed radiation fields generally do not cause problems for the applied measuring techniques. Table 4 shows an overview of the different dosimetry systems.

Table 4. Summary of different types of personal dosemeters

System type	External photon/beta radiation	Neutron radiation
Passive integrating systems	Photographic film Thermoluminescence (TLD) Optically Stimulated Luminescence (OSL) Radio-Photo-Luminescence (RPL) Direct Ion Storage (DIS)	TLD albedo dosimeters Track etch techniques Direct Ion Storage (DIS) bubble detectors
Active systems	APD (different detectors) (3) (Direct Ion Storage - DIS)	APD (different detectors)
Wearing position	trunk (whole body), extremity	trunk (whole body),
Dose quantity	$H_p(10)$, $H_p(0,07)$, (in future $H_p(3)$?)	$H_p(10)$,

Several international standards exists for area and personal dosemeters. A summary of most recent IEC standards and a comparison of the main requirements for these dosemeters are given in Table 5 and in [1]. Fulfilling these requirements limits the measuring uncertainties for the measured dose. Uncertainty estimations result in typical expanded uncertainty (k=2) of ±25% for photon doses at higher dose levels (some mSv) [10].

Table 5. recent international IEC standards for area and personal dosemeters.

IEC Standards	Radiation protection dosemeters		
	Personal dosemeter		Area dosemeter
	active	passive	active
Standard ID	IEC-61526 (2005)	IEC-62387-1 (2007)	IEC-60846 (2002)
Dose quantity	$H_p(10)$, $H_p(0,07)$		H*(10), H'(0,07), [K_a]
Minimum. dose range	100 µSv to 1 Sv		4 orders of magnitude
Minimum. dose rate range	0,5 µSv h^{-1} to 1 Sv/h	-	4 orders of magnitude
Energy- and angular response	-29%, +67% (photons) -35%, +300% (neutrons) (±60°)	-29%, +67% (±60°)	±40% (±45°)
Linearity	±15%	-9%, +11%	±20%
Coefficient of variation	15% to 5% (11 µSv) (photons) 25% to 5% (5,1 mSv) (neutrons)	15% to 5% (1,1 mSv)	15% to 5% (11 µSv)
Temperature dependence	±15%	-17%, +25%	±10% (inside) ±20% bis 30%
Intrinsic error	No requirements		±20%

Calibration in terms of the operational quantities

Calibration in general is the procedure to establish a relationship between the indicated value of a measuring instrument and the conventional true value of the quantity to be measured under well-defined conditions.

Radiation protection instruments for external radiation are calibrated in well-defined reference radiation fields. Recommended fields for calibration as well as the calibration methods are defined in different documents of the International Organization for Standardization (ISO) for photon [16], [17], [18], neutron [13], [14], [15] and electron radiations [19], [20], [21]. These reference radiation fields are generally defined in terms of the basic physical quantities since only these are directly traceable to primary standards.

Conversion coefficients

For the calibration in terms of the operational quantities, conversion coefficients are necessary relating the basic physical quantities to the operational quantities. Based on the results of radiation transport codes and appropriate mathematical models both ICRU [7] and ICRP [10] recommend a set of conversion coefficients for photons, neutrons and electrons. These published values are however restricted to monoenergetic radiation fields. When fields with spectral distributions are used a spectrum weighted conversion coefficients must be applied.

According to the definition of the operational quantities the conversion coefficients for $H^*(10)$, $H'(10, \Omega)$ and $H'(0.07, \Omega)$ are derived from calculated dose distributions in the ICRU sphere.

Since the personal dose equivalent is defined in the individual body of the person wearing the dosemeter, individual conversion coefficients would be required. Since this is not practical, conversion coefficients for $H_p(10)$ and $H_p(0.07)$ are calculated only for

three standard phantoms representing typical wearing positions (trunk, wrist and ankle, finger) of common used dosemeter types. All these phantoms are composed of ICRU-tissue with a density: 1 g/cm³ and a mass composition of: 76.2% oxygen, 11.1% carbon, 10.1% hydrogen and 2.6% nitrogen. For the calibration of personal dosemeters similar shaped calibration phantoms are used – due to practical reasons the material of these phantoms however is different (see Table 6)

Calibration method

The calibration method and the application of the conversion factor are carried out in principal in the following way:

- Selection of the dosemeter to be calibrated and the calibration conditions (calibration quantity, radiation quality, direction of incidence etc.)
- Selection of a suitable reference radiation field and a point of test (parameters like radiation quality, intensity, field size and homogeneity etc. needs to be considered)
- Determination (measurement) of the value of the appropriate basic physical quantity in the point of test (without the dosemeter and any calibration phantom)
- Calculation of the value of the required operational quantity by application (multiplication) of the corresponding conversion factor. This result is considered to be the conventional true value.
- Positioning of the dosemeter (and a phantom if required) with its reference point at the point of test, irradiating the dosemeter and reading the indicated value.
- Calculation of the calibration factor of the dosemeter defined as the ratio of the conventional true value (step 4) to the indicated value (step 5).

Table 6. Phantoms for the calibration of personal dosemeters

Calibration phantoms				
Name	**Shape and dimension**	**Material**	**Calibration quantity**	**Wearing position of dosemeter**
Water slab phantom	Slab: 30 cm x 30 cm x 15 cm	PMMA walls (100 mm, on front side 2,5 mm thick) filled with water	$H_p(10)$; $H_p(0,07)$	trunk
Pillar phantom	Cylinder: Diameter 7,3 cm Length 30 cm	PMMA walls (2,5 mm on circum-ference, 10 mm on faces sides thick) filled with water	$H_p(0,07)$	wrist, ankle
Rod phantom	Cylinder: Diameter 1,9 cm Length 30 cm	PMMA	$H_p(0,07)$	finger

Calibration phantoms

Depending on which dosemeter type is calibrated, special calibration phantoms must be used. The calibration of area monitoring instruments in terms of $H^*(d)$ or $H'(d, \Omega)$ is carried out free in air. No calibration phantoms are needed.

The calibration of personal dosemeters in terms of $H_p(d)$ is carried out always on an appropriate calibration phantom. The function of these phantoms is the production of backscattered radiation. Three different calibration phantoms are recommended

depending on the wearing position of the personal dosemeter. Details for the water slab, pillar and rod phantom are given in Table 6. When using calibration phantoms the corresponding conversion factor must be applied.

Summary

Individual monitoring is required by international regulations to demonstrate the compliance of dose limits. Whole-body and extremity dosemeters are calibrated in the operational dose quantities $H_p(10)$ and $H_p(0,07)$. The question if $H_p(3)$ should be used in addition is still under discussion. In most cases it is assumed that these operational quantities are a conservative estimate for the limiting quantities effective dose and equivalent dose. Recent changes of the radiation weighting factor for neutrons by a factor of two improved the applicability of $H_p(10)$ for the estimation of effective dose for neutrons. International standards for personal dosemeters state minimum requirements for these dosemeters. Fulfilling these requirements limits the measuring uncertainties for the measured dose and guarantees reliable dose measurements.

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Radiobiology – Evaluation of health risks after ionising radiation

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Abstract

For radiological protection two classes of effects are grouped: deterministic and stochastic effects. Deterministic effects with threshold doses are mainly tissue effects, dose response is well-known from clinical experience and animal experiments. For stochastic effects (cancer and hereditary effects) no threshold dose is assumed. The knowledge of mechanisms is necessary for solving these open questions. DNA damage and its repair, apoptosis, adaptive response, bystander effects, genomic instability are important in this connection and will be discussed. Radiations with different LET as well as biological systems with different sensitivities like prenatal development and genetic diseases will be of high interest.

Clinical auditing and quality assurance

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Abstract

In the jungle of the concepts of quality management, with a diversity of approaches and procedures in order to improve and maintain high quality, the meaning of concepts can easily be confused with each other and this has been particularly true for clinical audit. Clinical audit is a systematic review of the procedures in order to improve the quality and the outcome of patient care, whereby the procedures are examined against agreed standards of good practices.

While it certainly is an important part of the overall quality assurance activities, it should not be confused with either overall quality assurance or quality control programmes. It should neither be confused with external quality assessments such as accreditations or certification of quality systems, or with regulatory inspections.

In Europe, the Council Directive 97/43/EURATOM introduced the concept of clinical audit to medical RADIOLOGICAL (diagnostic radiology, nuclear medicine and radiotherapy)[1] procedures. Recently, the European Commission has published further guidelines on clinical audits (Report Radiation Protection 159). In this refresher course, the purpose, scope and methods of clinical auditing are presented in accordance with the EC guidelines. The role of clinical audit as a part of overall quality assurance is explained and its relation to the concepts of quality management clarified.

Introduction

It has been estimated [UNSCEAR, 2000] that worldwide there are about 2000 million x-ray studies, 32 million nuclear-medicine studies and over 6 million radiation therapy patients treated annually, and the numbers are constantly increasing. The use of radiation for medical diagnostic examinations contributes over 95 % of the man-made radiation exposure and is only exceeded by natural background as a source of exposure [UNSCEAR, 2000]. Furthermore, the dose to patients for the same type of examination differs widely between centres, suggesting that there is considerable scope for management of patient dose. About 40 to 60 % of all cancer patients are treated at least once during their disease with radiotherapy and more than half of these with curative intent. The difference between the dose that is required to achieve local control and the dose that can cause serious side effects is often quite small.

[1] "RADIOLOGICAL", written in capital letters, is used throughout this paper to denote all three fields of application: diagnostic radiology, nuclear medicine and radiotherapy. When only diagnostic radiology is concerned, the term is written in small letters ("radiological"). .

The above facts have stressed the importance of proper justification, optimization and quality assurance procedures in order to ensure the high quality and safety of RADIOLOGICAL procedures. A lot of attention has been paid to the quality management in the medical use of radiation, and worldwide there has been a tendency to establish quality systems and introduce appropriate quality audits. Among the regulatory efforts, the Council Directive 97/43/EURATOM (the MED directive; Article 2 and Article 6(4)) [European Commission, 1997] introduced the concept of *clinical audit* for medical RADIOLOGICAL procedures. According to the MED directive, clinical audits shall be implemented in accordance with national procedures.

In the jungle of the concepts of quality management, with a diversity of approaches and procedures in order to improve and maintain high quality, the meaning of concepts can easily be confused with each other and this has been particularly true for *clinical audit*. While it certainly is an important part of the overall quality assurance activities, it should not be confused with either quality assurance or quality control programmes. It should neither be confused with external quality assessments such as accreditations or certification of quality systems, or with regulatory inspections.

In this refresher course, the purpose, scope and methods of clinical auditing are presented in accordance with the EC guidelines. The role of clinical audit as a part of overall quality assurance is explained and its relation to the concepts of quality management clarified.

EC Guidelines on clinical audit

The concept of clinical audit is not a new one but has long been applied in many health care practices. In the European Commission (EC) directive 97/43/EURATOM (MED) [European Commission, 1997], this concept has been introduced for the assessment of medical RADIOLOGICAL practices. The MED-directive defined clinical audit as

> "a systematic examination or review of medical radiological procedures which seeks to improve the quality and the outcome of patient care, through structured review whereby radiological practices, procedures, and results are examined against agreed standards for good medical radiological procedures, with modifications of the practices where indicated and the application of new standards if necessary".

EU Member States are required to implement clinical audits "in accordance with national procedures" (Article 6.4 of the MED). Due to the high variation between the approaches of the Member States in its implementation, in 2007-2008, the EC conducted a special project to provide guidance on clinical auditing for an improved implementation of Article 6.4 of the MED.

The EC guidelines [European Commission 2009] provide a general framework for the Member States in order to establish sustainable national systems of clinical auditing of RADIOLOGICAL practices (diagnostic radiology, nuclear medicine and radiotherapy). The guidelines are sufficiently flexible and will enable the Member States to adopt the model of clinical audit with respect to their national legislation and administrative provisions.

The EC guidelines document introduces the basic principles of clinical audit (objectives, coverage, standards of good practice etc) aiming at clarifying its profound

meaning and recommended application. It defines the topics which should be covered while the criteria of good practice are discussed only on generic levels. It discusses the interrelation of clinical audit with other audit systems such as certification of quality systems, accreditation, peer review and quality award, and also its interrelation with regulatory control. Finally, it gives general advice for the practical implementation of audits, including organization of audits, recommendations for auditors, models of financing, national coordination and the role of scientific and/or professional societies and regulatory authorities.

It is important to recognize that the EC guidelines are not a legal requirement. The guidelines will only give recommendations and highlight some possible "national procedures" as expected by the MED.

Simultaneously with the European development, the IAEA has developed comprehensive audit programmes under the term of clinical audit [IAEA 2007; 2010]. Further, several dosimetry or quality audit programmes traditionally applied in the field of radiotherapy (e.g. by the IAEA [Izewska et al. 2004] or the ESTRO [Ferreira et al. 2000]) have been recognized to form an important part of clinical audit. There are also several national efforts of clinical auditing providing examples of potential approaches, e.g. the AuditLive system in the UK [The Royal College of Radiologists 2009] and the Finnish nationwide organization [Soimakallio 2003]. The general framework published in the EC guideline is well supported by these other international and national developments.

Essentials of clinical audit

The general purpose of any clinical audit is to

- improve the quality of patients' care
- improve the effective use of resources
- enhance the provision and organization of clinical services
- further professional education and training

With these objectives, clinical audit is an integral part of the overall quality improvement process and should be considered as an integral part of quality management and clinical governance. In the conceptual frame of quality concepts, clinical audit is a type of quality audit, which is one of the methods to implement overall quality assurance, which itself is one of the tools of quality management as shown in Fig. 1.

Clinical audit should cover the whole clinical pathway, and address the three main elements of the RADIOLOGICAL practices: structure, process, and outcome For RADIOLOGICAL procedures, the priorities should be as shown in Table 1. Clinical audits should address both the critical issues of the radiation protection for the patient and key components of the overall quality system.

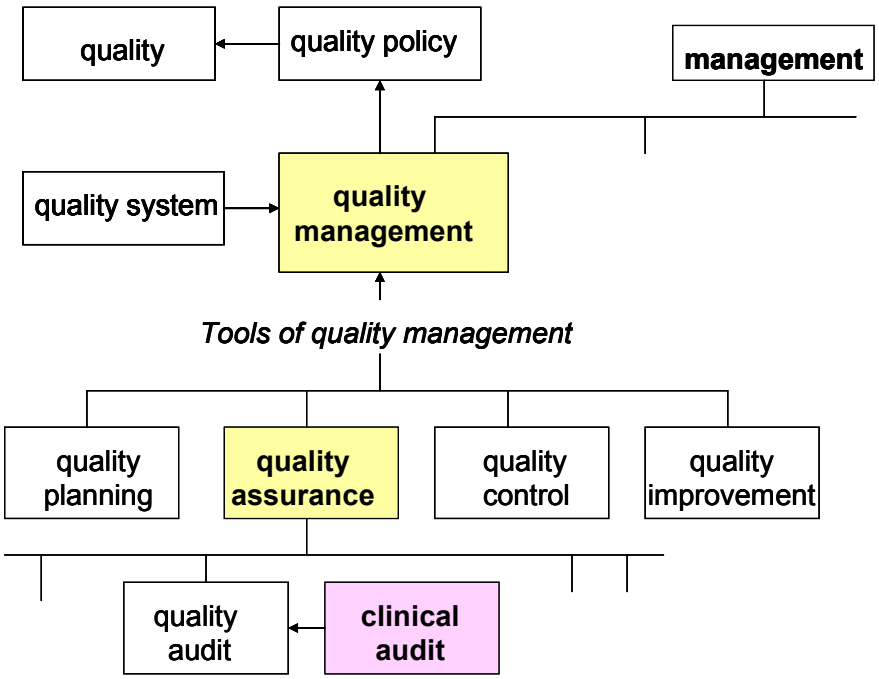

Fig. 1. Clinical audit in the "jungle" of quality management concepts; quality management scheme based on EN ISO 8402 [ISO 1995].

Table 1. The priorities of clinical audit of RADIOLOGICAL practices.

Structure	The mission of the unit for RADIOLOGICAL practices Lines of authorities and radiation safety responsibilities Staffing levels, competence and continuous professional development of staff, in particular for radiation protection Adequacy and quality of premises and equipment
Process	Justification and referral practices, including referral criteria Availability and quality of examination and treatment guidelines (protocols, procedures) Optimization procedures Patient dose and image quality in diagnostic radiology and nuclear medicine procedures, and comparison of patient dose with nationally accepted reference levels Procedures for dose delivery to the patient in radiotherapy (beam calibrations, accuracy of dosimetry and treatment planning) Quality assurance and quality control programmes Emergency procedures for incidents in use of radiation Reliability of information transfer systems
Outcome	Methods for the follow-up of outcome of examinations and treatment (short term and long term)

Clinical audit aims at continuous improvement of the medical practices. Therefore, it should be carried out regularly and it should be ensured that the audit cycle is

completed (Fig.2). The general audit cycle consists of selecting a standard of good practice, assessing and comparing local practice with accepted standards, implementing change when necessary, and re-auditing after a certain time. Regular re-audits will improve the quality or give reassurance that a good quality is maintained.

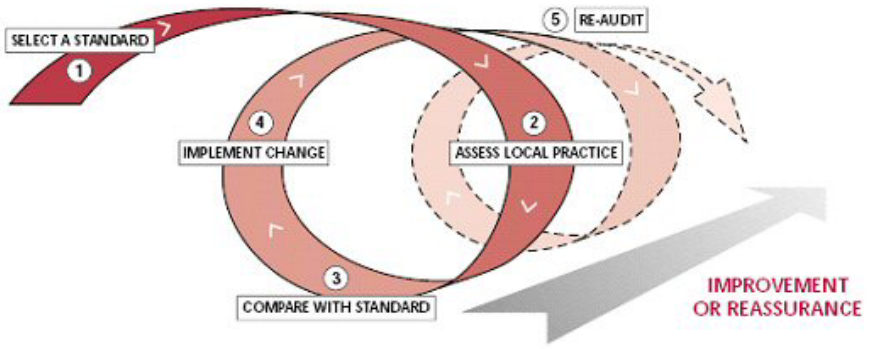

Fig.2 The audit cycle. Reprinted from Goodwin R, de Lacey G, Manhire A (eds). Clinical Audit in Radiology: 100+ Recipes, 1996 by permission of The Royal College of Radiologists.

It is evident from the definition that clinical audit is a truly multi-disciplinary, multi-professional activity. It must be carried out by auditors with extensive knowledge and experience of the radiological practices to be audited, i.e., they must generally be professionals involved in clinical work within these practices. Further, the general understanding of the concept "audit" implies that the review or assessment is carried out by auditors independent of the organizational unit or practice to be audited.

Both internal audits and external audits should be implemented. These should be of equal importance and supplement each other. *Internal* audits are undertaken within a given health care setting by staff from the same institution, while the audit findings can be also externally reviewed. In small health care units, internal audits would rather be self-assessments. *External* audits involve the use of auditors who are independent of the radiology department/institution. External audits may help identify other, unrecognised areas for improvement. They are needed to remove possible “blindness” of internal experts to recognize weaknesses of own unit and to give more universal and broader perspectives. External auditors should also possess better benchmarking skills in relation to the assessment.

The *standards of good practice* should be derived from evidence based data, long term experience and knowledge gained. In practice, these can be adopted from legal requirements, results of research, consensus statements, recommendations by learned societies or local agreements (if there is no other more universal reference). International medical/scientific/professional societies could play an important role in developing such standards. In a case where a consensus of the good practice is not easily achieved, the criteria of good practice adopted should be regarded as giving only preliminary orientation, and the results of audit should then be used like a benchmarking tool to achieve improved evidence and possible adjustment of the chosen criteria.

Clinical audits can be of various types and levels, either reviewing specific critical parts of the RADIOLOGICAL process (partial audit) or assessing the whole process (comprehensive audit). Dosimetry audit is an important example of partial audits and should be within the scope of a comprehensive clinical audit, as assured dosimetry is a vital component of accurate clinical practice. The comprehensive clinical audit must include the full patient pathway from referral to follow up.

Audits can address various "depths" of the procedure, from generic features to details of a given examination or treatment. Auditing the detailed practice for a given examination or treatment can usually mean only a few selected processes per audit run. Full details of the procedures should be assessed at least for the items of the procedure where a reasonable consensus on a good practice can be achieved.

Before starting the clinical audit the critical areas should be identified and the objectives agreed. For internal audits, the objectives are set by the management of the health care unit to be audited. For external audits, the detailed objectives should be agreed between the auditing organization and the unit to be audited, and should be based on any legal requirements on audit programmes, as well as on any recommendations by national coordinating organizations or by health professional and/or scientific societies when available.

The practical organizing of *external* clinical audits can be through site visits of an audit team or, for a limited part of practices with relevant documented or measurable data, by the collection of data via mail or internet, with central assessment of the data. Examples of the latter technique are the assessment of the quality of CT referrals by mailed questionnaire [Almen et al., 2009] and a national audit of provision of MRI services [Barter et al., 2008]. For comprehensive audits, a site visit is needed and should comprise a series of interviews, observations, document and data reviews, measurements, collection of data samples and analysis. Due to the multidisciplinary nature of the audit, a team of auditors is usually needed, comprising different professionals - radiologist, medical physicist, radiographer etc - depending on the scope of the audit. Besides the basic clinical competence, the auditors should receive specific training on the general audit procedure and techniques, as well as the agreed audit programme and the criteria of good practices to be applied.

Once the clinical audit has been completed and the auditor's report with recommendations is available to all staff, the unit should respond to the recommendations with an agreed timeline for improvement. This is important not only to achieve maximum benefit from the audit but also to retain the respect and motivation of the staff for subsequent re-audits.

In summary, clinical audit is an important tool of quality improvement and can have a major impact on developing the medical RADIOLOGICAL practices in compliance with the most recent data on good practices. The audits will yield multiple benefits to the health care system such as:

- improvement of practice
- recognition for quality and awareness of good practices
- recognition of outdated practice
- motivation of staff to increase quality
- improvement of local standards and adherence to national standards
- prevention against litigation

- improvement of communication within the institution,
- revealing weak points and
- promoting development of quality systems.

Relationship with other audits

While it is obvious that clinical audit has some similarities with other activities for the development and control of RADIOLOGICAL practices, it is imperative not to confuse it with such activities as

- research
- quality control program for equipment
- regulatory inspection nor any other regulatory activity.

Further, clinical audit should not be confused with other external quality assessments systems, broadly categorized as [European Commission 2009]:

- certification by International Standards Organization (ISO)
- accreditation
- professional peer review –based schemes
- award seeking such as European Quality Award and their national variants (i.e. European Foundation for Quality Management (EFQM) Excellence Model).

Alhough the methodology and terminology of these external review systems differ, a constant movement towards collaboration and convergence of those models has been observed [European Commission 2009]. The purpose of these other activities should be properly understood, and conversely to duplicating any efforts, clinical audits should be developed to supplement the other activities.

Research is a systematic investigation to increase the sum of our knowledge. For clinical audit, the aim of research is to determine what a good practice is, while audit itself should ask the question: "Are we actually following good practice?"

The *quality control* of RADIOLOGICAL equipment aims at ensuring adequate performance characteristics and safe operation throughout the lifetime of the unit, and the responsibility for the checking lies solely on the health care organization, the user of the equipment or the practitioner. Any audit or external control does not remove or release this responsibility while, however, can give desirable confidence on the results of the quality control.

A legislative and statutory framework is needed in each country to regulate the safety of facilities and activities, including medical use of radiation. The *regulatory requirements* will generally depend on the level of risk or complexity associated with the medical use, as determined by the regulatory body. In radiotherapy, for example, the high risk profile justifies maximum regulatory efforts, including regulatory inspections. The purpose of such an inspection is to verify that various detailed requirements for radiation protection are being met. The methods of verification can include both documentary assessments and verification measurements. While the verification procedures can be partly similar to some clinical audit procedures, the basis of the review and the use of the results are quite different (Table 2) and clinical audits should not be confused with *regulatory inspections*.

Table 2. Main differences between clinical audit and regulatory inspection.

	Clinical audit	Regulatory inspection
Focus of review	"Agreed standards for good medical RADIOLOGICAL procedures". These are often not requirements but recommendations to the users. There may be more than just one agreed standard.	Legislative and statutory framework (laws, statutes and other regulations). These are unambiguous and usually binding requirements to the users.
Use of the results	Auditor's report, with the findings and recommendations, is given to the user. *The auditor cannot enforce any actions*, but the actions are solely decided by the user.	The non-compliance with specified conditions and requirements leads to *enforcement actions* by the regulatory body. The regulatory inspector may impose on the spot corrective requirements to the user.

A quality system in conformance with international quality standards such as the ISO 9001 [ISO, 2000] is generally considered a good basis for the overall quality management in a RADIOLOGICAL unit. To ensure that the local quality system conforms to the specifications of the quality standard, *certification* by a certification body can be acquired by special quality (system) audits carried out regularly by quality system experts. These experts are usually not clinical experts and the assessment does not address the quality of the clinical practice but solely its conformance with the general quality rules. Conversely, in clinical audits, clinical experts concentrate on the evaluation of the conformance of the clinical practice to the agreed good practice. The difference between quality system audits and clinical audits is demonstrated further in Fig. 3.

While clinical audit is not a systematic assessment of the quality system, the quality system documentation is a prerequisite for meaningful evaluation of the practices against standards of good practices. Therefore, an important supplementary benefit of clinical audits has been noted to be that their implementation has speeded up the development of appropriate quality systems in health care units.

The system of *accreditation* deals with the competence of the unit to perform certain practices, in accordance with given standards. This can become very close to the aims of clinical auditing, while usually the scope is narrower and limited to the definite standards. In RADIOLOGICAL practices, such standards are either very general or not very common and limit the applicability of accreditation in the sense of replacing the clinical audits.

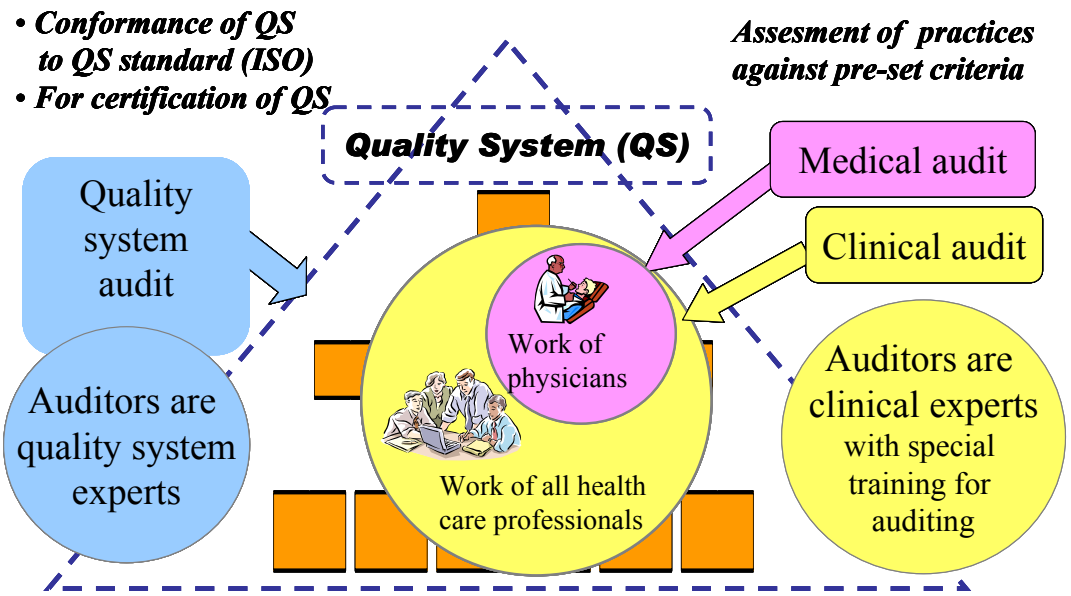

Fig. 3. Quality audits for a health care organization: Clinical audit versus quality system audit.

Conclusions

Clinical audit is a systematic review of the procedures in order to improve the quality and the outcome of patient care, whereby the procedures are examined against agreed standards of good practices. It is an important tool of quality improvement and can have a major impact on developing the medical RADIOLOGICAL practices in compliance with the most recent data on good practices. While it is an important part of the overall quality assurance activities, it should not be confused with either quality assurance or quality control programmes. It should neither be confused with external quality assessments such as accreditations or certification of quality systems, or with regulatory inspections.

Acknowledgements

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Natural radiation environment and NORM

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Abstract

We are exposed to natural radiation all the time and everywhere. However, the types and levels of exposure vary considerably from "insignificant background" to situations where workers or members of the public receive doses which would not be accepted in any planned uses of radiation sources. In some cases, human activities have modified or even caused new pathways of exposure to natural radiation sources.

The refresher course will introduce the most common sources of natural radiation, as well as, types and levels of exposures delivered to workers and the members of the public. It will also highlight key aspects and challenges in reducing exposures and in introducing regulatory requirements for e.g. radon in workplaces, natural radioactivity in building materials and industries involving NORM (Naturally Occurring Radioactive Material).

The aim of the refresher course is to provide a general overview on natural radiation environment and NORM and to highlight practical aspects in managing related exposures.

Introduction

This refresher course introduces the most common sources of natural radiation, as well as, types and levels of exposures delivered to workers and the members of the public. The aim is also to highlight practical aspects in managing related exposures by means of regulation and regulatory control.

The United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) has provided a comprehensive overview on sources and levels of exposures to natural radiation. The following chapters are mostly summaries condensed from the UNSCEAR 2000 report (UNSCEAR, 2000). Because condensing always imply personal preferences it is recommended to refer to the full report for complete information.

The current EU Basic Safety Standards Directive (BSS, 1996) covers work activities which involve the presence of natural radiation sources. The EU Member States are required to identify by means of surveys or by any other appropriate means work activities which lead to significant exposure of workers or members of the public. Corrective measures to reduce exposures and a system of radiological protection shall be applied to occupational and public exposures, where appropriate.

In this refresher course, the also Finnish experiences in implementing the above mentioned EU provision are discussed (separate Power-Point slides) as examples of imposing regulatory requirements and controls on exposures to natural radiation.

Cosmic radiation

Cosmic rays

The earth is continually bombarded by high-energy particles that originate in outer space. These cosmic rays interact with the nuclei of atmospheric constituents, producing a cascade of interactions and secondary reaction products that contribute to cosmic ray exposures. The cosmic ray interactions also produce a number of radioactive nuclei known as cosmogenic radionuclides.

Galactic cosmic rays consist of a nucleonic component, which in aggregate accounts for 98% of the total, and electrons, which account for the remaining 2%. The nucleonic component is primarily protons (88%) and alpha particles (11%), with the remainder heavier nuclei. These primary cosmic particles have an energy spectrum that extends from 10^8 eV to over 10^{20} eV.

Another component of cosmic rays is generated near the surface of the sun by magnetic disturbances. These solar particle events are comprised mostly of protons of energies generally below 100 MeV and only rarely above 10 GeV. These particles can produce significant dose rates at high altitudes, but only the most energetic affect dose rates at ground level.

The most significant long-term solar effect is the 11-year cycle in solar activity, which generates a corresponding cycle in total cosmic radiation intensity. At times of maximum solar activity the field is at its highest and the galactic cosmic radiation intensity is at its lowest.

The 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 lower-Z nuclei. The secondary nucleons in turn generate more nucleons, producing a nucleonic cascade in the atmosphere.

The pions generated in nuclear interactions are the main source of the other components of the cosmic radiation field in the atmosphere. The neutral pions decay into high-energy photons, which produce high-energy electrons, which in turn produce photons etc., thus producing the electromagnetic, or photon/electron, cascade.

Exposure at ground level

The world average effective dose from cosmic radiation is 380 µSv/a. Dose is influenced by geomagnetic latitude, altitude, solar activity and shielding effect of buildings.

At high altitude areas doses may be significantly higher than at sea level. For example, in La Paz, Bolivia, which is located at the altitude of 3,9 km, the effective dose is about 2 mSv/a.

The effective dose rate at the equator is about 34 nSv/h and at polar regions about 43 nSv/h. Variation during the 11-year solar activity period is about 10%. At high altitudes the variation is more significant than at sea level. The shielding factor for

buildings vary between 0.4 - 1, the average being about 0.8. Values like 0.4 may apply to some lower storeys of substantial concrete buildings. For houses with minimum vertical shielding (e.g. small wooden houses) the shielding factor is close to 1.

Exposures at aircraft altitudes

Aircraft passengers and crew are subject to cosmic radiation exposure rates much higher than the rates at ground level. Total exposure on a given flight depends on the particular path taken through the atmosphere and flight time; that is, it depends on the duration of exposure at various altitudes and latitudes. Complicating the situation is the fact that the exposure associated with any flight path may vary with time.

There are two possible approaches to dose assessment under these circumstances: area and/or individual monitoring for each flight, and determining the radiation fields as a function of time and space and calculating the effective dose for any flight path.

Commercial subsonic aircraft generally have cruising altitudes of 7 to 12 km. The annual number of hours flown by crew members varies. The range is 300 - 900 hours per year, with an average of about 500.

For altitudes of 9 - 12 km at temperate latitudes, the effective dose rates are in the range 5 - 8 μSv/h, such that for a transatlantic flight from Europe to North America, the route dose would be 30 - 45 μSv. At equatorial latitudes, the dose rates are lower and in the range of 2 - 4 μSv/h.

Cosmogenic radionuclidies

The interactions of cosmic-ray particles in the atmosphere produce a number of radionuclides, including ^{3}H, ^{7}Be, ^{14}C, and ^{22}Na. Production is greatest in the upper stratosphere, but some energetic cosmic-ray neutrons and protons survive in the lower atmosphere, producing cosmogenic radionuclides there as well.

Except for ^{3}H, ^{14}C, and ^{22}Na, which are elements with metabolic roles in the human body, the cosmogenic radionuclides contribute little to radiation doses and are mainly of relevance as tracers in the atmosphere and in hydrological systems after deposition. The annual effective doses from cosmogenic radionuclides are about 12 μSv from ^{14}C, 0.15 μSv from ^{22}Na, 0.01 μSv from ^{3}H, and 0.03 μSv from ^{7}Be.

Terrestrial radiation

Naturally occurring radionuclides of terrestrial origin (also called primordial radionuclides) are present in various degrees in all media in the environment, including the human body itself. Only those radionuclides with half-lives comparable to the age of the earth, and their decay products, exist in significant quantities in these materials.

Irradiation of the human body from external sources is mainly by gamma radiation from radionuclides in the ^{238}U and ^{232}Th series and from ^{40}K. Some other terrestrial radionuclides, including those of the ^{235}U series, ^{87}Rb, ^{138}La, ^{147}Sm, and ^{176}Lu, exist in nature but at such low levels that their contributions to the dose in humans are small.

External exposures outdoors

The levels of external exposures from terrestrial radionuclides are related to the types of rock from which soil originate. Higher radiation levels are associated with igneous

rocks, such as granite, and lower levels with sedimentary rocks. There are exceptions, however, as some shales and phosphate rocks have relatively high content of radionuclides.

Three components of the external radiation field, namely from the gamma-emitting radionuclides in the ^{238}U and ^{232}Th series and ^{40}K, make approximately equal contributions to the externally incident gamma radiation dose to individuals in typical situations both outdoors and indoors.

The activity concentration of ^{40}K in soil is an order of magnitude higher than that of ^{238}U or ^{232}Th. The worldwide median values are 400, 35, and 30 Bq/kg, and the population weighted values are 420, 33, and 45 Bq/kg for ^{40}K, ^{238}U, and ^{232}Th, respectively. The population-weighted average absorbed dose rate in air outdoors from terrestrial gamma radiation is 59 nGy/h and typical variation at different locations is from 10 to 200 nGy/h. Some areas with much higher levels exist, for example Guarapari in Brazil and Kerala in India (thorium baring monazite sand deposits).

In addition to variations from place to place, the ambient background gamma dose rate in air at any specific location is not constant in time. It is subject to fluctuation, in particular from the removal of radon progeny in air by rainfall, soil moisture and snow cover. Washout and rainout of radon progeny from air may result in the short-term enhancement, by 50% - 100%, of the gamma ray dose rate in air. The extent of the elevation depends on rain interval, as well as, the rainfall amount. Snow cover depresses the background level by about 1% for each centimetre of snow.

External exposures indoors

Indoor exposure to gamma rays, mainly determined by construction materials, is inherently greater than outdoor exposure if earth materials have been used; the source geometry changes from half-space to a more surrounding configuration indoors. When the duration of occupancy is taken into account, indoor exposure becomes even more significant. Buildings constructed of wood add little to indoor exposures, which may then be comparable to outdoor exposures.

The world-wide population-weighted average is 84 nGy/h with national averages ranging from 20 to 200 nGy/h. The indoor to outdoor ratios range from 0.6 to 2.3, with a population-weighted value of 1.4. Thus indoor exposures (absorbed dose rate in air from terrestrial gamma radiation) are, in general, 40% greater than outdoor exposures.

Effective dose

To estimate annual effective doses, account must be taken of (a) the conversion coefficient from absorbed dose in air to effective dose and (b) the indoor occupancy factor. The average numerical values of those parameters vary with the age of the population and the climate at the location considered. Assuming 0.7 Sv/Gy for the conversion coefficient from absorbed dose in air to effective dose received by adults and 0.8 for the indoor occupancy factor, i.e. the fraction of time spent indoors and outdoors is 0.8 and 0.2, respectively, the components of the annual effective dose are determined as follows:

Indoors: $84\ \text{nGy/h} \times 8{,}760\ \text{h} \times 0.8 \times 0.7\ \text{Sv/Gy} = 0.41\ \text{mSv}$

Outdoors: $59\ \text{nGy/h} \times 8{,}760\ \text{h} \times 0.2 \times 0.7\ \text{Sv/Gy} = 0.07\ \text{mSv}$

The resulting worldwide average of the annual effective dose is 0.48 mSv, with the results for individual countries being generally within the 0.3 - 0.6 mSv range. For children and infants, the values are about 10% and 30% higher, in direct proportion to an increase in the value of the conversion coefficient from absorbed dose in air to effective dose.

Internal exposures other than radon

Internal exposures arise from the intake of terrestrial radionuclides by inhalation and ingestion. Doses by inhalation result from the presence in air of dust particles containing radionuclides of the ^{238}U and ^{232}Th decay chains. The dominant component of inhalation exposure is the short-lived decay products of radon, which because of their significance are considered separately under section "Radon and decay products".

Doses by ingestion are mainly due to ^{40}K and to the ^{238}U and ^{232}Th series radionuclides present in foods and drinking water. It should be noticed that potassium concentration in the body is under homeostatic control and, therefore, the dose due to ^{40}K remains essentially constant, irrespectively of the amount of intake.

Inhalation intake of radionuclides of the ^{238}U and ^{232}Th decay chains (other than radon and its decay products) makes only a minor contribution to internal exposure. These radionuclides are present in air because of resuspended soil particles. Typical concentrations in air are 1 - 2 μBq/m^3 corresponding to a dust loading of 50 μg/m^3 and ^{238}U and ^{232}Th concentrations in the dust in the range 25 - 50 Bq/kg. The age-weighted annual effective dose is about 6 μSv from inhalation of uranium- and thorium-series radionuclides in air.

Ingestion intake of natural radionuclides depends on the consumption rates of food and water and on the radionuclide concentrations. Concentrations of naturally occurring radionuclides in foods vary widely because of the differing background levels, climate, and agricultural conditions that prevail. There are also differences in the types of local foods that may be included in the categories such as vegetables, fruits, or fish.

The total effective dose from inhalation and ingestion of terrestrial radionuclides is 310 μSv, of which 170 μSv is from ^{40}K and 140 μSv is from the long-lived radionuclides in the uranium and thorium series. It should be noted, however, that variations may be very significant. Some ground waters and waters from bed rock (drilled wells) may contain very significant levels of natural radionuclides including radon. Levels leading to doses up to several tens of mSv have been observed.

Radon and decay products

Radon (^{222}Rn) and thoron (^{220}Rn) are the gaseous radioactive products of the decay of the radium isotopes ^{226}Ra and ^{224}Ra, which are present in all terrestrial materials. Radon and its short-lived decay products in the atmosphere are the most important contributors to human exposure from natural sources.

The half-life of ^{222}Rn is 3.824 d. It has four short-lived decay products: ^{218}Po (3.05min), ^{214}Pb (26.8min), ^{214}Bi (19.9 min), and ^{214}Po (164 μs). Both polonium isotopes are alpha-emitters. The relatively short half-life of thoron (55.6 s) means that it does not have much time to travel from its production site to the immediate environment of human beings.

The evaluation of exposure to radon and thoron and their decay products must take account of the actual activity concentrations of the various alpha-emitting radionuclides in the decay series. An equilibrium factor F is defined that permits the exposure to be estimated in terms of the potential alpha energy concentration (PAEC) from the measurements of radon gas concentration. This equilibrium factor is defined as the ratio of the actual PAEC to the PAEC that would prevail if all the decay products in each series were in equilibrium with the parent radon. However, it is simpler to evaluate this factor in terms of an equilibrium equivalent radon concentration, C_{eq}, in the following manner:

$$F = C_{eq}/C_{rn}$$

$$C_{eq} = 0.105\ C_1 + 0.515\ C_2 + 0.380\ C_3\ (^{222}Rn \text{ series})$$
$$C_{eq} = 0.913\ C_1 + 0.087\ C_2\ (^{220}Rn \text{ series})$$

where the symbols C_1, C_2, and C_3 are the activity concentrations of the decay progeny, namely ^{218}Po, ^{214}Pb, and ^{214}Bi, respectively, for the ^{222}Rn series and ^{212}Pb and ^{212}Bi (C_1 and C_2) for the thoron series. The constants are the fractional contributions of each decay product to the total potential alpha energy from the decay of unit activity of the gas.

In this way, a measured radon concentration can be converted to an equilibrium equivalent concentration (EEC) directly proportional to PAEC. This provides a measure of exposure in terms of the product of concentration and time. The EEC can be converted to the PAEC, when desired, by the relationships 1 $Bq/m^3 = 5.56 \cdot 10^{-6}$ mJ/m^3 = 0.27 mWL (222Rn) and 1 $Bq/m^3 = 7.6 \cdot 10^{-5}$ mJ/m^3 = 3.64 mWL (thoron).

The value of the equilibrium factor may vary considerably depending on the prevailing circumstances, even from as low as 0.1 up to 0.9. However, typical value of the equilibrium factor in indoors is 0.4 and outdoors 0.6. If equilibrium factor is not determined with measurements, these values can usually be assumed.

Radon as source of indoor radon and exposure is discussed in a separate refresher course “Indoor radon sources, remediation and prevention in new construction”. Therefore, characteristics of radon, its behaviour and exposures caused are not further discussed here.

Industrial activities

There are a number of circumstances in which materials containing natural radionuclides are recovered, processed, used, or brought into position such that radiation exposures result. This human intervention causes extra or enhanced exposures.

Industry uses many different raw materials that contain naturally occurring radioactive materials, sometime abbreviated NORM. These raw materials are mined, transported, and processed for further use. The consequent emissions of radionuclides to air and water lead to the eventual exposure of humans. Industries and materials for which NORM-issues have been of interest are, for example:

- Phosphate processing
- Metal ore processing
- Zircon sands

- Titanium pigment production
- Fossil fuels
- Oil and gas extraction
- Building materials
- Thorium compounds
- Scrap metal industry

The natural radionuclides present in the raw materials or wastes of these industries are those of the ^{238}U and ^{232}Th series. Releases are mainly to air or water, although landfills after dredging or wastes disposed on land may also provide pathways of exposure.

Both external and internal exposures may result from naturally occurring radionuclides released by industrial activities. In general, installations are located away from residential areas, and because external radiation levels decrease with distance from the plant, local residents are not significantly exposed. The workers, however, may receive low doses in connection with ore stock piles or waste deposits. Estimated and measured doses are in the range 0.1 - 300 μSv/a from direct exposures.

Inhalation contributes to exposure only in the vicinity of the plant, particularly with mineral sands-processing plants. Doses depend on distance and could be up to 50 μSv/a for office workers in a building just outside the plant site. In most other industries the contributions to total dose appear to be negligible.

Because most food products consumed by individuals are produced in large agricultural regions, possible dose from ingestion of radionuclides are small. For a typical situation, a small population in the vicinity of an elementary phosphorus plant, the calculated dose would be of the order 100 μSv/a. More generally, the estimated doses would be 1 - 10 μSv/a. Ingestion doses that could result from discharges of wastes to water are negligible compared to those by the other pathways.

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Internal dosimetry and individual monitoring

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Abstract
A description is given of the main principles and methods of internal dosimetry. Factors that are unique to internal dosimetry (e.g. protraction of doses over time) are discussed, and the implications for the dose quantities used in internal dosimetry are explained. The principles underlying the use of biokinetic models are discussed, and ICRP's models of the respiratory tract and human alimentary tract described. As an example of the current approach to systemic modelling of the elements, the biokinetic model for plutonium is described. Lastly, the use of individual monitoring for the retrospective assessment of internal doses is discussed.

Introduction
Internal dosimetry addresses the determination of radiation doses resulting from the intake of radioactive materials into the body by inhalation, ingestion, absorption through the skin or *via* wounds. Internal doses may be determined for particular organs or tissues in the body, or for the whole body. It is not possible to measure internal doses directly, although the activity of a radionuclide in the body can be measured. *Individual monitoring* for internal contamination is the direct or indirect measurement of the amount of a radionuclide in the body, and the retrospective assessment of internal doses using the results of such measurements.

In this paper, the main steps in the calculation of internal dose are first summarised. Subsequent sections describe the dose quantities used in internal dosimetry, the method used to calculate internal doses, and the biokinetic models used to describe the movement of radionuclides through the respiratory and gastro-intestinal tracts. As an example of the current approach to modelling the biokinetics of radionuclides after uptake from blood, the biokinetic model used to predict the systemic behaviour of plutonium is described. Finally, individual monitoring for internal contamination is discussed.

This paper aims to provide a summary of the basics of internal dosimetry and individual monitoring for internal contamination, and to give an account of recent developments in the field. It is aimed at radiation protection professionals who may have limited experience of internal dosimetry, or who wish to learn about recent advances.

Assessment of Internal Doses

The risk or severity of any adverse effects on health resulting from an intake of a radionuclide is in almost all cases determined by making an assessment of the internal radiation dose. Risks to health are then evaluated making use of information on the relationship between risk and dose such as that provided by epidemiology studies, notably the follow up of Japanese A-bomb survivors.

Since internal doses cannot be measured directly, they have to be calculated using mathematical models describing the behaviour of radionuclides in the body. The main steps in a calculation of internal dose resulting from a specified intake (i.e. a prospective assessment of dose) are as follows:

- Modelling of the **intake** process to determine the location(s) at which the radionuclide is initially deposited or through which it transits (e.g. regions of the respiratory tract; organs in the gastro-intestinal tract), and the amounts deposited in those locations
- Modelling of the **clearance** of the radionuclide from the initial site of deposition, to determine the amount cleared and rates of clearance
- Modelling of the **uptake** of the radionuclide from the initial deposition site to the blood and then to the rest of the body
- Determination of the amounts then **deposited** in the various organs and tissues of the body, and the rates at which the radionuclide is removed from these organs and tissues
- Calculation of the amounts of **energy deposited** in each target organ as a result of the decay of the radionuclide in each source organ
- Calculation of the **absorbed dose** received by each organ over time, from the amount of energy deposited per unit mass of the organ
- Calculation of the **equivalent dose** received by each organ over time, determined by applying the appropriate radiation weighting factors to the absorbed dose for each radiation type, so allowing for the different effectiveness of each radiation type in causing stochastic effects
- Calculation of the **committed equivalent dose** for each organ, by summing the equivalent dose received from the intake over a fixed time period (e.g. 50 years for occupational exposure)
- Calculation of the committed **effective dose**, determined by applying the appropriate tissue weighting factors to the committed equivalent dose for each organ and then summing, so allowing for the different radio-sensitivities of the organs for stochastic effects

Dose Quantities

The calculation of doses from radionuclides within the body (internal emitters) is less straightforward than with external radiation. A number of factors that are specific to internal dosimetry influence the dose quantities that are used, as described below.

Internal doses are protracted over time

Individuals cease to receive doses from external radiation when they leave the radiation field. In contrast, radionuclides within the body continue to deliver dose until they are removed from the body by normal biological processes or until the amount remaining is reduced to a negligible level because of radioactive decay. Thus, ICRP recommends that the quantity to be controlled should be the *committed dose*, defined as the dose incurred in the fifty years following intake for an adult, and from intake to age 70 years for children (equation 1).

$$D(70 - t_0) = \int_{t0}^{70} \dot{D}(t)dt \qquad (1)$$

where "*D_dot*" is the absorbed dose rate at time t, and t_0 is the age at intake

Organs are the sources of irradiation

A particular organ may receive a dose either as a result of radionuclide decay within the same organ, or in other organs. For the purposes of a dose calculation, an organ in which radionuclide decay is taking place is called a *source organ* (denoted S), while an organ that receives a dose is called a *target organ* (denoted T). All organs in which radionuclides are present are in principle source organs, and all source organs are also target organs. Other organs may or may not be target organs, depending on how penetrating the radiation type (e.g. α, β, γ) is.

Radiosensitivity of organs is variable

Some radiation types are more effective than others in causing stochastic effects such as cancer. For radiation protection purposes, the absorbed dose associated with each radiation type is multiplied by a *radiation weighting factor*, w_R, which reflects these differences. In general, the decay of any radionuclide gives rise to the emission of more than one type of radiation, and so the absorbed dose from each radiation type must be weighted separately and then summed. The result is the *equivalent dose*, defined in equation (2) for a particular target organ T, and denoted H_T.

$$H_T = \sum_R w_R \overline{D}_{T,R} \qquad (2)$$

where D is the absorbed dose, which is averaged over the organ T, and the summation extends over all radiations R. The unit of equivalent dose is the sievert (Sv).

The radiation weighting factors, w_R, represent judgments regarding the potential relative biological damage of radiation R, taking into account all of the possible target organs or tissues (e.g. the lung) and all the possible health outcomes (e.g. lung cancer). In part, they reflect the density of ionization within the target organ, which is associated with the linear energy transfer (LET) of the radiation. Clearly, the use of radiation weighting factors represents a very simplified view of the health effects of different types of radiation. Their use is justified provided it is clear that the resulting assessed doses are only to be used for radiation protection purposes (for example, to confirm that doses are well below dose limits). The values of w_R on which current legislation is based are given in ICRP Publication 60 (ICRP, 1991), while ICRP's current recommended values are given in ICRP Publication 103 (ICRP, 2008).

Dose distributions in the body may be very inhomogeneous

It is not unusual for doses to target organs to differ by several orders of magnitude. Nevertheless, for radiation protection purposes, it is useful to derive a single dose quantity that broadly reflects the overall risk of stochastic effects on health. If all the organs and tissues of the body had the same sensitivity to ionising radiation, then a "mean dose" could be determined simply by taking a weighted average of the organ/tissue doses, with the weight factors equal to the ratio of the organ/tissue mass to the total body mass. However, there are significant differences between the radio-sensitivities of the organs/tissues, and these are reflected in the set of *tissue weighting factors* specified by ICRP. The sum of the weighted organ doses is the *effective dose*, E, which is defined as:

$$E = \sum_T w_T H_T \qquad (3)$$

where the summation extends over the organs/tissues assigned tissue weighting factors w_T. The values of w_T on which current legislation is based are given in ICRP Publication 60, while ICRP's current recommended values are given in ICRP Publication 103.

Calculation of internal dose

The committed equivalent dose rate at time t in target tissue T can be expressed as:

$$\dot{H}_T(t) = c \sum_S \sum_j q_{S,j}(t) \cdot SEE(T \leftarrow S;t)_j$$

where $q_{S,j}(t)$ is the activity of decay chain member j present in the source region S at age t; $SEE(T \leftarrow S;t)_j$ is the equivalent dose in the target tissue T per nuclear transformation in region S at age t for radionuclide j ; and c is any numerical constant required by the units of q and SEE (the Specific Effective Energy).

The committed equivalent dose in the target tissue T accumulated by age 70 y due to a single intake of a radionuclide j at age t_0, $H_T(70\text{-}t_0)$ is:

$$H_T(70 - t_0) = \int_{t_0}^{70} \dot{H}_T(t)dt = c \cdot \int_{t_o}^{70} \sum_S \sum_j q_{S,j}(t) \cdot SEE(T \leftarrow S;t)_j \, dt$$

For each radionuclide, the SEE at age t takes into account the contributions of each radiation emitted by the radionuclide weighted by the appropriate radiation weighting factor. The SEE is :

$$SEE(T \leftarrow S;\, t) = \frac{1}{M_T(t)} \left[\sum_i E_i \, Y_i \, w_{R,i} \, AF(T \leftarrow S;\, E_i, t) + w_{R,\beta} \int_0^\infty Y(E) \, E \, AF(T \leftarrow S;\, E, t) \, dE \right]$$

where E_i is the energy of the i^{th} discrete radiation emitted by the radionuclide with intensity Y_i per nuclear transformation, $M_T(t)$ is the mass of the target tissue T at age t, $w_{R,i}$ is the radiation weighting factor applicable to the i^{th} radiation, AF $(T \leftarrow S;\, E_{i,\,t})$ is the absorbed fraction, which is the fraction of the energy E_i emitted in S that is absorbed in T for an individual of age t, and $Y(E)$ dE denotes the number of electrons in the beta or positron spectrum, with energy between E and E+dE.

In most cases for non-penetrating radiations (alphas, betas, positrons, electrons), energy is assumed to be deposited only in the region where the radionuclide resides,

and so *AF* is equal to 1 when S and T are identical and zero otherwise. However, in some dosimetric models where the location of target cells is given special consideration (e.g. bone, the alimentary tract and the respiratory tract), the calculation of *AF* is more complicated, generally being undertaken using Monte Carlo methods in radiation transport models.

For penetrating radiations (gammas and X-rays), *AF* generally has to be calculated for all possible combinations of source and target organs. Calculations are performed using a computer model of the body often termed a "phantom" which describes the geometric relationship between the different tissues and organs of the body. Two types of phantom are currently in use: mathematical phantoms made up of arrangements of geometric shapes (e.g. the MIRD phantom); and voxel phantoms, derived from computed tomography (CT) or magnetic resonance images (MRI) obtained from high resolution continuous scans of a single individual. The MIRD phantom was used for dosimetry calculations for many years, but voxel phantoms are now coming into general use.

Biokinetic Models

The mathematical models describing the movement of radionuclides in the body are called biokinetic models. All of the models currently in use are compartment models which use first order kinetics to describe the transfer of material between compartments. Models contain a number of compartments, each representing a particular organ, part of an organ or tissue. A compartment is assigned a deposition or uptake fraction that determines its initial content (if any), and each path out of a compartment is assigned a rate constant which determines the removal rate *via* that pathway. The sum of the rate constants of the pathways leaving a compartment determines the retention half-time (also known as the biological half-time) for that compartment. A simple model may contain only two or three compartments linked by a single pathway in a linear chain. In more complex models, there may be networks of interconnected compartments in which recirculation between compartments may occur.

Biokinetic models describe the distribution of activity within the body as a function of time. Radioactive decay can be included, and the model can then be used to calculate the number of decays that take place in each compartment as a function of time. For single intakes of radionuclides with short physical half-lives or biological half-times, such as ^{131}I, almost all the dose is delivered over a period of days or weeks, whereas for radionuclides such as ^{239}Pu, with long radioactive decay half-lives and biological half-times, dose is delivered over the lifetime of the individual.

Ideally, model parameter values are determined from human studies, but for many elements, animal data must be used, and for some elements where no data is available, parameter values must be determined from data on chemically similar elements.

Biokinetic models can be used to describe the processed of intake, distribution, retention and excretion. Radionuclides entering the body by inhalation or ingestion cause the respiratory tract and gastro-intestinal tract to be irradiated. (Other organs will also be irradiated if the radiation is penetrating). Entry through contaminated wounds will also result in local irradiation of tissues. The subsequent behaviour of the radionuclide depends on the element concerned and its chemical form, which determine the solubility of the radionuclide and the extent to which it dissolves and is absorbed

into blood. The chemical nature of the element determines the distribution between the organs of the body, and the retention times in each organ. If a radionuclide is an isotope of an element that is normally present (e.g. Na, K, Cl) then it behaves like the stable element. If it has similar chemical properties to an element normally present, then it tends to follow the metabolic pathways of that element, although its rate of movement between the various compartments in the body may be different (e.g. ^{90}Sr and ^{226}Ra behave similarly to Ca, ^{137}Cs and ^{86}Rb similarly to K). The behaviour of other radionuclides depends upon their affinity for biological ligands and other transport systems in the body. Radionuclides entering the blood may distribute throughout the body (e.g. ^{3}H, ^{40}K, ^{137}Cs); they may selectively deposit in a particular tissue (e.g. ^{131}I in the thyroid; ^{90}Sr in bone); or they may deposit in significant quantities in a number of tissues (e.g. ^{144}Ce, ^{210}Po, ^{239}Pu, ^{241}Am).

In addition to the direct entry of radionuclides by absorption to blood, insoluble materials may be moved slowly through the lymphatic system in the form of small particles. Movement along lymphatic vessels can lead to accumulation of radionuclides in regional lymph nodes, e.g. the tracheo-bronchial lymph nodes following movement from the lungs or the axillary lymph nodes following movement from a wound in the hand. Particles may slowly reach the bloodstream and are then rapidly removed from the circulation by phagocytic cells of the reticulo-endothelial system in the liver, spleen and bone marrow.

This paper describes some of the biokinetic and dosimetric models developed by the International Commission on Radiological Protection (ICRP). Dose coefficients for intakes of radionuclides (that is, dose per unit intake values) derived using these models and published by ICRP (ICRP, 1989, 1993, 1994b, 1995a, 1995b), are incorporated into legislation in the European Union and others parts of the world.

The ICRP Human Respiratory Tract Model

The ICRP model of the Human Respiratory Tract, HRTM (ICRP, 1994a), applies explicitly to all members of the population, utilising reference parameter values for 3 month-old infants, 1, 5, 10 and 15 year old children and adults. The model can be used both for dosimetry and for the interpretation of individual monitoring data (such as measurements of activity in the lungs).

Physiology

The respiratory tract (Figure 1) is represented by five regions. The extrathoracic (ET) airways are divided into the ET_1 region, the anterior nasal passages, and the ET_2 region, which comprises the posterior nasal and oral passages, the pharynx and larynx. The other regions are within the lungs. They are the bronchial region (BB: trachea, generation 0 and bronchi, airway generations 1-8), the bronchiolar region (bb: airway generations 9-15), and the alveolar-interstitial region (AI: the gaseous exchange region). Lymphatic tissue is associated with both the extrathoracic and thoracic airways (LN_{ET} and LN_{TH} respectively).

Intakes and deposition

The HRTM can be used to calculate doses per unit exposure to radionuclides in air as well as doses per unit intake. Intake depends on ventilation, i.e. the product of the

breathing frequency and tidal volume of the lungs. This is the main aspect of the model that depends on age and level of exercise. The model can determine doses per unit exposure for four levels of exercise: sleep, sitting, light exercise, and heavy exercise.

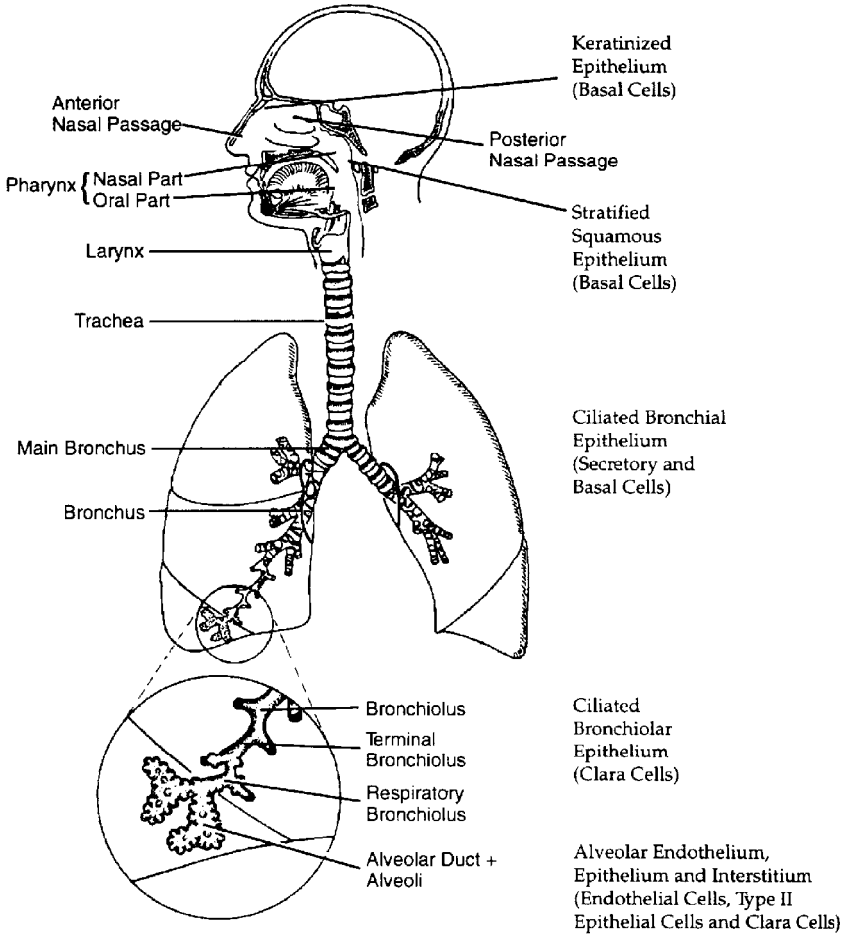

Fig. 1. Respiratory tract regions.

Deposition is calculated for each region of the respiratory tract taking account of both inhalation and exhalation. This is done as a function of particle size, breathing parameter values and/or work load, according to the age and sex of the subject. Deposition parameter values are specified for a particle size range of 0.006 µm AMTD[1] to 100 µm AMAD[2]. Default deposition parameter values for individuals are based on average daily patterns of activity. In general, deposition in the extrathoracic airways is greater for a 5 µm AMAD aerosol (the default value for occupational exposure), and conversely penetration to the alveoli is greater for a 1 µm AMAD aerosol (the default value for environmental exposure).

[1] Activity Median Thermodynamic Diameter (AMTD) is used for particle sizes < 0.5 µm, where thermodynamic processes dominate. AMAD (defined below) is used for larger particles sizes where aerodynamic processes (gravitational sedimentation, inertial impaction) dominate.
[2] Activity Median Aerodynamic Diameter (AMAD), defined such that 50% of the activity of an aerosol is associated with particles with aerodynamic diameters greater than the AMAD.

The model can be used for intakes of gases and vapours as well as particulate material. Three classes are defined: SR-0 for insoluble and non-reactive materials for which uptake/retention is negligible; SR-2 for very soluble and reactive materials for which absorption to blood is assumed to be complete; and SR-1, an intermediate category.

Clearance

The HRTM describes three clearance pathways. Material deposited in ET_1 (the front of the nose) is removed by extrinsic means such as nose-blowing. In other regions, clearance is competitive between (a) particle transport to the gastrointestinal (GI) tract and the lymphatic system, and (b) absorption into blood. Particle transport to the GI tract occurs by mucociliary clearance. It is assumed that particle transport rates are the same for all materials. Absorption into blood is material-specific and is assumed to occur at the same rate in all regions except ET_1, where none occurs.

Figure 2 shows the compartment model for particle transport. It is assumed that the AI deposit is divided between AI_1, AI_2 and AI_3 in the ratio 0.3:0.6:0.1 (half-times of about 30d, 700d and 7000d). The fraction of the deposit in bb and BB which moves slowly (bb_2 and BB_2) is particle size dependent (up to 50% for <2.5 μm) with 0.7% retained in the walls (bb_{seq} and BB_{seq}). A small fraction is retained in the wall in the ET_2 region (ET_{seq}) and the rest is cleared rapidly to the GI tract. Simultaneous absorption to blood occurs as a competing process in all compartments except ET_1.

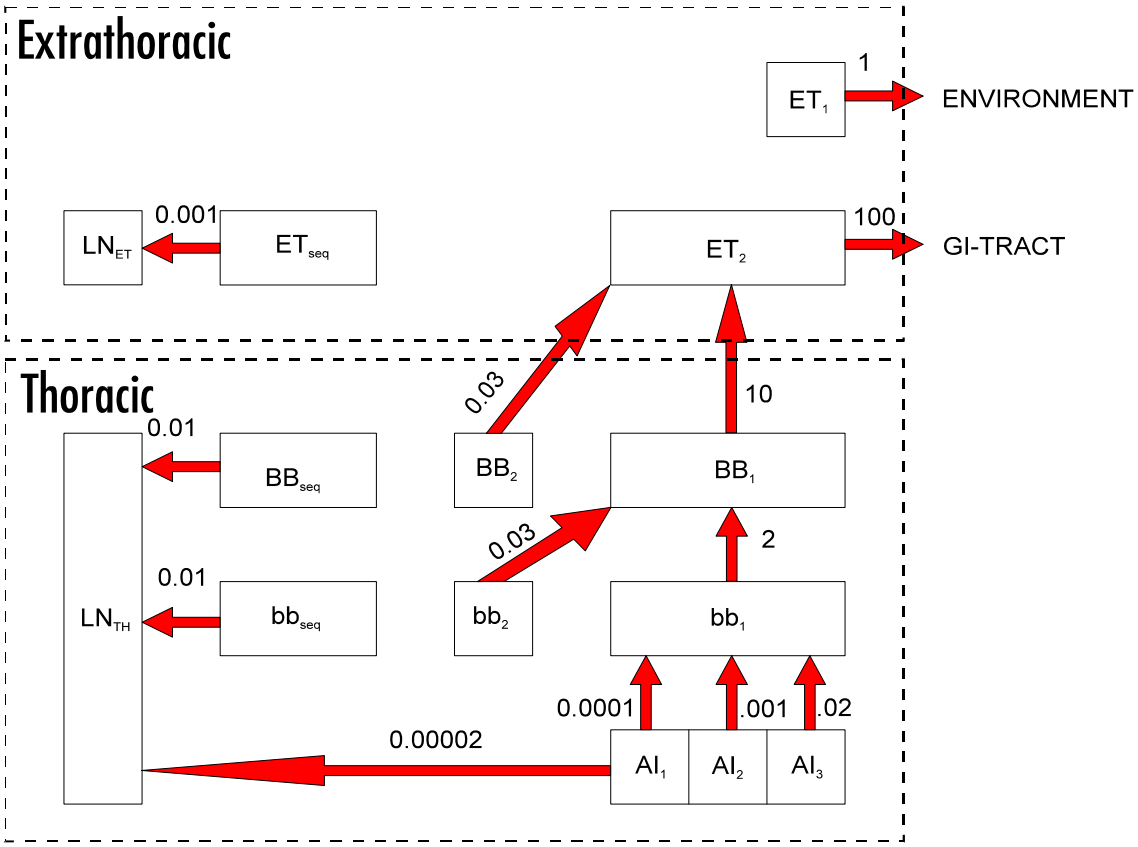

Fig. 2. Particle transport in the HRTM (ICRP Publication 66). Transfer rates are in units of d^{-1}.

Absorption

Absorption to blood depends on the chemical nature of the element and the chemical form deposited but is assumed to be independent of deposition site (with the exception of ET_1). The model allows for changes in dissolution and absorption to blood with time. The use of material-specific dissolution rates is encouraged but defaults are given; i.e. Types F (fast), M (medium) and S (slow). Type F has a single clearance component, whereas Types M and S have two components. Expressed as approximate half-times, these absorption rates are: Type F - 10 min (100%); Type M - 10 min (10%), 140d (90%); Type S - 10 min (0.1%), 7000d (99.9%).

For the example of a 1 μm AMAD aerosol, the total amounts reaching blood from the respiratory tract as predicted by the HRTM are 24% (F), 9% (M) and 1% (S) of the total inhaled. In each case, a further fraction of the inhaled material reaches blood by absorption from the gut after mechanical clearance and swallowing; this fraction will be greatest for Type F materials (most soluble) and least for Type S materials.

Dosimetry of the respiratory tract

In the respiratory tract model (as well as in models for the bone and gastro-intestinal tract), special attention is given to the locations of both source tissues (where radionuclides reside) and targets (tissues whose cells are thought to be at risk of cancer). Radionuclides are known to be deposited in the fluids lining the airways or alveoli, from where they can be absorbed to blood, perhaps residing in the airway wall for some time, or cleared to other regions of the lung by the action of mucus and cilia. There is good evidence that the target cells for cancer induction are the basal and secretory cells in the epithelia of the conducting airways, and certain cells of the alveolar region such as the Clara cells. To calculate doses to the sensitive tissues, cylindrical models of the airways are used, with the various cells modelled as concentric layers. Calculations are performed using Monte Carlo simulations of radiation transport. To calculate the dose to the lungs as a whole, a weighted average of the doses to the tissues is determined, independently of their relative masses. In the current model, the tissues are given equal weighting. For calculations of effective dose, the tissue weighting factor for the lungs (w_T = 0.12) is apportioned equally to the AI, BB and bb regions (i.e. 0.04 each).

Table 1. Committed doses from inhalation of radionuclides (1 μm AMAD aerosol) by adults (Sv Bq^{-1} intake).

Nuclide	Principal emissions	Half-life	Absorption Type	Equiv. Dose to lungs	Effective dose	% lung dose [1]
^{131}I	β,γ	8d	F	6.0×10^{-11}	7.4×10^{-9}	<1
^{137}Cs	β	30y	F	4.3×10^{-9}	4.6×10^{-9}	11
^{210}Po	α	138d	M	2.6×10^{-5}	3.3×10^{-6}	95
^{238}U	α	4.5×10^{9}y	M	2.2×10^{-5}	2.9×10^{-6}	91
^{239}Pu	α	2.4×10^{4}y	M	3.3×10^{-5}	5.0×10^{-5}	8
^{239}Pu	α	2.4×10^{4}y	S	8.7×10^{-5}	1.6×10^{-5}	65

1. % of effective dose contributed by equivalent dose to lungs

Table 1 gives examples of committed equivalent doses to the lungs and committed effective dose. The lung dose varies from 9 x 10^{-5} Sv Bq^{-1} for insoluble compounds of ^{239}Pu (i.e. Type S) to 6 x 10^{-11} Sv Bq^{-1} for ^{131}I, i.e. by over five orders of magnitude, reflecting the different retention times in the lung and the energy and type of radiation emitted.

Similarly, the contribution of lung dose to the effective dose varies from high values of about 90% (^{210}Po, ^{238}U) to a value of <1% for ^{131}I. The rapid absorption of Type F materials from the lungs to blood makes lung dose less important compared to doses to other tissues - the thyroid in the case of ^{131}I. For insoluble ^{239}Pu the lung dose dominates the effective dose (65%) due to its long retention in lung, whereas this is not true for more soluble Type M forms (8%) since in this case doses to systemic tissues are higher.

ICRP Human Alimentary Tract Model

Until 2007, the model of the gastrointestinal tract used by ICRP was that described in Publication 30 (ICRP, 1979). This is a simple model, containing four compartments in a linear chain representing the stomach, small intestine and upper & lower large intestines. In 2007, the Publication 30 model was superseded by ICRP's Human Alimentary Tract Model, HATM (ICRP, 2007), and future ICRP publications will use this model. However, current EU legislation contains dose coefficients generated using the Publication 30 model. This section describes the main features of the HATM, shown in Figure 3.

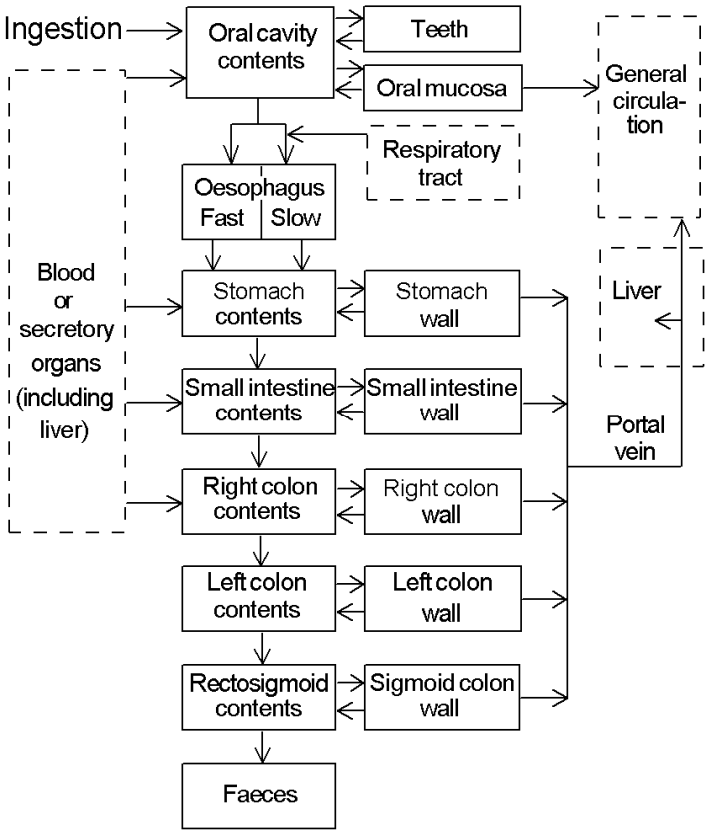

Fig. 3. Structure of the HATM. The dashed boxes are included to show connections with the respiratory tract model and systemic models.

The increased complexity reflects the wider scope and greater degree of realism of the HATM as compared to the Publication 30 model. The main features of the HATM are:

- Age-dependent parameter values are specified for the dimensions of alimentary tract regions and associated transit times of contents through the regions, as are gender-dependent parameter values of dimensions and transit times for adults.
- Retention of radionuclides in the mucosal tissues of the walls of alimentary tract regions is included. For some radionuclides, such as isotopes of iron and related elements, there may be significant retention in the intestinal wall.
- Explicit calculations are made of dose to target regions for cancer induction within each alimentary tract region. The sensitive cells for cancer induction are thought to be the stem cells in the base of the crypts which are likely to receive negligible doses from α-emitting nuclides in the gut lumen and lower doses for most beta emitters.
- All alimentary tract regions are included. Doses are calculated for the oral cavity, oesophagus, stomach, small intestine, right colon, left colon and recto-sigmoid (including the rectum). Colon doses are combined as a mass-weighted mean.
- Transit times are specified for food and liquids, as well as for total diet, for the mouth, oesophagus and stomach.

Table 2. Comparison [1] of ingestion dose coefficients derived using the HATM and ICRP Publication 30 models for adult males.

Nuclide	Equivalent doses [1]			Effective dose [1]
	Stomach	Right colon [2]	Left colon [3]	
^{90}Sr	1.0	0.2	0.1	-6%
^{106}Ru	1.2	0.2	0.1	-60%
^{239}Pu	0.8	0.4	0.2	0%

1. Equivalent doses compared as a ratio (HATM : Pub 30), effective doses as a fractional change (%).
2. Compared with the upper large intestine of the Publication 30 model.
3. Compared with the lower large intestine of the Publication 30 model.

Table 2 compares doses calculated using the HATM and Publication 30 models for three examples. For the beta-emitting nuclides ^{90}Sr and ^{106}Ru, the substantial reduction in colon doses in the HATM reflects reduced Specific Absorbed Fractions (SAF). For ^{239}Pu, there is no direct dose to alimentary tract regions using the HATM; all of the dose is due to absorbed ^{239}Pu deposited in soft tissues. The extent to which differences in equivalent doses to the tissues of the alimentary tract affect the effective dose depends on the relative contribution of these tissues to the effective dose. For ^{90}Sr, effective dose is dominated by skeletal doses (bone surfaces and red bone marrow) and there is little difference in the effective dose calculated using the HATM and Publication 30 models. For ^{106}Ru, doses to alimentary tract regions contribute significantly to effective dose with the result that the value obtained using the HATM is about half of that using the Publication 30 model. For ^{239}Pu, as for ^{90}Sr, doses to

alimentary tract regions make only a small contribution to effective dose, the dominant contributors being liver and skeletal tissues.

Biokinetics and dosimetry for plutonium

The absorption of Pu and other actinide elements from the GI tract is very low; default fractional absorption (f_1 values) recommended by ICRP for adults are 10^{-5} for Pu oxide, 10^{-4} for Pu nitrate and similar compounds and 5 x 10^{-4} for other chemical forms and environmental forms of Pu. For inhalation, the default absorption assumption is Type M for Pu (Type S for Pu oxide).

After absorption to blood, the principal sites of deposition of Pu and related elements are the skeleton and liver. In Publications 56 and 67 (ICRP 1989, 1993) a physiologically-based age-dependent model was developed and applied to Pu, Am and Np and subsequently to Th and Cm (ICRP 1995a,b). The model is shown in Figure 4.

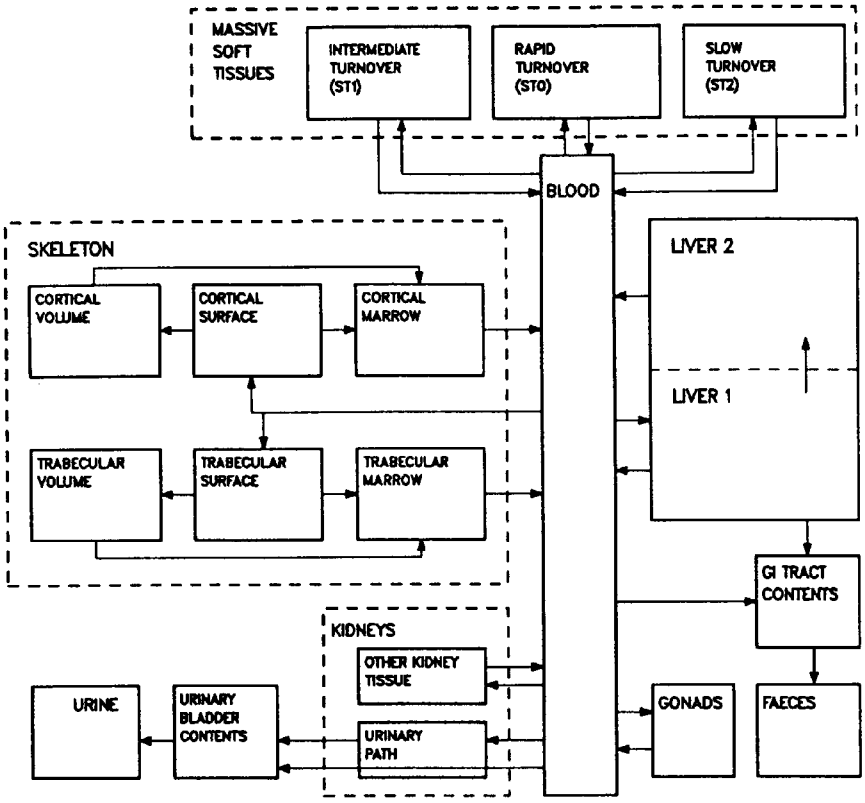

Fig. 4. Biokinetic model for the actinide elements: Pu, Am, Cm, Th, Np.

The radiation-sensitive parts of the skeleton are the red bone marrow, for leukaemia induction, and a layer of cells (the endosteal cells) on the inner bone surface, for induction of bone cancer. In adults, red bone marrow is contained in the trabecular or "spongy" bone which constitutes, for example, the vertebrae and the ends of the long bones. Cortical or compact bone, which constitutes the shafts of the long bones, contains only inactive fatty marrow in the adult. The model treats cortical and trabecular regions of the skeleton separately, with initial deposition on bone surfaces. A

specific aspect of the biokinetics of Pu and related elements is that incorporation into bone volume results from burial surface deposits of new bone.

Pu removed from the bone surfaces may be retained in bone marrow. Accordingly, the model includes transfer to bone volume and to bone marrow. It also includes transfer from bone volume to bone marrow due to resorption. Activity is eventually released from bone marrow to blood and becomes available for recycling to all tissues including the skeleton. For simplicity, rates of bone growth and resorption are all taken to be the same and show the same age-dependence.

The liver has two compartments for all elements except Am and Cm, for which one compartment is used. Activity taken up by Liver 1 is transferred to Liver 2 except for a small proportion excreted in bile to the GI tract. Activity is lost to blood from Liver 2. Figures 5 and 6 illustrate the retention of Pu in the skeleton and liver predicted by the model for different ages at intake.

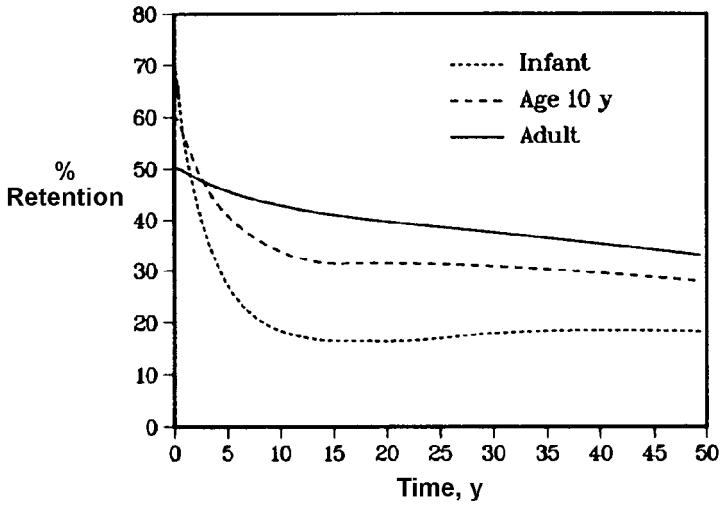

Fig. 5. Model predictions for the retention of Pu in the skeleton as a function of time after entry to blood (% total entering blood).

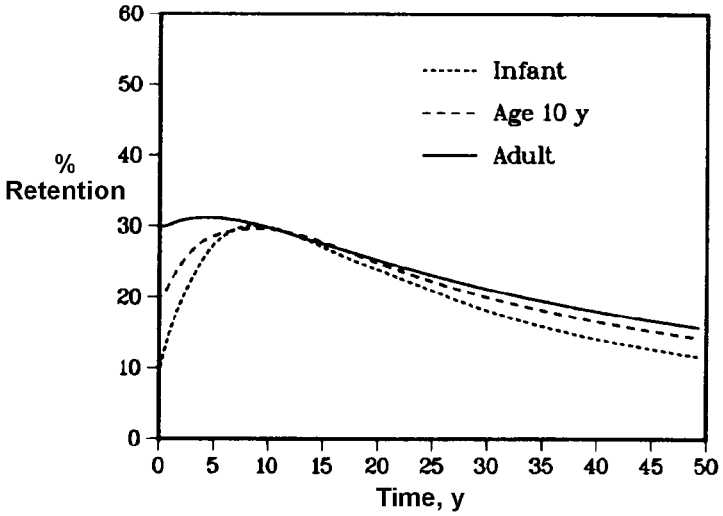

Fig. 6. Model predictions for the retention of Pu in the liver as a function of time after entry to blood (% total entering blood).

Table 3 shows dose coefficients for the ingestion or inhalation of ^{239}Pu. Committed effective doses after ingestion are due very largely to contributions from doses to skeletal tissues and liver with smaller contributions from doses to the gonads (testes or ovaries).

Table 3. Committed effective doses from the ingestion or inhalation of ^{239}Pu, 10^{-6} Sv Bq^{-1}.

Age	Ingestion *	Inhalation (1 μm AMAD)†	
		Type M	Type S
3 months	4	80	43
1 year	0.4	77	39
5 years	0.3	60	27
10 years	0.3	48	19
15 years	0.3	47	17
Adult	0.3	50	16

*Environmental forms; f_1 = 5 x 10^{-4}, 5 x 10^{-3} for infants

‡Type M is default; Type S = e.g. Pu oxide

After inhalation, doses to the lungs are also important, particularly for Type S materials. Doses are lower after ingestion than inhalation because of the small amount absorbed to blood (most activity passes unabsorbed through the GI tract; >99%). Doses per unit intake are greatest at younger ages because of the smaller tissue masses, greater relative uptake in the skeleton (and greater f_1 value in infants).

Monitoring for ^{239}Pu is usually based on urine monitoring followed by mass- or alpha-spectrometry. Modern laboratories can detect levels less than a few mBq per litre. Faecal monitoring can also be used with some advantage, but there are often acceptability problems with workforces. Plutonium-239 emits 17 keV X-rays during alpha decay and these emissions can be detected by whole-body counting.

In general ^{239}Pu, like other alpha-emitters, presents problems for monitoring since intakes of a relatively small mass can lead to dose limits being exceeded. Excretion levels resulting from a small mass are low, and the non-penetrating nature of alpha emissions make detection difficult. In some cases the presence of other Pu isotopes, ^{241}Am, and fission products can make detection easier because of higher energy emissions but, of course, the composition of the mixture needs to be established.

Individual monitoring for internal contamination

The previous sections have described methods for calculating doses from a known intake received by a reference person using standard phantoms and biokinetic models. This is a prospective assessment of dose, because it answers the question "What would the dose be if a person receives an intake of x Bq?". Usually a unit intake (1 Bq) is assumed and the results are known as dose coefficients (Sv Bq^{-1}) and are published in standard compendia (e.g. ICRP, 1994b). To obtain estimates of doses to individuals the dose coefficients must be multiplied by the known intake (Bq). For environmental intakes this can be derived from measurements of concentrations in foodstuffs (e.g. Bq kg^{-1}) and habit data giving the typical consumption of the foodstuff in a set time period.

In occupational settings however, workers are monitored on a regular basis and the results of measurements on their bodies or excreta can be used to estimate intake. This is a retrospective assessment of dose, because it answers the question "What is our best estimate of the dose, give that x Bq (or y Bd d^{-1}) has been measured in the lungs (or in urine, etc)?". This section gives general information on the design and use of these methods; it is taken largely from ICRP Publication 78 (ICRP, 1997).

Methods of Individual Monitoring

The purpose of this section is to describe briefly the main measurement techniques, their advantages and their limitations. In most cases, individual monitoring for intakes of radionuclides may be achieved by body activity measurements, excreta monitoring, air sampling with personal air samplers, or a combination of these techniques. The choice of measurement technique will be determined by several factors: the radiation emitted by the radionuclide; the biokinetic behaviour of the contaminant; its retention in the body taking account of both biological clearance and radioactive decay; the required frequency of measurements; and the sensitivity, availability, and convenience of the appropriate measurement facilities.

Routine monitoring programmes usually involve only one type of measurement if adequate sensitivity can be achieved. For some radionuclides, only one measurement technique is feasible, e.g. urine monitoring for intakes of tritium. For radionuclides such as plutonium isotopes that present difficulties for both measurement and interpretation, a combination of techniques should be employed. If different methods of adequate sensitivity are available, the general order of preference in terms of accuracy of interpretation is: body activity measurements; excreta analysis; personal air sampling. Results of monitoring of the working environment (area monitoring) may provide information that assists in interpreting the results of individual monitoring, e.g. information on particle size, chemical form and solubility, time of intake.

The results of workplace monitoring for air contamination may sometimes be used to estimate individual intakes. However the interpretation of the results of measurements from air sampling in terms of intake is not simple and may be misleading. The most common form of representative sampling is by using fixed samplers at a number of selected locations intended to be reasonably representative of the breathing zone of the worker.

Monitoring in relation to a particular task or event may often involve a combination of techniques so as to make the best possible evaluation of an unusual situation, for example, a programme of both body activity and excreta measurements.

In some cases of suspected incidents, screening techniques (such as measuring nose blow samples or nasal smears) may be employed to give a preliminary estimate of the seriousness of the incident.

In vivo measurements

The IAEA (IAEA, 1996) has given guidance on the direct measurement of body content of radionuclides. Direct measurement of body or organ content provides a quick and convenient estimate of activity in the body. It is feasible only for those radionuclides emitting radiation that can escape from the body. In principle, the technique can be used for radionuclides that emit: X or γ radiation; positrons, since they can be detected by

measurement of annihilation radiation; energetic β particles that can be detected by measurement of bremsstrahlung; and some α-emitters that can be detected by measurement of the characteristic X rays.

Many facilities for the measurement of radionuclides in the whole body or in regions of the body consist of one or a number of high efficiency detectors housed in well-shielded, low-background environments (IAEA, 1996). The geometrical configuration of the detectors is arranged to suit the purpose of the measurement, e.g. the determination of whole-body activity or of activity in a region of the body such as the thorax or the thyroid. The skull or knees may be used as a suitable site for measurement of radionuclides in skeleton.

Care must be taken to remove surface contamination before body activity is measured. For routine measurements, determination of whole-body content is often adequate for radiological protection purposes.

Commonly encountered fission and activation products, such as ^{131}I, ^{137}Cs and ^{60}Co, can be detected with comparatively simple equipment at levels that are adequate for radiation protection purposes. Such simple equipment may consist of a single detector, viewing the whole body or a portion of the body, or, for iodine isotopes, a small detector placed close to the thyroid. The advantage of simple equipment is that it may be operated at the place of work, thereby avoiding the time required to visit a remote whole-body monitoring facility. Measurements may then be made more frequently so that any unusually large intake would be recognised soon after it had occurred.

In contrast, high sensitivity techniques are needed for monitoring a few radionuclides at the levels that are required for protection purposes. Examples are the α-emitting radionuclides such as plutonium isotopes.

The activity present in a wound can be easily detected with conventional β-γ detectors if the contaminant emits energetic γ-rays. In the case of contamination with α-emitting radionuclides, detection is more difficult since the low energy X rays that follow the α-decay will be severely attenuated in tissue; this effect is more important the deeper the wound.

Analysis of Excreta and other Biological Materials

In some cases, excreta monitoring may be the only measurement technique for those radionuclides which have no γ-ray emission or which have only low energy photon emissions. Excreta monitoring programmes usually involve analysis of urine, although faecal analysis may be required in some circumstances, for example where an element is preferentially excreted via faeces or to assess clearance of insoluble material from the respiratory tract. Other samples may be analysed for specific investigations. Examples are nose blow or nasal smears as routine screening techniques or blood, in the case of suspected high level contamination.

The collection of urine samples involves three considerations:

- Firstly, care must be taken to avoid adventitious contamination of the sample.
- Secondly, it is usually necessary to assess the total activity excreted in urine per unit time from the sample provided. For most routine analyses, a 24 hour collection is preferred but, if this is not feasible, it must be recognised that smaller samples may not be representative. (Tritium is a particular case for which it is

usual to take only a small sample and to relate the measured activity concentration to the concentration in body water.)

- Thirdly, the volume required for analysis depends upon the sensitivity of the analytical technique. For some radionuclides, adequate sensitivity can be achieved only by analysis of several days' excreta.

Radionuclides that emit γ-rays may be determined in biological samples by direct measurement with scintillation or semiconductor detectors. Analysis of α- and β-emitting radionuclides requires chemical separation followed by appropriate measurement techniques. Measurement of so-called total α or β activity may occasionally be useful as a simple screening technique, but there is no method that will determine accurately all the α and β activity in the sample. The technique may be used in routine monitoring situations where intakes are expected to be very low compared with annual limits. The results would not be interpreted quantitatively, but would be used to provide confirmation of satisfactory conditions, an unusual result indicating the need for further investigation which would include radiochemical analysis. Total activity measurements may also be useful following a known contamination event or to identify those samples that merit early attention. Measurements of total α or β activity cannot be used in quantitative evaluations of intake or committed effective dose, unless the radionuclide composition is known.

Measurement of activity in exhaled breath is a useful monitoring technique for some radionuclides. In practice this method may be useful for ^{226}Ra and ^{228}Th since the decay chains of both these radionuclides include gases which may be exhaled. It can also be used to monitor $^{14}CO_2$ formed *in vivo* from the metabolism of ^{14}C-labelled compounds.

Air Sampling

A Personal Air Sampler (PAS) is a portable device specifically designed for the estimation of intake by an individual worker from a measurement of time-integrated concentration of activity in air in the breathing zone of the worker. A sampling head containing a filter is worn on the upper torso close to the breathing zone. Air is drawn through the filter by a calibrated air pump carried by the worker. Ideally, sampling rates would be representative of typical breathing rates for a worker (~1.2 m3 h^{-1}). However, sampling rates of current devices are only about 1/10 of this value. The activity on the filter may be measured at the end of the sampling period to give an indication of any abnormally high exposures. The filters can then be retained, bulked over a longer period, and the activity determined by radiochemical separation and high sensitivity measurement techniques. An estimate of intake during the sampling period can be made by multiplying the measured integrated air concentration by the volume breathed by the worker during the period of intake.

A PAS does not provide information on particle size. Nevertheless, it is important either to determine the particle size distribution of the inspirable material or to make realistic assumptions about it, since it can have a marked effect on deposition fractions in the respiratory tract, and hence on dose estimates. All samplers are size selective to a greater or lesser extent, under- or over-sampling at particular particle sizes, and this can result in errors in intake estimation. The aspiration efficiency of a PAS should therefore be determined to indicate whether corrections are necessary.

Static air samplers (SAS) are commonly used to monitor workplace conditions, but can underestimate concentrations in air in the breathing zone of a worker, typically by a factor of up to about 10. Nevertheless, if SAS devices are sited appropriately, a comparison of PAS and SAS measurements can be used to define a PAS:SAS air concentration ratio which can be used in the interpretation of SAS measurement for dose assessment purposes. It should however be recognised that the use of SAS is a relatively indirect method for assessing doses, and use of the results to estimate individual dose requires a careful assessment of exposure conditions and working practices. Apart from their potential use for dose estimation, SAS devices can also provide useful information on radionuclide composition, and on particle size if used with a size analyser such as a cascade impactor.

Estimation of Intake and Dose

In an investigation of an incident for which the time of intake is known, the intake can be estimated from the measured results using the values of measured quantities predicted by the biokinetic models for a unit intake. If only a single measurement is made, the intake can be determined from the measured quantity, M, by:

$$Intake = \frac{M}{m(t)}$$

where m(t) is the predicted value at the time of intake t. The intake can be multiplied by the dose coefficient to give the committed effective dose; this can then be compared with the dose limit or some investigation level based on dose.

If the measurement indicates that an investigation level has been exceeded, further investigation is required. The nature of the investigation will depend upon the circumstances and the extent to which the investigation level is exceeded. The following should be considered:

- repeated measurements to confirm or refine the initial evaluation, and
- the use of additional monitoring techniques.

The predicted values can then be scaled to obtain the best fit to the measured data points. The best fit is usually taken to be that fit which minimizes the sum of the squares of the residuals, a residual being defined as the number of standard deviations separating a measurement from the fitted curve. The intake is then equal to the value by which the predicted values are scaled. Many laboratories have developed software for fitting the intake in this, and other more mathematically advanced, ways, for example HPA's IMBA program.

When a positive measurement is detected by routine monitoring, say at monthly intervals, certain assumptions have to be made about the time of intake when calculating doses. It is often assumed that intake took place in the middle of the monitoring interval of T days. If the dose is large further measurements would be necessary. It should be noted that an intake in a preceding monitoring interval may influence the actual measurement result obtained.

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Optimisation of radiation protection for pediatric and adult patients in radiography and computed tomography

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Introduction

The need for optimization of radiation protection of patients in diagnostic medical imaging became clear already some 30 years ago. Scientists that pioneered in radiation protection observed that large variations in radiation exposure of patients existed for one and the same diagnostic examination. This triggered many national and international field studies. The field studies that have been carried out in the United Kingdom are particularly well known, and the observed steady reduction of third quartile adult patient dose values from three reviews of UK national patient dose data for the mid-1980s; the year 1995; and the year 2000 illustrates that improvements in acquisition protocols and imaging equipment resulted in substantial optimization of radiation protection [Wall]. The need for optimization of radiation protection of young patients became evident when the number of clinical indications for CT examination of children grew rapidly mainly due to the enhanced speed of the CT scanners around the year 2000. It was observed in the United States that many CT scans of children were performed with acquisition protocols that were developed for adults, and that used too high exposures for small children [Brenner et al.]. This led to special efforts worldwide in optimization of pediatric CT examinations; the Alliance for Radiation Safety in Pediatric Imaging of the United States of America (USA) encourages on its website [www.pedrad.org] increased awareness of opportunities to lower radiation dose in pediatric CT procedures. A recently published field study in the USA revealed CT that as the major contributor to the population dose [Mettler et al.]. In 2006, Americans were exposed to more than seven times as much ionizing radiation from medical procedures as was the case in the early 1980s. Radiation exposure due to medical examinations in the United States is now assessed at about 3 mSv per year, the USA is the first country were the population exposure is dominated by exposures from diagnostic imaging, whereas in other countries the population exposure is dominated by exposure to natural sources of ionizing radiation (Figure 1). The high levels of radiation exposures from CT as were observed in the USA are not (yet) observed in Europe. Nowadays the manufacturers of CT scanners offer improved opportunities for optimization of radiation protection in CT examinations; the biggest challenge for the future in optimization of CT seems now to be achieving proper (evidence based)

justification of the referral for CT examinations. It has to be established which CT examination really contribute to the proper clinical care of patients, and which referrals are mainly based on too defensive diagnostic strategies.

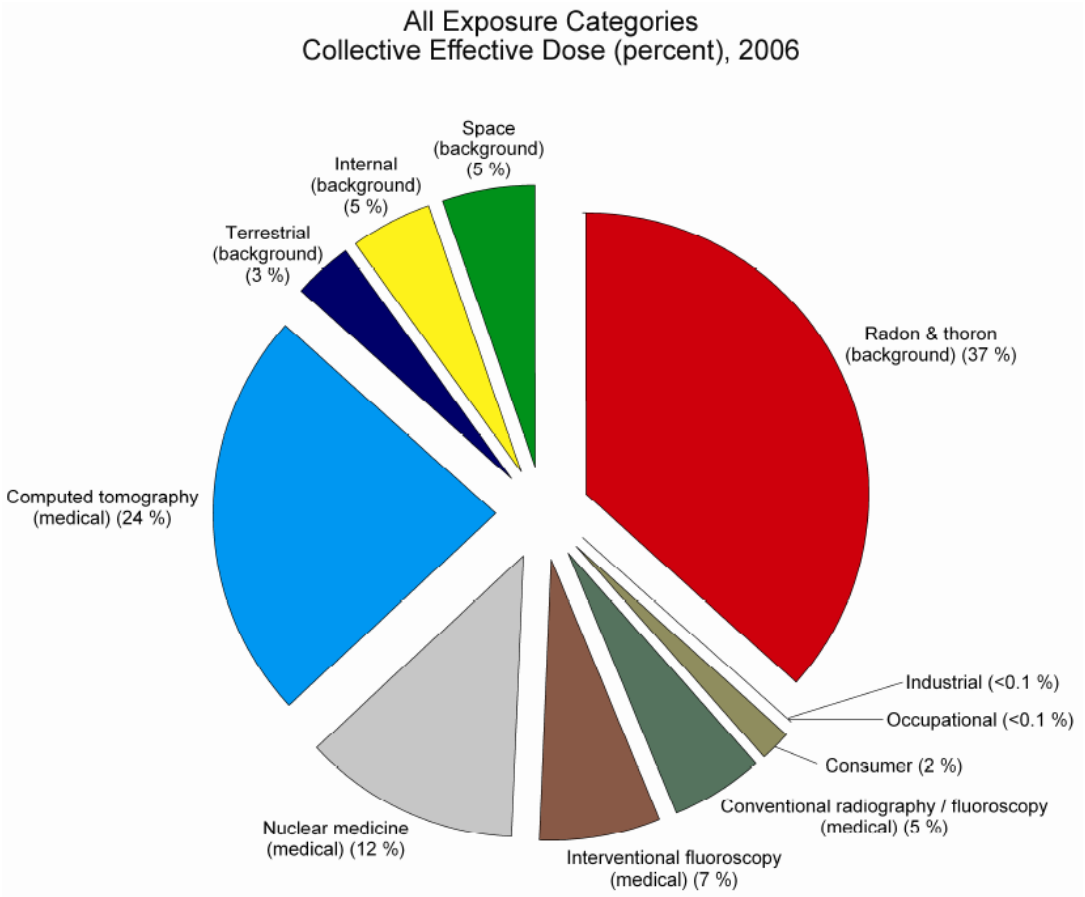

Fig. 1. The distribution of the population dose in the United States. Note that CT is a major contributor to the population dose. In 2006, Americans were exposed to more than seven times as much ionizing radiation from medical procedures as was the case in the early 1980s. Radiation exposure due to medical examinations in the United States is now about 3 mSv per year [Mettler et al.].

Optimization in radiography

X-ray projection radiography is a well known, widely available, and relatively cheap imaging procedure for medical diagnosis; dental X-ray radiography is the most frequently performed examination, next comes chest x-ray radiography. Radiographs of the extremities (legs, arms, hand, feet), of the abdomen, and of the spine are also often made (Figure 2). Radiography of the skull is nowadays rarely performed, the skull is preferably examined with 3D imaging techniques like computed tomography of magnetic resonance imaging.

Radiographs were traditionally recorded on x-ray film. The latent image that is recorded by x-ray film became visible after developing the film. Digital detectors became available about 30 years ago. There were initially high expectations with regard to the improvement of image quality and the reduction of patient dose when the first systems for computed radiography (CR) were introduced. The latent image that is

recorded on CR plates is translated into light in readout units by photo-stimulation with a laser. The generated light is then translated into an electrical signal, and the electrical signal into a digital image. Unfortunately in the early years of CR, the high expectations were not fulfilled. Users of CR systems often had to increase dose in order to achieve images with sufficiently good image quality. Advances in detector design made that CR finally evolved into a dose efficient alternative for x-ray radiography on film. Systems for digital radiography (DR) became available somewhat later, these systems use a detector that both records the image, and that translates the image into an electrical signal. DR detectors generally provide the best dose efficiency in radiography, they can be operated at dose levels well below the doses that are used in radiography with film or CR. The opportunity for dose reduction that detectors like CR and DR offer is not the only advantage. It is nowadays generally recognized that, compared to film, CR and DR systems are capable to provide better image quality, and that with these systems less retakes are required. Digital archives and fast networks make it possible to store and distribute digital images much more efficiently compared to film.

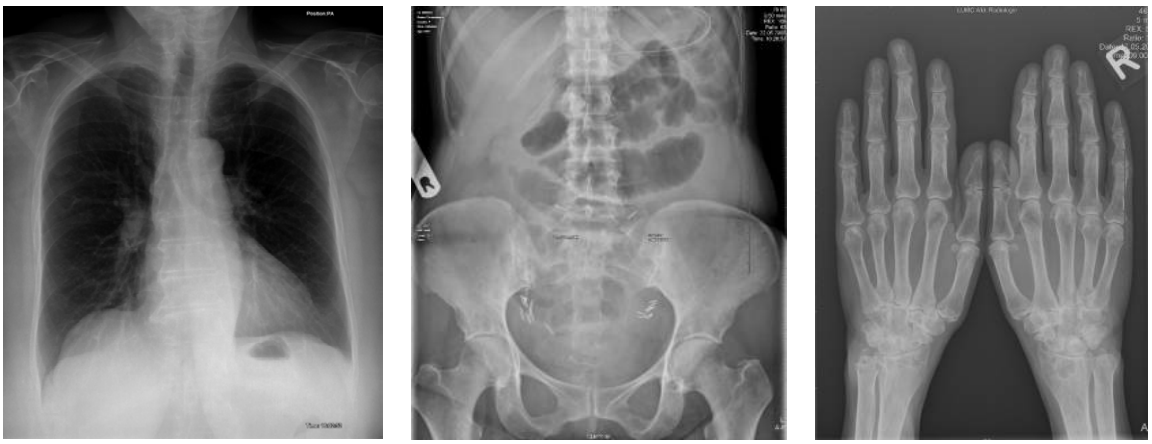

Fig. 2. Three examples of radiography, from the left to the right radiography of the chest (frontal view), radiography of the abdomen (frontal view), and radiography of the hands. Note that radiography yields an superposition in a 2D plane of all tissues within the examined area.

Acquisitions of radiographs have to be optimized; this can be achieved by optimizing the x-ray spectrum. The 'color' of the spectrum (or the distribution of photons of different energies in the spectrum) can be optimized by choosing the appropriate tube voltage (kV) and tube filtration (in addition to the inherent filtration by the window of the x-ray tube, additional aluminium and copper filters are being used). Spectra with photons with relative low energies can be used for imaging relatively thin objects (extremities, children), spectra with higher photon energies are used for imaging thicker body parts (abdomen, spine). Very high photon energies are used in chest radiography to reduce the contrast between bone (ribs) and soft tissue (lungs), this is required since the ribs may, at lower photon energies, obscure the lesions in the lung soft tissue. The 'brightness' of the spectrum (or the intensity of the spectrum) can be optimized by selecting the most appropriate tube charge (mAs), which is the product of tube current (mA) and exposure time (s). Most x-ray units make use of automatic exposure control, in this case a radiation detector measures the dose that the detector

receives, and when the required preset dose level is reached, the system terminates the exposure. It is necessary that the preset dose level is optimized to provide the best balance between the required image quality and at the same time the lowest possible dos for the patient. It is essential that the automatic exposure system of modern CR and DR systems is configured to operate well below the preset dose that would be used for recording images on film. Sometimes it is not possible to use automatic exposure control, for example when radiographs are being made with a mobile x-ray unit at the department of intensive care, in this case a table with appropriate and optimized acquisition techniques that cover all relevant examination should be available for the radiographer. In addition the field size should be optimized, by collimation of the x-ray beam the field size should be just sufficiently large to visualize the body part of interest.

It is also important to realize that for one clinical question; often more that one radiograph is being made. For chest radiography, generally two radiographs are being made; one provides a frontal view, the other a lateral view of the patient. For some clinical referrals, it has been observed that great variety exists in the number of the radiographs that are being taken, and in the views of these radiographs, for example when imaging the cervical spine, the abdomen, or the pelvis. These large variations in clinical practices and acquisition protocols provide a substantial, and until now hardly exploited, opportunity for optimization in x-ray radiography, Figure 3 illustrates that for one and the same clinical question (cervical spine, neck pain), and in only 11 hospitals, eight different views were being used in routine practices, and the numbers of views varied between two and five [Teeuwisse et al.].

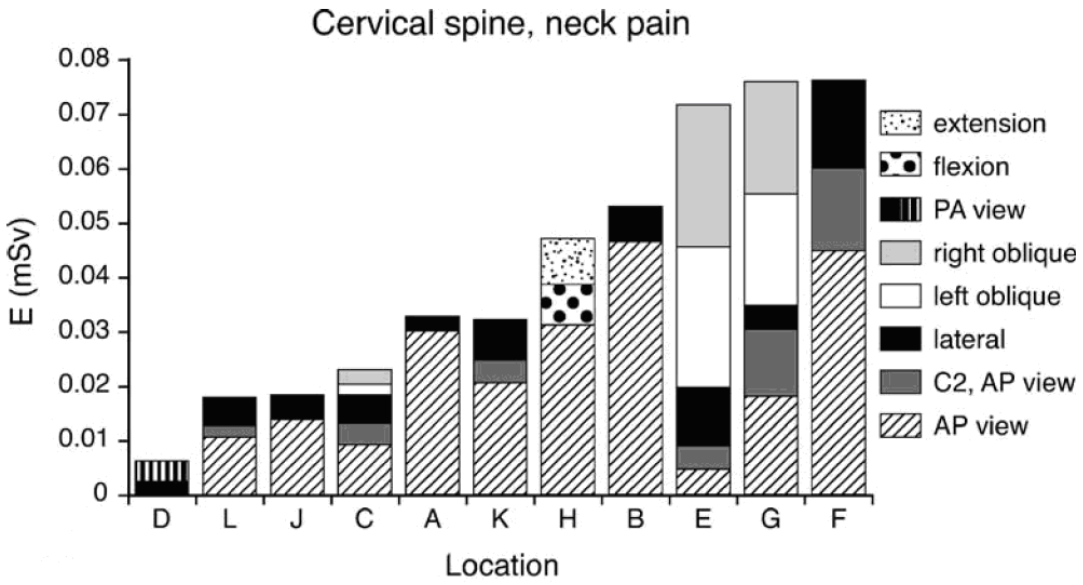

Fig. 3. Cumulative effective dose (mSv) for the clinical radiography indication cervical spine, neck pain illustrates the impact of the clinical protocol (the number and type of views) and the dose for the different views on the spread in integral patient dose [Teeuwisse et al.].

Dosimetry in radiography is usually based on measurement of the dose-area product (DAP, Gy.cm^2), this is a useful quantity, since the DAP incorporates the effect on patient dose of all relevant acquisition parameters like tube voltage, tube charge, and field size. DAP can be converted into effective dose by using appropriate effective dose

conversion factors or preferably by dedicated software (PCXMC, Radiation and Nuclear Safety Authority in Finland). The so-called DICOM header that is added as supplementary information to the digital images, generally also stores the dose-area product. Ideally, the dose-area product could be retrieved with a simple query from the digital archive in which the radiographs are being stored; however, unfortunately such a functionality is not always available. European countries defined diagnostic reference levels for selected projections in x-ray radiography of adults; some countries also established diagnostic reference levels for pediatric radiography.

Optimization in computed tomography

Computed tomography (CT) was introduced in 1974. The substantial advances in CT technology for 3D imaging with excellent image quality led to a fast increase of its clinical use. CT is often applied for the diagnosis of tumours, vascular disease, and trauma. CT shows anatomy and pathology in arbitrary planes, or in 3D reconstructions (Figure 4). The excellent image quality of CT comes, compared to radiography, at the price of a substantial higher radiation exposure. CT is nowadays the major source of radiation exposure of patients which makes optimization (and justification) of applications of CT of particular relevance.

CT scanners are equipped with a rotating x-ray tube and detector. During one rotation, of the x-ray tube and detector, the early CT scanners were only capable of scanning one thick axial slice, these scanners were relatively slow, they provided good image quality in the axial plane, but the 3D image quality was poor. Advances in CT technology like multislice CT (CT scanners that were able to scan more than one image in one rotation) and helical CT (CT scanners that combined the rotation of the x-ray tube and the detector with translation of the patient) made CT an excellent imaging modality for fast 3D imaging. With the introduction of volumetric CT (CT scanners that allow for imaging 320 slices during one rotation), entire organs like the brain or the heart can be imaged within a fraction of a second.

Optimisation of CT was mainly enhanced by technical improvements, but can also be achieved by adapting optimal imaging protocols. An important step in the improvement of the dose efficiency of CT scanners was achieved by the transition from air filled CT detectors to solid state CT detectors. Another important step was the introduction of automatic exposure control in CT, this allowed for adapting the tube current for the size of the patient but also for the local attenuation of the patient. To achieve consistent image quality within the scanned range, CT scanners nowadays automatically use higher exposure in thick patients and lower exposure in thin patient. During the CT scan the exposure can also be adapted to the local attenuation, during a CT of the chest for example the tube current is higher when the CT scan passes the shoulder (with its highly attenuating bones) and lower when it scans the lungs. Sophisticated noise reduction algorithms contribute further to the dose efficiency of CT scanners. CT scans can be optimized by carefully selecting the most appropriate preset for the automatic exposure control, which determines the tube charge (mAs) during the scan. Figure 5 shows the differences in image quality within the brain when different presets of the tube current are being used. Experiments to establish the optimal tube current cannot be performed by scanning the patient several times but such experiments have to be performed using low dose simulation algorithms and observer studies

[Joemai et al.]. CT scans can be further optimized by carefully selecting the most appropriate tube voltage, generally a tube voltage of 120 kV is applied in CT, for obese patients a higher tube voltage of for example 140 kV might be more appropriate, but for iodine contrast enhanced vascular studies a lower tube voltage of 100 kV or even 80 kV might be preferred.

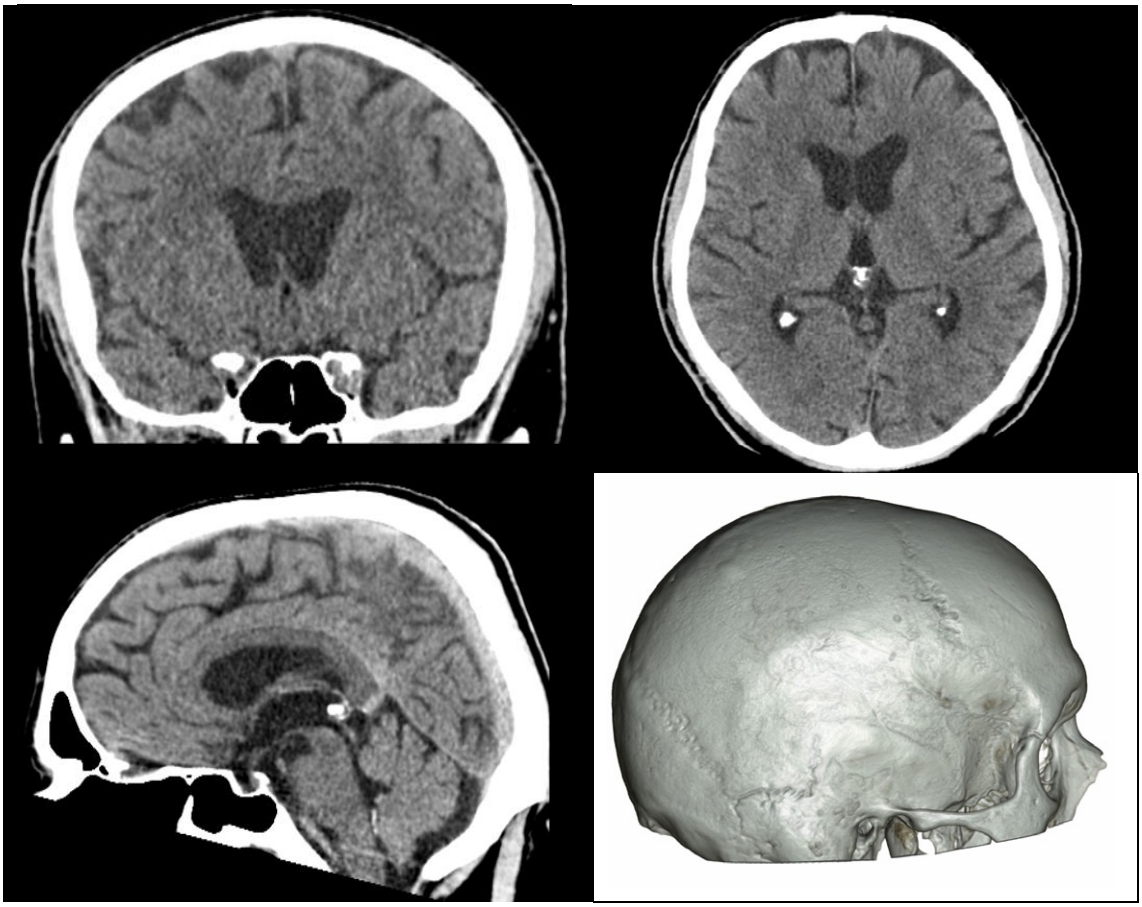

Fig. 4. Computed tomography allows for 3D imaging with superior low contrast resolution compared to radiography. Above respectively reconstructions of a CT head scan in a coronal, axial and sagittal plane; and a 3D surface rendering of the skull.

Dosimetry in CT is based on the concepts of the CT dose index (CTDIvol, mGy) and the dose-length product (DLP, mGy.cm). The CTDIvol represents the average dose that is absorbed during one rotation in the axial plane of either a head phantom or a body phantom (these are cylindrical PMMA phantoms of respectively 16 and 32 cm diameter and 15 cm length). The DLP is the product of the CTDIvol and the actually exposed range. The CTDI takes into account the effects on patient dose of tube voltage, rotation time of the x-ray tube, and tube current, but only represents the local dose within the scanned range. In addition, the DLP takes into account the extent of the exposed range, and therefore DLP represents best the patient dose. DLP can be converted into effective dose by using appropriate effective dose conversion factors or preferably dedicated software like the ImPACT CT dosimetry calculator (Shrimpton et al.).

Like in radiography the DICOM header in CT generally also stores the CTDIvol and DLP. Ideally, these quantities could be retrieved with a simple query from the digital archive in which the CT scans are being stored; however, unfortunately such functionality is not always available. European countries defined diagnostic reference levels for selected acquisitions projections in CT of adults; some countries also established diagnostic reference levels for pediatric CT.

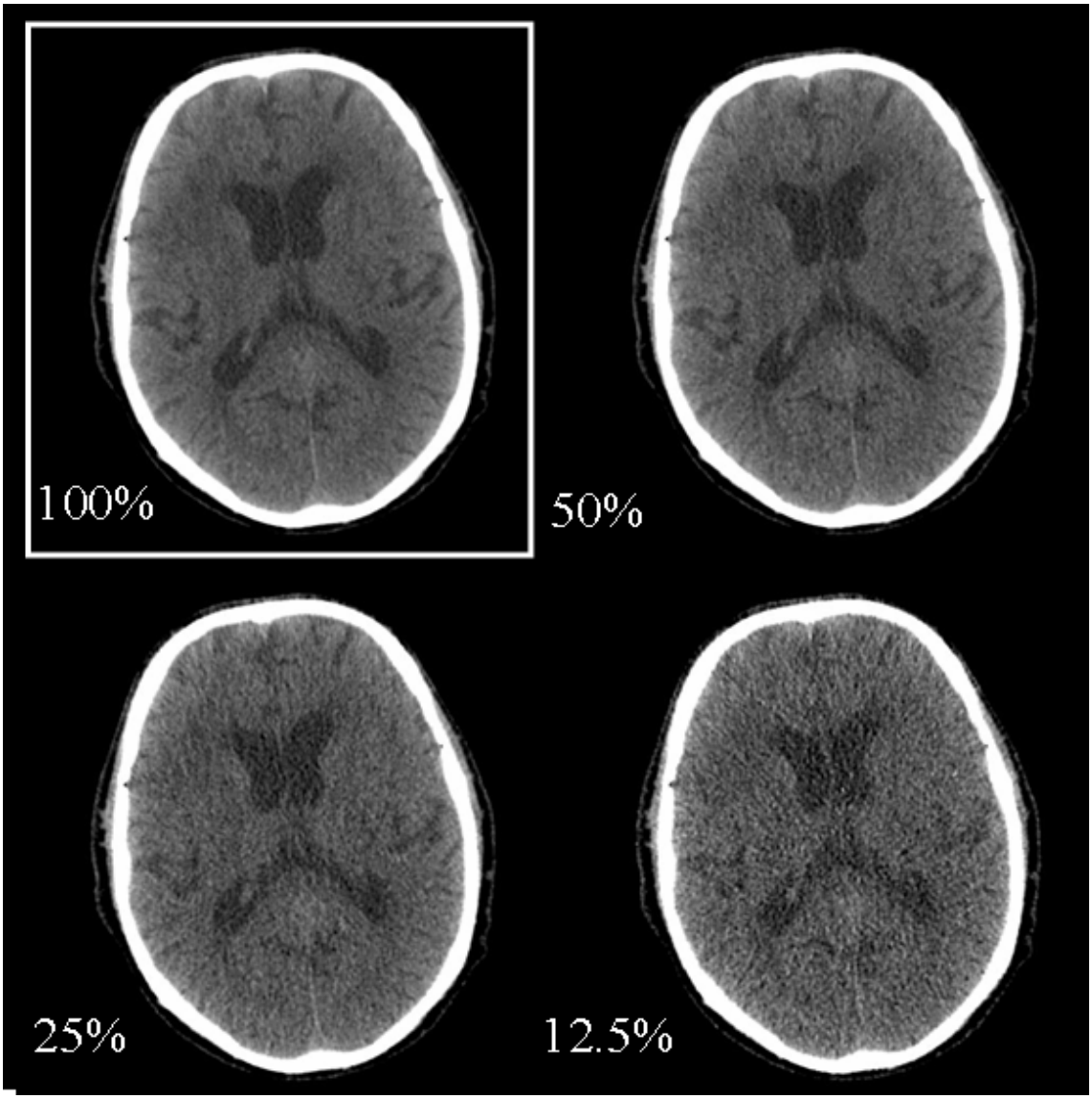

Fig. 5. Four axial images of the brain. The 100% image is acquired as a clinically indicated scan. The other images represent the image quality that would be achieved when a 50%, 25%, and 12.5% lower tube current would have been used. These brain images at reduced dose levels are not scanned images but they are simulated to using a dedicated software algorithm [Joemai et al.]. Note the increased noise in the low dose images. Such low dose simulations allow for studying the image quality of low dose acquisition protocols in observer studies without the need of repeating the CT examination on patients.

Clinical exposures

Table 1 provides an indication of the radiation exposure from natural and medical exposures. The yearly dose limits can be regarded as a frame of reference. For most radiographs of adults, exposures are well below 0.1 mSv; and of the order of magnitude of natural exposures like one week skiing or an intercontinental flight. Exposures from radiographs of extremities are extremely low; radiographs of the abdomen are relatively high and may be several tenths of a millisievert. For CT exposures can be several millisieverts, but remain in general below 10 mSv; this is true for both CT scans of adults and for proper optimized acquisition protocols for pediatric CT.

Table 1. Typical effective doses for exposures to natural and medical sources of ionizing radiation.

Natural exposures and dose limits	
One week skiing	0.015 mSv
Intercontinental flight (London-Los Angeles)	0.08 mSv
Population exposure to natural sources of radiation per year	2.5 mSv/year
Public dose limit per year	1 mSv/year
Occupational dose limit per year	20 mSv/year
Medical exposures radiography, adults	
Dental radiography	< 0.001 mSv
Extremities (e.g. knee, ankle, or elbow)	< 0.001 mSv
Chest, frontal view	0.02 mSv
Chest, lateral view	0.04 mSv
Pelvis, frontal view	0.3 mSv
Abdomen, frontal view	0.5 mSv
Medical exposures CT, adults	
CT head, stroke	1 mSv
CT chest, pulmonary embolism	4 mSv
CT spine, fracture L1	4 mSv
CT abdomen, abscess	8 mSv
Medical exposures, pediatric	
CT head, follow up of hydrocephalus, 4-6 year old child	1 mSv
CT head, battered child or trauma, 1-12 months old child	2 mSv
CT chest, bronchiectasis, 4-6 year old child	3 mSv
CT chest, congenital abnormality, 1-12 months old child	4 mSv

Future developments

Further optimization of diagnostic medical imaging with x-rays can be expected. This will be mainly achieved by technical improvements of current technologies and the development of new technologies. A recent and particular dose efficient technology for detecting x-rays, by photon counting detectors, is already implemented in mammography and may find its way to other imaging modalities. In computed

tomography the development of new reconstruction algorithms may allow for lower dose acquisitions for certain, yet to be established, clinical applications. In cardiac CT examinations an enormous reduction of patient dose becomes available for scanners that allow for prospective, ECG triggered, scanning of the heart. Sliding collimators that optimize the beam width at the start and the end of a helical acquisition contribute to the optimization of helical CT acquisitions. Volumetric CT, which allows for scanning entire organs in one single rotation allows for dose reduction for certain acquisition protocols [Kroft et al.]. Tomosynthesis, which is already a clinical application in mammography, overcomes to a certain extend the problem of overprojection in projection mammography, and is expected to be introduced also for chest radiography. Compared to CT tomosynthesis can be performed at relatively low doses, similar to doses in radiography, but does not yield the excellent 3D image quality of CT. However, dedicated ultralow dose acquisitions in CT, combined with dedicated reconstruction and noise reduction algorithms, may provide better performance compared to tomosynthesis. Dedicated low dose CT of the breasts has for example been developed, and it offers the opportunity to scan the breast in real 3D, with superior image quality compared to projection mammography or tomosynthesis of the breast, and at the same dose of projection mammography [Boone et al.]. Low dose cone beam CT scanners that are available for dental applications may also provide opportunities for low dose scanning of other parts of the skull like the inner ear.

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Radiation epidemiology

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Radioecology and environmental exposure pathways

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Abstract
Radioecology ("radiation ecology") is the scientific field encompassing the relationships between ionizing radiation or radioactive materials and the environment or subunits thereof. Radioecology and its study constitutes an important component of radiological protection of both humans and environment mainly, although not solely, through its relevance in understanding and describing environmental exposure pathways and quantifying radionuclide transport along them. Such pathways can be described as the route radioactive substances take from their source to their end point and how humans or biota can be exposed to the substance. This lecture introduces the basics of radioecology and discusses the role of radioecology with respect to environmental exposure pathways for radioactive contaminants. It presents the fundamental concepts of radioecology, the tools used by radioecologists with respect to the study of exposure pathways and future directions of research in the field.

Radioecology and environmental exposure pathways
Radioecology is the term used to describe the scientific discipline concerned with the relationship between ionizing radiation and radioactive substances and the environment or its constituents. These constituents may be populations, communities, ecosystems or biomes and include both humans and biota. Radiobiology on the other hand differs from radioecology with respect to its focus on the effects of ionizing and non-ionizing radiation on biological systems – from the molecular to the individual organism. Three main scientific areas are of interest with respect to radioecology, these being:

1. Radionuclide movement and accumulation in environmental components such as soil, water and biota and within ecosystems;
2. The effects of ionizing radiation upon species, populations, communities and ecosystems;
3. The use of radionuclides and ionizing radiation in understanding the structure, form and function of ecosystems and the constituent parts of those ecosystems.

The pursuit of radioecology as a discipline can be traced back to the early 1940's with the formation of the Applied Fisheries Laboratory in the United States, the purpose of which was to elucidate the effects of releases of radioactive substances to the Columbia River as a result of operation of reactors at the Hanford site (see Whicker and Schultz, 1982). Since then radioecology has matured as a science, that maturation having occurred, somewhat sporadically, during periods such as the early years of nuclear power, the decades of nuclear weapons testing and in the aftermath of accidents such as those at Chernobyl in 1986 and Kyshtym and Windscale in 1957.

In recent years radioecology has addressed new challenges such as those afforded by non-nuclear industries, the nuclear legacies of many countries and the increased focus on protection of the environment as well as man. As radioecology has grown as a discipline, the fundamental set of questions that its pursuit seeks to answer has been distilled to perhaps four that serve to adequately describe the scientific challenges facing radioecology:

1. How, at what rate and to what extent do radionuclides move between parts of an ecosystem and what are the mechanisms and pathways by and along which such transfers occur?
2. What are the concentrations and resultant doses from radionuclides in ecosystem components in relation to the overall contamination levels in the environment?
3. What are the long-term effects and behaviour of these contaminants in the environment?
4. What environmental concentration of radionuclides will result in effects at population level?

Intrinsically linked to these four fundamental questions, and therefore to the study of radioecology is the concept of the *environmental exposure pathway*.

Environmental exposure pathways

An environmental exposure pathway can be defined as the path a substance travels from its source to its environmental endpoint and how people or biota can be exposed to that substance. In essence, environmental exposure pathways can be considered as consisting of five common components:

1. the source of the contamination (e.g. point discharge from a power plant, fallout, abandoned source, etc.);
2. an environmental media and transport mechanism (e.g. soil – soil water, physical transport, bioturbation; water – transport as dissolved substance, particulate transport etc.);
3. an exposure point (i.e. a geographic or temporal point at whicgh the exposure actually occurs);
4. a route of exposure (e.g. eating contaminated food or prey, drinking contaminated water, inhalation of contaminated soil or dust)
5. a receptor population (e.g. rabbits living in contaminated soil, people eating contaminated fish, etc.).

Consideration of these five components of environmental exposure pathways indicates that the nature of the pathway is to a large extent governed by the nature of the environment in which the contaminant is present and the receptor population being considered. Consequently there are a large number of potential exposure pathways (see Figure 1) some of which are generally more dominant than others and some of which are relatively specific to certain isotopes, geographic regions, areas or populations. For the purposes of elaboration it is therefore convenient to breakdown the discussion of environmental exposure pathways with respect to environment "types" – terrestrial, marine and freshwater.

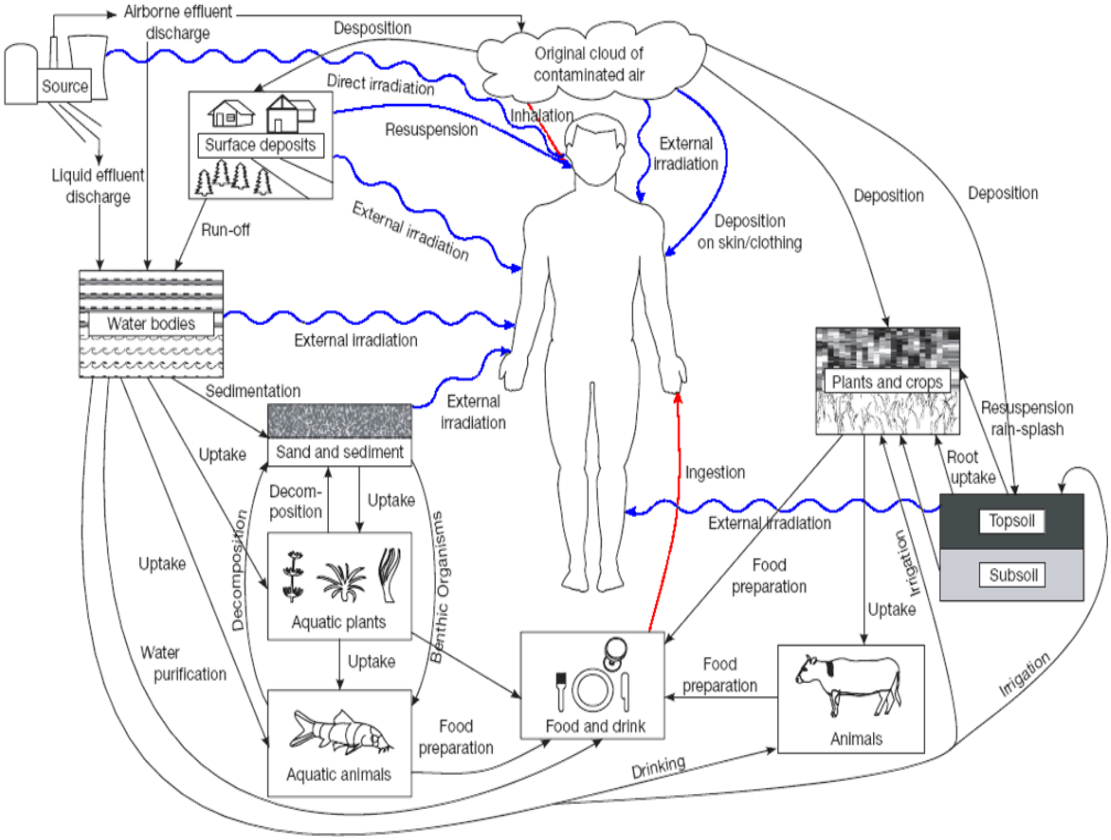

Fig. 1. Potential environmental exposure pathways for radioactive substances.

The isotopes of most interest with respect to radioecology and exposure pathways have varied over time and with respect to which source terms were of most relevance. In general those isotopes produced by fission (controlled or uncontrolled) or nuclear cycle processes and which have a half-life long enough to represent a potential impact on humans or the environment dominate radioecological studies. Typically the suite includes isotopes of plutonium, cesium, strontium, iodine, technetium, uranium, radium, thorium, carbon, hydrogen and cobalt among others.

Terrestrial exposure pathways

Although terrestrial environments can be broken down with respect to climate etc., the main mechanisms and processes involved in governing terrestrial exposure pathways are the same for all although the pathways may differ in their relative importances. As an example, wind resuspension and inhalation of contaminated soils may be an exposure pathway common to both European agricultural ecosystems and tropical deserts but will probably be of greater significance in the latter. Numerous transport processes are of relevance with respect to the terrestrial environment and can include transport by groundwater, particulate transport with soil particles, resuspension of soils and dusts by wind or rain, runoff/snowmelt etc. The most direct exposure pathways for terrestrial environments include direct exposure of the organism to external radiation from deposited radionuclides or ingestion/inhalation of contaminated soils or dusts. Uptake of radionuclides by plant species with subsequent transfer to humans or biota via the food chain is perhaps the exposure pathway that has received most attention over the decades.

Radioecology has played an important role in the study of terrestrial exposure pathways although the years up to the early 1990s were primarily focussed on studies regarding human exposure. This resulted in much information dealing with agricultural systems as the main source of food for large populations with a concomitant neglect of forest, semi-natural and natural and tropical ecosystems. This earlier neglect has been addressed in the last two decades with the radioecology of these ecosystems being studied in greater detail. The heightened vulnerability of Arctic populations to radioactive contaminants such as isotopes of cesium appears to be caused by a short environmental exposure pathway that efficiently transfers cesium to a human endpoint. Deposited cesium is taken up efficiently by Arctic vegetative and lichen species due to the nature of the Arctic climatic regime and adaptations of Arctic biota and is transferred to reindeer which feed almost exclusively on these plants. Arctic populations consuming large amounts of reindeer and foodstuffs such as fungi and berries which also accumulate cesium can be exposed to higher levels of radioactive contaminants that would be expected for a similar amount of environmental contamination in more temperate regions.

Marine and freshwater exposure pathways

Although there can be differences in the radioecological behaviour of radionuclides in freshwater and marine systems, the exposure pathways are generally the same. Direct exposure of humans and biota can occur via external radiation from contaminated water and sediments. Ingestion of contaminated water or sediment or inhalation of spray can also result in exposures although the main exposure pathway is ingestion via the food chain. Benthic biota can be exposed due to living in and on contaminated sediments. The extent to which radionuclides are transferred along aquatic pathways is dependant on a range of factors including organic matter content, suspended sediment content, salinity, isotope type, residence times of waters, nutrient status, species etc. Radioecological studies of freshwater and marine systems have been plentiful although the vast majority of these have been concerned with radiocesium and ^{90}Sr. An important radioecological control in aquatic systems is the salinity of the system – uptake of

isotopes such as ^{137}Cs and ^{90}Sr in aquatic fish species has repeatedly shown to be higher in systems with low levels of K^+ and Ca^{2+}.

Marine systems have been the focus of much radioecological study over the years and this work has gone a long way towards a better understanding of exposure pathways in this environment. For marine systems, radionuclides present in the particulate or colloidal form may tend to be accumulated to a greater extent by marine animal species than those present in dissolved form – this being in contrast to freshwater and terrestrial systems where the converse generally tends to be true (hence the predominance of ^{90}Sr and ^{137}Cs in the exposure pathways of the latter two environments). This is primarily due to the chemical nature of the marine environment which features large amounts of chemical analogues for both strontium and cesium and the fact that many marine species on the lower rungs of the food chain (and hence at the early stages of exposure pathways) are filter feeders, a feeding mechanism ensuring high uptake of particulate bound radionuclides.

Quantifying transfer in exposure pathways

An important focus of radioecology regarding environmental exposure pathways has been the means of quantifying radionuclide transfer along these pathways and a number of means of doing so have been developed. The first of these can be encapsulated by three "factors" used within radioecology to describe transfer:

$$\text{Transfer Factor (TF)} = \frac{\text{Nuclide conc. in plant tissue (Bq/kg)}}{\text{Nuclide conc. in soil (Bq/kg)}}$$

$$\text{Concentration Factor (CF)} = \frac{\text{Nuclide conc. in biota tissue (Bq/kg)}}{\text{Nuclide conc. in filtered water (Bq/l)}}$$

$$\text{Aggregated Transfer Coefficient } (T_{ag}) = \frac{\text{Nuclide conc. in plant tissue (Bq/kg)}}{\text{Nuclide deposition on soil (Bq/m}^2\text{)}}$$

Such factors have proved useful in quantifying transfer along exposure pathways for assessment purposes related to regulation, emergency response, rehabilitatory efforts etc. Large scale efforts in the evaluation and collation of empirical data sets regarding radionuclide transfer as expressed by factors such as those above have resulted in compendia published by bodies such as the IAEA (IAEA, 2004; IAEA, 2010). An important and significant common characteristic of such compendia is the high degree of variability in the data sets (often orders of magnitude) for individual isotopes reflecting the complexity of the processes involved in the transfer of radionuclides to plants and animals (see Figure 2). Addressing and accounting for this variability has proven to be a significant driver in radioecological research over the years.

It should also be considered that the transfer of radionuclides to animals or plants is largely a dynamic process and that quantification of transfer by expressions such as those above only represents a temporal "snapshot" and one that is likely to vary considerably depending on the time it was taken if equilibrium had not been achieved.

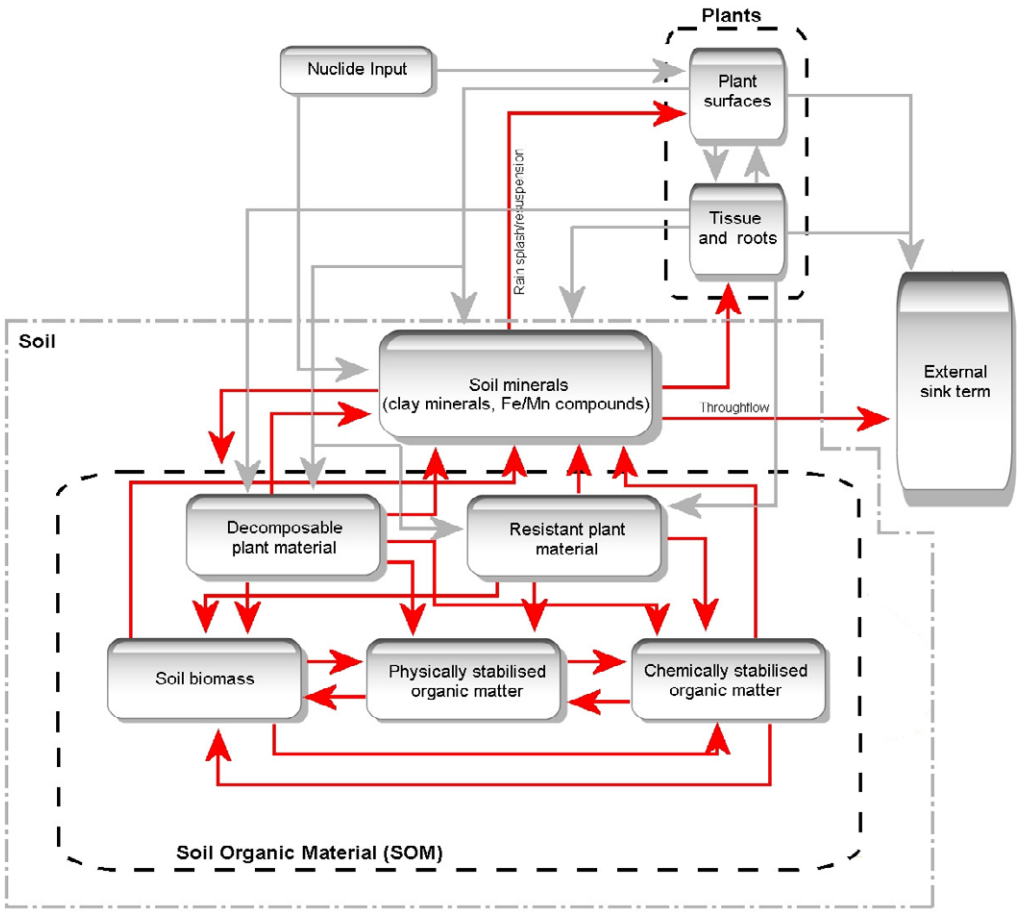

Fig. 2. Schematic of soil processes governing transfer of radionuclides from soil to plant.

For the many situations or radionuclides where data is lacking or the inherent weaknesses of the CF/TF approach are unacceptable (see Green et al., 1997), alternative methods for estimating transfer have been developed. Mechanistic kinetic approaches have been available since the 1980s (see Whicker and Schulz, 1982) and such methods include the advantage that the transfer at any time can be estimated as opposed to the equilibrium assumption inherent in CF/TF approaches. Despite this, mechanistic kinetic approaches have still not fully fulfilled their potential. Allometry, which relates radionuclide uptake to body size, has proved a potentially useful avenue of research in describing transfer (see Higley et al, 2003) as has phylogenetic approaches to extrapolate information on transfer of radionuclides from information derived for one species to another (see Willey and Fawcett, 2006).

Bioavailability

A key concept in the study of environmental exposure pathways is that of bioavailability - bioavailability generally refers to the extent to which humans and ecological receptors are exposed to contaminants in soil or sediment (see Desmet et al, 1991) and with respect to radioecology can be understood as the quantity of a radionuclide available for biological uptake. It is accepted that bioavailability is the factor that ultimately controls the exposure of both humans and biota and therefore the

impact of contaminant radionuclides. In studying bioavailability within radioecology, much emphasis has been upon the transfer of radionuclides from soil to plant and an extensive corpus exists in this field although much work remains to be done. Factors such as climate, soil properties and radionuclide speciation have been focussed upon as controllers of terrestrial bioavailability and have been the subject of radioecology research over six decades. As ^{137}Cs came to be recognised as a radionuclide of importance during the period of atmospheric weapons testing and in the aftermath of the Chernobyl accident, significant effort was expended on understanding the bioavailability of this isotope. Cs-137 and study of its behaviour in the environment highlighted the complexity of the problem and although important advances were made in understanding, for example, the role of organic soil materials and potassium in its environmental transfer, much work remains to be done in order to develop mechanistic models describing its uptake by plants and further transfer through ecosystems although some models are currently available. In this respect, bioavailability is a significant area of research in radioecology as models and predictions with respect to environmental exposure pathways are reliant on a thorough understanding of the processes and mechanisms involved.

Current and future directions

Perhaps the most significant development of relevance to radioecology in recent years has been the acceptance that one can not always assume that if humans are protected from the effects of radioactive contaminants, then so is the environment. This acceptance has led to consideration of environmental exposure pathways that are not strictly directed towards the human endpoint but which are of relevance to wildlife species. This expansion in the scope of the pathways and species that must be considered has presented new challenges for radioecology and will continue to be a focus in coming years. In addition, radioactive contaminants occur in the environment as part of a large suite of other contaminants. Radioecology has begun to recognise this and view radioactive contaminants and exposure pathways as part of the wider spectrum of contaminants.

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Malicious events: scenarios, consequences and response

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Abstract

The potential use of radioactive materials by individuals with malicious intent has become an area of increasing international concern in recent years. This refresher course session will cover a number of key areas:

– Consideration of a range of potential scenarios and consequences both in the short and longer term;
– Examination of how lessons identified from past accidents can be used to develop preparedness in this area;
– Discussion of the key elements of planning arrangements required to respond effectively to a scenario of this nature; and
– Reviewing the key international planning guidance that is available to assist in the preparation for such scenarios.

Indoor radon sources, remediation and prevention in new construction

Arvela, Hannu
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Abstract

Radon is the second leading cause of lung cancer in the general population, after smoking. Strategies for both radon prevention in new dwellings and mitigation in existing dwellings are needed to achieve an overall risk reduction. The inflow of radon laden soil air into living spaces is the main source of elevated radon concentrations.

The methods of indoor radon remediation are normally based on depressurization of soil under floor construction and decreasing soil air radon concentration. Sealing of entry routes, improvement of air exchange rate and decreasing underpressure level in living spaces has also been used. Sub-slab depressurization (SSD) and radon well have been found the most efficient methods. Typical reduction factors are 70-90%. Ventilation based methods and sealing of entry routes result normally in a clearly lower radon reduction. A SSD system comprises a cavity in the ground immediately under the floor slab. This suction cavity is linked by pipe work to the outside. A radon well sucks air from a well pit constructed outside the building. This air flow reduces the soil air concentration in a wide area round the well. A radon well can be applied on coarse soil types.

Most prevention strategies address steps to limit soil gas infiltration due to air pressure differences between the soil and the indoor occupied space. Both sealing based and sub-slab depressurization based measures have been recommended. The available research results show an average efficiency of 50 % for passive prevention measures. Fan activated sub-slab depressurization reduces radon concentration by up to over 95%. Widespread use of radon resistant construction has resulted in Finland in an average reduction of 30% in radon concentrations of new low rise residential buildings.

1 Introduction

Radon is the second leading cause of lung cancer in the general population, after smoking. Strategies for both radon prevention in new dwellings and mitigation in existing dwellings are needed to achieve an overall risk reduction. In EU countries the typical proportion of low rise residential houses exceeding the national reference value of 400 Bq/m^3 is 0.1% - 5%. Presently the reference value for design and new construction is typically 200 Bq/m^3. In the case of this lower reference value the proportion is typically 0.5% - 15%. New international recommendations have resulted also in a reference value of 100 Bq/m^3 (WHO 2009). The proportion of houses above

100 Bq/m^3 may be 5% - 50%. Respectively the country specific number of houses needing remedial measures can be as high as 400 000 - 2 000 000.

The inflow of radon-bearing soil air into living spaces is the main source of elevated radon concentrations. Radon prevention strategies are therefore based on limitation of soil gas infiltration through reduction of the air pressure difference between soil and indoor spaces and on decreasing the radon concentration in soil air. Prevention goals and strategies are especially important because neglecting the strategy leads to increase of the total number of dwellings with elevated indoor radon.

In this presentation the new WHO radon handbook, several national guides and website material of remedial measures and prevention in new construction has been utilized. Special emphasis has been given also to Finnish experience, especially regarding radon wells and new sample survey results of radon prevention in new construction.

2 Radon sources

Inflow of radon laden soil air is the main entry mechanism increasing radon concentrations in indoor spaces. This inflow is forced by underpressure in rooms above the floor construction compared with soil or spaces beneath the floor. The underpressure is created by indoor-outdoor temperature difference, wind and when in use also by mechanical ventilation. Table 1 shows typical underpressure levels in Finnish low rise residential houses when outdoor temperature is 0 deg C. Mechanical exhaust ventilation creates a high underpressure, depending on the air tightness of the building shell. In the case of a mechanical supply and exhaust ventilation the underpressure can be controlled through adjustment of air flows.

In living spaces building materials are normally only in special cases the reason to elevated indoor radon concentration above 200 Bq/m^3. If the air exchange rate is well below the recommended values (ACH 0,5 1/h), even normal radon emissions can cause elevated indoor radon concentrations.

Figure 1 presents typical entry routes in UK. Figure 2 presents a typical Finnish slab-on-grade foundation. Permeable light-weight concrete blocks promote flow of soil air into living spaces.

Table 1. Typical underpressure levels in Finnish low rise residential houses when outdoor temperature is 0 deg C (STUK 2008).

Ventilation strategy	Typical heating season underpressure, Pa
Natural	1 - 2
Mechanical exhaust	5 - 10
Mechanical supply and exhaust (heat recovery included)	2 - 5

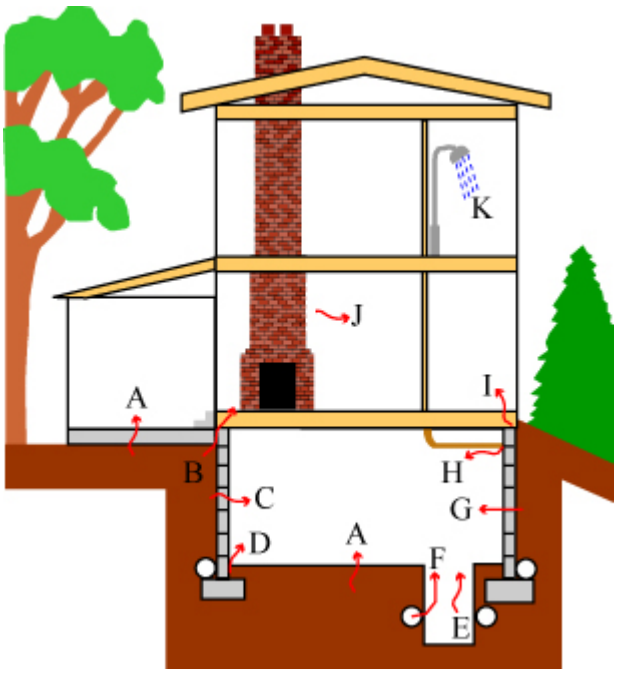

Fig. 1. Major radon entry routes of radon

A. Cracks in concrete slabs.
B. Spaces behind brick veneer walls that rest on uncapped hollow-block foundations.
C. Pores and cracks in concrete blocks.
D. Floor-wall joints.
E. Exposed soil, as in a sump or crawl space.
F. Weeping (drain) tile, if drained to an open sump.
G. Mortar joints.
H. Loose fitting pipe penetrations.
I. Open tops of block walls.
J. Building materials, such as brick, concrete, rock.
K. Well water (not commonly a major source).

Source: http://www.rpwradonwales.co.uk/

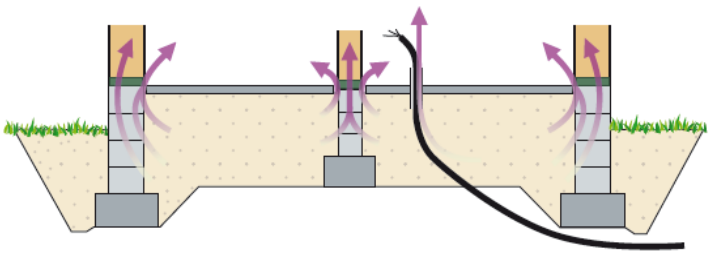

Fig. 2. Entry routes of soil air in a Finnish slab-on-grade foundation. The foundation wall is of permeable light-weight concrete blocks (STUK 2008).

3. National radon prevention and remediation programmes

In the recent WHO Radon Handbook (WHO 2009) the following key elements for successful radon prevention and mitigation actions within the framework of a national radon programme have been given.

1. Radon control actions should consider a combination of building types:
   - new and existing homes, since the greatest amount of radon exposure is generally in homes;
   - buildings where public is likely to be exposed for long periods, such as schools, preschool facilities, state-owned or leased buildings, and lodging facilities.

2. Research on buildings should be used to identify the most cost-effective radon control strategies for prevention and mitigation. Structural, foundation, and ventilation systems as well as construction practices vary from region to region. Specifically, this research should be used to develop:
   - radon prevention standards and regulations such as building codes for new dwelling construction;
   - radon mitigation standards and requirements for remediation of existing dwellings.

3. The contribution of different radon sources varies between countries and even regions. The following mechanisms may be considered:
   - pressure driven soil gas infiltration;
   - emanation of radon from building materials;
   - water transport of radon.

4. Appropriate training and certification of building professionals should be implemented to ensure the efficiency of prevention and mitigation actions.

WHO Radon Handbook gives the following design criteria for the radon control systems for prevention as well as mitigation (WHO 2009):

- able to reduce radon concentrations considerable below the reference level
- safe and not creating back-drafting
- durable and functional for the expected life of the building;
- easy monitoring of the performance
- quiet and unobtrusive
- low cost for installation, operation and maintenance
- easy to install an additional fan when passive soil depression systems (PSD) are used.

4 Remediation in existing dwellings

4.1 Overview on the efficiency of remediation methods

The methods of indoor radon remediation are normally based on the following principles.

1. Depressurization of soil under floor construction and decreasing of soil air radon concentration
2. Sealing of entry routes
3. Improvement of air exchange rate and/or decreasing underpressure level in living spaces
4. Combination of these methods

Table 2 shows the efficiency of different methods in UK and the influence of house characteristics (Naismith et al. 1998). Reduction factors of 1,5 -3 are typical for the methods based on sealing and ventilation corresponding to a percentage reduction of 30-70 %. Figure 3 shows radon reduction factors achieved using various mitigation methods in Finland (STUK 2008). The Finnish results are based on a questionnaire study in 400 houses. Well designed and implemented mitigations result often in reductions which are better than the typical reduction (middle 50 % of results) in figure 3.

Table 2. The influence of house characteristics on radon reduction factors. Reduction factors has been defined as the ratio of radon concentration before and after the remedial measure. A reduction of 50 % corresponds in the table to the reduction factor of 2.0 (Naismith et al. 1998).

Remedial measure	Grouping of data	Number of homes	GM of initial radon level	GM reduction factor	Percentage of homes reduced below the action level
Sump	Installed by major contractor, post-war house	166	540	17	95
	Installed by major contractor, pre-war house	102	630	9	77
	Installed by local builder, post-war house	65	430	11	82
	Installed by local builder, pre-war house	43	530	6	70
	Installed by householder, post-war house	31	550	7	77
	Installed by householder, pre-war house	22	560	4	64
Positive ventilation	Living room and bedroom on different floors	43	500	1.9	42
	Living room and bedroom both on ground floor	51	580	3.7	63
Additional permanent ventilation of the house	Without additional sealing of cracks or service entry points	59	370	1.5	49
	With additional sealing of cracks or service entry points	56	430	2.4	60
Mechanical ventilation of the underfloor space	All data	63	540	2.8	54
Natural ventilation of the underfloor space	All data	171	390	1.9	53
Sealing of cracks or service entry points	All data	71	470	1.7	41

Sub-slab depressurization (SSD) and has been found the most efficient method. Typical reduction factors are 70-90%. Ventilation based methods and sealing of entry routes result normally in clearly lower radon reduction. In the Irish guide to likely effectiveness of remediation techniques (RPI, 2009, Fig. 4) the fan-assisted sump is the only technique with a high likelihood of success above 800 Bq/m^3. The target concentration in Ireland is 200 Bq/m^3 or below. The radon concentration range for good success for other methods is typically 200 - 700 Bq/m^3, for sealing and improving ventilation 200 - 400 Bq/m^3. This guidance is agreement with the Finnish results in figure 3. Radon well technique (chapter 4.3) has been used in Finland and Sweden. The results are on average as good as those for SSD, figure 3.

4.2 Sub-slab depressurization (SSD)

Sub-slab-depressurization is the most common and also most efficient methods in radon remediation. Following names have been used for this system:

- Sub-slab depressurization (SSD)
- Sub-floor depressurization
- Active or passive sub-slab depressurization (ASD, PSD)
- Radon sump (UK, Ireland)
- Sub-slab suction system
- Fan-assisted sump

SSD is based on two processes, first on depressurization of soil beneath the leaking floor structures and second on decreasing the radon concentration of the soil air flowing through the floor. This depressure reverses the pressure differential between soil under floor and the room above. A radon sump comprises a cavity (suction pit) about the size of a bucket (10 - 30 litres) in the ground immediately under the floor slab that is open to the surrounding under-floor hardcore. The sump is linked by pipe work to the outside. A suction pit is needed in order to minimize the flow losses in soil round the sump. The pipe penetrations and possible gaps and cracks in floor construction close to the suction pit should be sealed. Leakage through such gaps can decrease the efficiency remarkably.

Using a sump with an electrical fan is called as an active system. A reduced efficiency can be achieved through a passive sub-slab depressurization. In this case the system works without a fan. Typical radon reduction for an active SSD is 70-90%. In difficult cases additional sealing work is needed in order to achieve a low enough radon concentration.

The best location of the suction pit is in the central area of the floor slab or in a part of floor area which is divided by load bearing foundation walls into separate blocks. In the case this kind of foundation walls prevent the extension to other parts of foundation, two or more suction pits may be needed. In the case of one undivided floor block, one suction pit is normally sufficient for a floor area of 250 m^2 (BRE 2003) or 120 m^2 (STUK 2008), depending on the soil and foundation constructions.

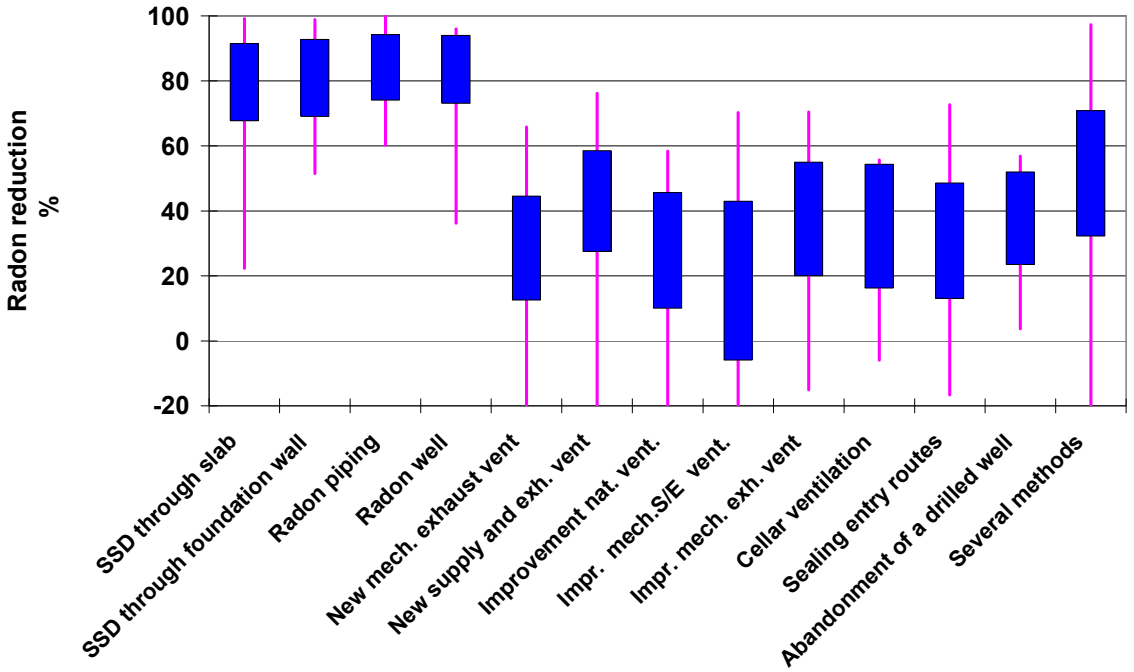

Fig. 3. Radon reduction factors achieved using various mitigation methods, minimum, 25% percentile, 75% percentile and maximum. The results are based on a questionnaire study in 400 Finnish houses. SSD: Sub-Slab Depressurization (STUK 2008).

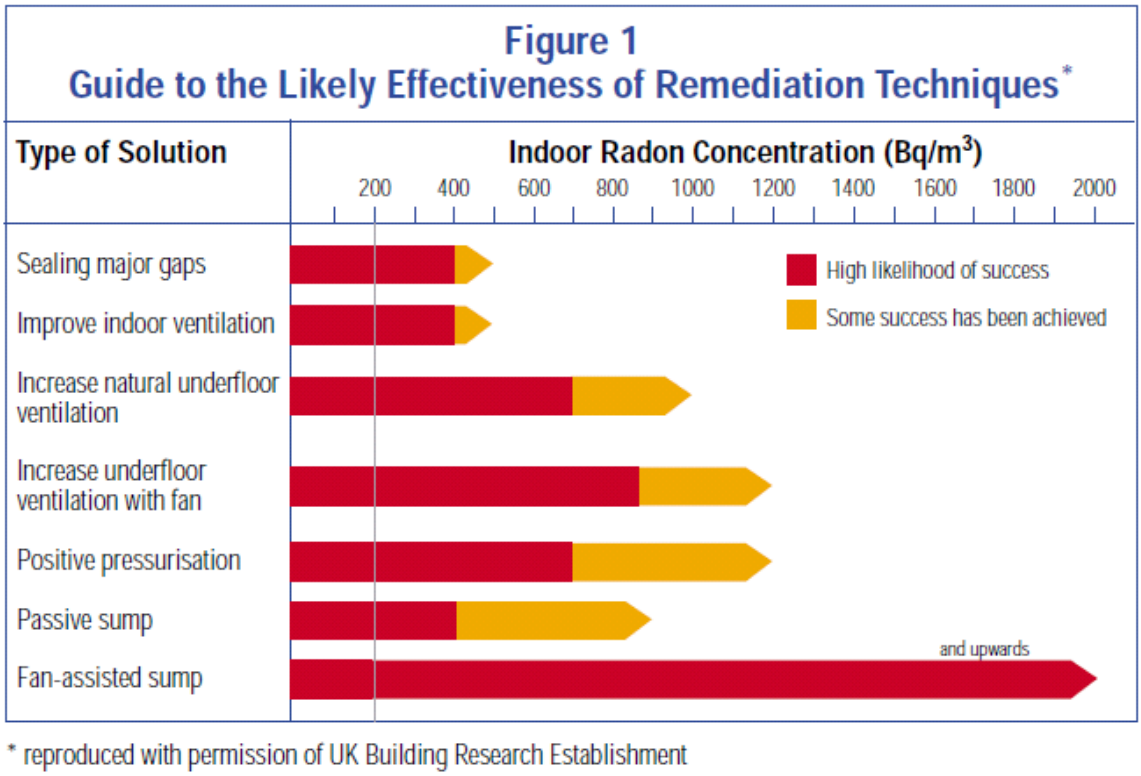

Fig. 4. Guide to likely effectiveness of solutions in Ireland (RPI 2009).

In the basic UK-guide (BRE 2003) **the standard sump** should be located close to the centre of the dwelling. It can be constructed using a paving slab 600*600 mm and a few rows of bricks with air gaps left between them or by using a prefabricated sump (Figure 5A). Typical fan power is approx. 70 watts.

A simple **mini sump** (Figure 5B) can be constructed by breaking out or core drilling a 120 mm diameter hole in the floor slab and excavating about a bucketful (10 -30 litres) of material from below (clearing out a space approximately 200mm in radius). A good volume in the suction pit decreases air flow resistance and increases thus the extension of the air flow.

A SSD can be constructed also from outside the building, According to the BRE guide, an **externally excavated mini sump**, figure 5C can be constructed by breaking out or core drilling a 120mm diameter hole through the external wall just below the floor slab and excavating about a bucketful of material (clearing out a space approximately 200 mm in radius).

Passive sump systems

Experience has shown that in cases where the indoor radon level is relatively low, i.e. just above the action level, it is possible, in the right circumstances, to operate a sump system *passively* – without the need for an electric fan (BRE 2003). If it works, a passive sump system is always preferable to a fan driven one as it is easier and cheaper to install, run and maintain, and will be quieter. However they are not as effective as systems fitted with fans. On average passive sump systems produce a 50% reduction in radon levels, whereas fan powered systems achieve relatively larger reductions – often more than 90%. The important point is that it is a simple task to add an electric fan later if the passive system does not adequately reduce the radon level.

Multiple suction points

Example in figure 6 shows a Czech application (Jiranek 2003) of multiple suction point system in a typical Czech cellar house. The effectiveness of such systems varies between 70 and 98 %, which means that indoor radon concentration decreases to 30 % up to 2 % of the initial values. The effectiveness is mainly influenced by the vertical profile of soil permeability and by the air tightness of the building substructure.

Depressurized hollow floor

Figure 7 shows an application of depressurized hollow floor (Swiss Radon Handbook). The system is appropriate when radon bearing soil air is coming mainly through the floor. In this example a reduction from 1400 Bq/m^3 to 50 Bq/m^3 was achieved.

Installation of radon piping and a SSD system

In cases where floors in old houses have been opened and renewed, it is practical to install a radon piping below the new floor. Figure 8 shows a French example of piping installation (CSTB 2008). Piping has been used also for controlling the leakages through the lower parts of the walls. Piping has been installed into a permeable aggregate layer and has been covered with geotextile and plastic membrane. Radon concentrations before the remedy were 1300 - 2800 Bq/m^3. Radon reductions up to 90% were achieved.

Depressurization of drainage system

In houses with existing drainage system depressurization of the drainage piping can be utilized. According to the Swiss Radon Handbook this system is effective only in 10 % of cases. US EPA (USEPA 2003) gives an average reduction factor of 50 %. The efficiency can be enhanced through installation of sealed valves or drain traps at exits of the drainage system. A sufficient quantity of water opens the valves. In Finland the method has been tested with a success in several houses. Creating a good depressure in the piping is the key problems. Further studies are needed for a good guide material for this method.

4.3 Radon well

Principle of radon well

A radon well is constructed outside the house and the well sucks air from the soil from the depth of 3 - 5 meters (STUK 2008). Figure 9 shows the principle of a radon well. Soil ventilation caused by the exhaust fan decreases efficiently the radon concentration of the soil air below the house foundations. Close to the house a radon well can also create underpressure below floor slab. A single radon well can reduce the radon concentration in several dwellings at the distance of up to 20-30 meters.

Building a radon well requires experience and expertise in soil geology. The efficiency of a radon well is affected e.g. by soil permeability, layer structure, depth of the suction pit and the power of the suction fan. Radon well is effective only when built on coarse soils as gravel and coarse sand. If the soil is fine and impermeable, such as clay, silt and moraine, it is not possible to create an air flow field with a good extension. Also fine sands may be problematic. Soil layers with coarse and fine material affect the extension and form of the air flow field.

Figure 10 presents the construction of a radon well. The best depth for the suction pit is normally 4-5 meters, below the footings of the house. The depth of the well promotes the extension of the flow field to a wide area.

The exhaust pipe can be either on the cover of the well ring or at good distance from the well using a horizontal transfer pipe below soil surface. In the case of houses on hillside (split level house), the best location of a radon well is on the upper side of the house.

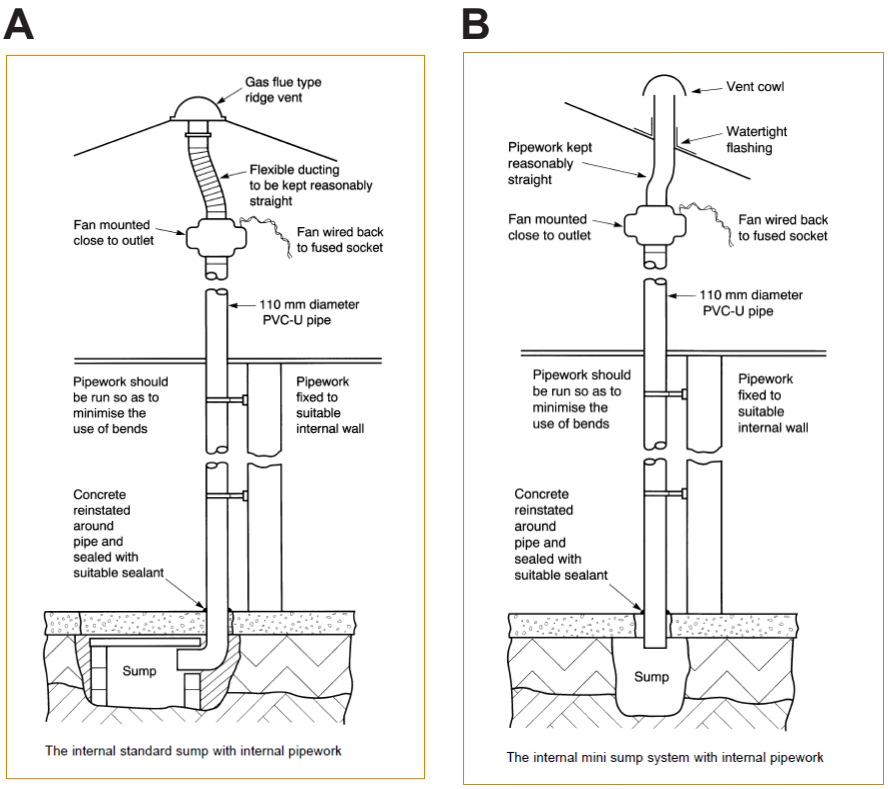

Fig. 5. Sump systems in the UK BRE guide (BRE 2003).
A: Internal standard sump system
B: Internal mini sump system
C: Externally excavated sump system

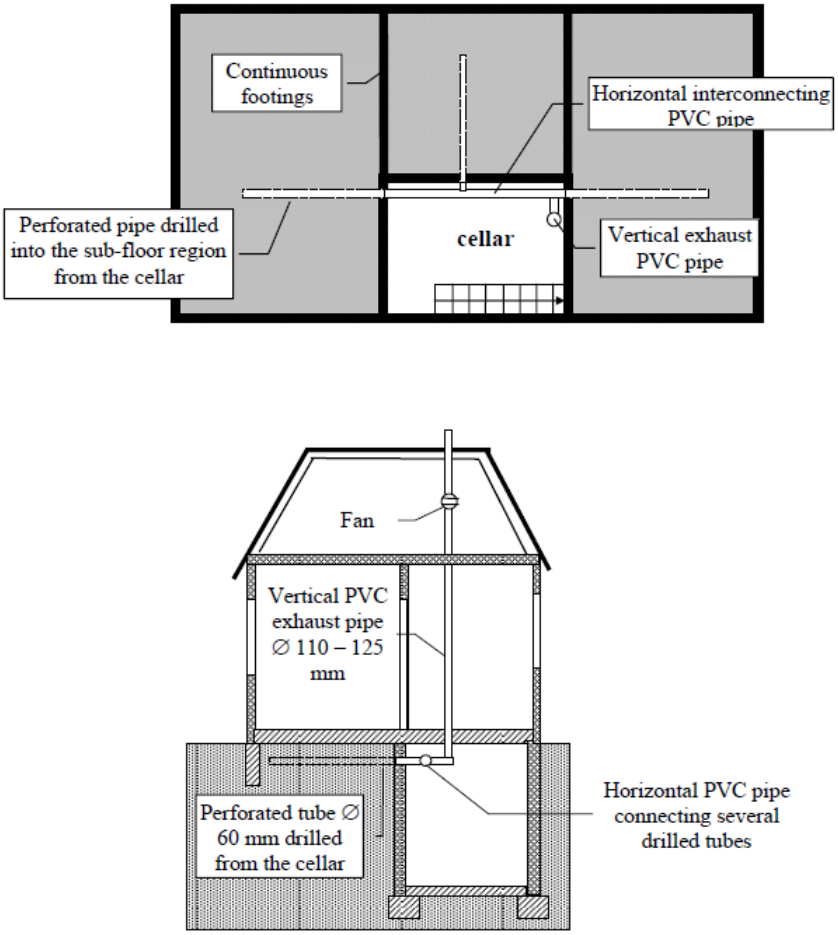

Fig. 6. Sub-slab depressurization based on perforated tubes drilled from cellar according the Czech mitigation standard (Jiranek 2003).

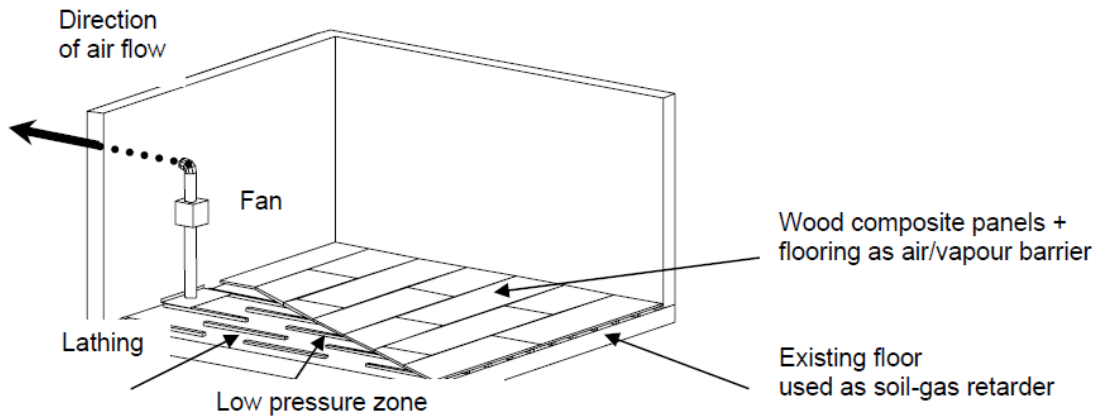

Fig. 7. Depressurized hollow floor, Swiss Radon Handbook.

Fig. 8. Sub-floor piping installation for a sub-slab depressurization system in France (CSTB 2008).

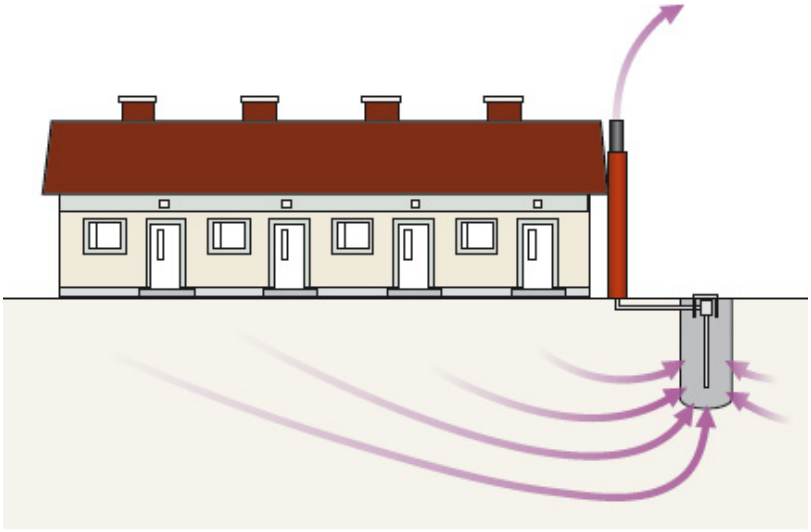

Fig. 9. Principle of a radon well. A radon well ventilates soil air and decreases the soil air radon concentration in a large area (STUK 2008).

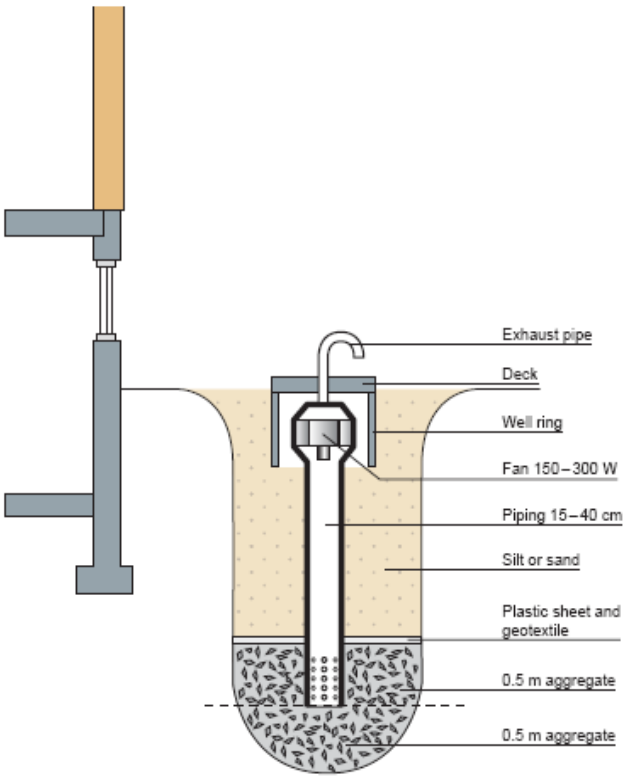

Fig. 10. Construction of a radon well (STUK 2008).

Radon well in Hollola

Figures 11 and 12 present a radon well installation of a row house company with four buildings and 20 dwellings (STUK 2008). The aim was to achieve good results using two radon wells. Fig 11A presents the exhaust fan of 300 W installed in a well ring. A transfer pipe is leaving the well ring towards the gable of the house. Fig. 11B presents the cover of the radon well and the discharge pipe installed above the eave height of the building. The thermally insulated pipe ensures that the pipe does not freeze in the winter. This high exhaust pipe releases the radon bearing exhaust air away from the yard area (including a playground). The depths of the wells are 3.5 - 4 m. The air flows measures were 68 l/s (250 m^3/h, well no 1) and 58 l/s (210 Bq/m^3, well no 2). The average reduction of indoor radon concentration, in 17 dwellings originally above 400 Bq/m^3, was 88% (Fig. 12). The best reduction rate was 99%.

Fig 11. Exhaust fan in a well ring (A) and cover of the well ring as well as the exhaust pipe installed above the eave height of the building (STUK 2008).

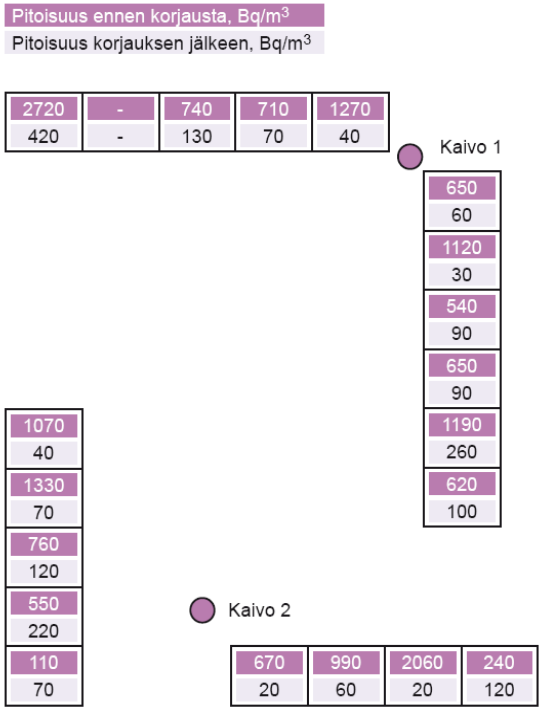

Fig. 12. Reduction of indoor radon concentration in 20 dwellings of a row house company after starting of two radon wells. Radon concentrations before and after the mitigation (STUK 2008).

4.4 Improving ventilation and sealing

Improving ventilation of occupied spaces

The British guide recommends the measures presented in figure 13. The guide emphasizes that it is important to recognise that in most cases the reduction in radon level that can be achieved by changing the way in which you ventilate your home will be small. Generally ventilation approaches are more common in mechanically ventilated schools and other large buildings than in small houses. The systems may be especially useful in the following cases.

- Building materials are the major entry route.
- There are other indoor air quality problems, for example very low air exchange rate.
- In the case of mechanical exhaust ventilation and tight building shell, reduction of underpressure level may be effective.

Radon reduction measures based on ventilation reduce radon concentration either through increased ventilation or decreased underpressure level. Reduction factors above 50% have been achieved only in cases where the original air exchange rate has been defective or when the underpressure level has been high. The typical reduction factors in Finland have been 10-40% (Fig. 3). Increasing the operation time or the power of mechanical ventilation, opening the existing or installing new fresh air vents are typical

mitigation measures. Installation of new fresh air vents does not normally result in reduction factors above 50%.

In the Finnish experience the best results have been achieved through installation of a new supply and exhaust ventilation system. The underpressure created can be minimized and controlled in this strategy, which is important in radon control.

Temperature and moisture aspect should always be taken into account when changing the ventilation or depressure conditions.

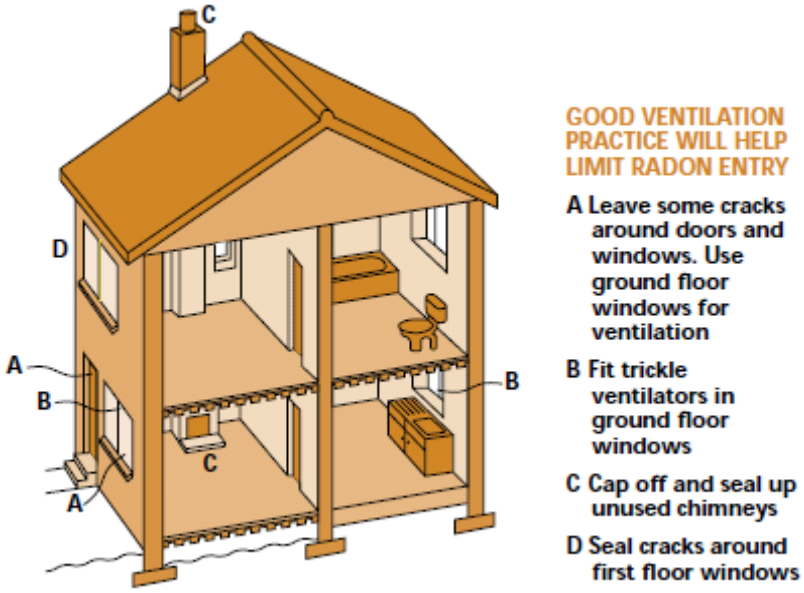

Fig. 13. Good ventilation practices in the UK guidance (BRE 2003).

Sealing entry routes

Sealing entry routes aims at reduction of leakage flows of radon bearing soil air into the living spaces. Sealing may be very demanding. In many cases the results of sealing are sufficient only when the entry routes have been sealed almost completely.

Floor joints with foundation walls of porous light-weight concrete cannot be sealed with normal sealing methods. Porous elements provide new leakage routes although the wall-floor gap has been sealed. Hollow block wall elements need similarly special consideration. Best results in Finland, when sealing the joint of floor and wall have been achieved in houses where the foundation wall is of casting concrete.

5 Radon prevention strategies

Most prevention strategies address steps to limit soil gas infiltration due to air pressure differences between the soil and the indoor occupied space. Radon prevention strategies should consider the specific mix of construction practices, radon sources, and transport mechanisms in the region or country, in order to be cost-effective. Under certain conditions a combination of strategies may be necessary such as in buildings with multiple types of foundations.

Preconstruction site assessment

The WHO Radon handbook (WHO 2009) presents following aspects on preconstruction site assessments. A number of approaches are used to assess the potential for elevated indoor radon concentrations across geographic areas. The most common approach involves mapping of indoor radon concentration in existing houses. This approach can be complemented through utilization of geological information. Areas of permeable sand and gravel e.g. can be classified to the high radon category. Another approach used in some countries, such as the Czech Republic, involves testing individual building sites prior to construction using soil gas measurements to establish a radon index for the site. The index is then used to define the degree of radon protection needed for building on that site. However, in countries including Finland, Ireland, Norway, Sweden, Switzerland, the United Kingdom, and the United States of America, the most cost-effective approach appears to be the use of radon control options in all new homes or in homes in specified areas.

Requirements in Ireland and UK

As an example the requirements in UK and Ireland are considered (BRE 2007, Ireland 2007). The basis for the requirements is the following: It is not possible to accurately predict the concentration of indoor radon likely to occur in a proposed building on the basis of a pre-construction site investigation. Both UK and Ireland have published the results of a national survey of radon levels in existing houses. Based on the results of the survey the countries have identified **High Radon Areas**. In Irish high radon areas, it is predicted that more than 10% of dwellings in the area will have radon gas concentrations above 200Bq/m^3. However, houses with high concentrations of radon gas are not confined to these areas and can occur in individual dwellings in any part of the country. In UK the Health Protection Agency defines Radon Affected Areas as those with 1% chance or more of a house having a radon concentration at or above the Action Level of 200 Bq/m^3.

In Irish high radon areas **full protection** is required. In this case a fully sealed membrane and a radon sump should be provided. In other than high radon areas buildings should be provided with potential means of extracting radon for example a radon sump or sumps with connecting network.

In the radon affected areas of UK sufficient protection will be provided by a well installed damp-proof membrane. This is called the **basic radon protection**. In areas of higher potential **full protection** is required. This means supplementary provisions for radon sump or sub-floor ventilation.

Prevention strategies

Radon prevention practices include both passive and active methods. The passive systems include providing barriers to soil gas entry. Membranes and sealing of openings can be used. Active systems are based on sub-floor depressurization and/or ventilation. Passive systems are to be preferred. However the systems should include provisions for active systems.

Sub-slab depressurization provisions in new construction may be similar with those used in remediation. However in new construction well designed and efficient piping arrangements can be used.

A basic UK approach for a radon-proof membrane and sub-floor depressurization is presented in Figure 14. Figure 15 shows an active soil depressurization approach in USA (USEPA 1993).

In the Finnish guidance (Building Information Ltd 2003) membranes covering the whole floor area are not recommended. Instead the joint of foundation wall and floor slab should be sealed using a strip of reinforced bitumen felt, figure 16. A preparatory radon piping should always be installed, figure17.

The Austrian guide for citizens gives the following general instructions (Lebensministerium, Austria 2005).

- Do not build earth floors.
- Avoid cracks in floor slab.
- Seal the joint of foundation, basement wall in contact with soil and floor slab.
- Seal pipe penetrations.
- Seal pipes if these can serve as routes for soil air.

Radon prevention in new construction, Finnish survey 2009

The building code for radon prevention and the associated practical guidelines were revised in Finland in 2003 to 2004. Thereafter, preventive measures have become more common and prevention practices more effective. Consequently, indoor radon concentrations in new construction have been markedly reduced. In the 2009 study, the indoor radon concentration was measured in 1 500 new low-rise residential houses (Arvela et al. 2010). The houses were randomly selected and represented 7% of houses that received building permission in 2006.

The average radon concentration of all houses measured, which were completed in 2006 to 2008, was 95 Bq/m^3, the median being 58 Bq/m^3. The average was 30% lower than in houses completed in 2000 to 2005. The decrease was 50% in provinces with the highest indoor radon concentration and 20% elsewhere in the country. In houses with a slab-on-ground foundation that had both passive radon piping (discharge pipe uncapped) and sealing measures carried out using a strip of bitumen felt in the joint between the foundation wall and floor slab (Fig. 16), the radon concentration was on average reduced by 55% compared to houses with no preventive measures.

Preventive measures were taken in 50% of single family houses, and in provinces with the highest radon concentration in 90% of houses. Active prevention in areas with high indoor radon concentrations has reduced the regional differences in the radon concentration. Slab on ground is the prevalent type of foundation and necessitates

careful radon prevention measures throughout the country. The most serious defects were observed in prevention practices in houses with walls made of lightweight concrete blocks that were in contact with soil. The foundation types with the lowest radon concentrations were those with a crawl space and a reinforced uniform floor slab.

Effectiveness of protective measures

Research results of the effectiveness of protective measures are much more uncommon than results of remedial measures. In order to estimate the effectiveness of protective measures one needs also relevant reference values for houses without preventive measures. For example previous local values can be utilized. In the case of active methods as activation of sub-slab-depressurization, radon concentration measured for the passive system can be used as reference values.

In UK the use of membranes resulted in a reduction of approx. 50 % for both block and beam (under floor ventilation) and in-situ concrete floor (Figure 14) types (Woolliscroft 1994). The results are based on field studies involving over 400 dwellings.

Studies in 44 US homes showed a reduction of 50 % for passive sub-slab depressurization (Dewey and Novak 1994). Two-week tests were conducted in each home with the discharge pipe capped and uncapped.

Results from Czech republic show a reduction of 40% - 80% for membrane and sub-slab depressurization (Jiranek 2003).

Finnish new construction study 2009 in 160 houses (above) gives a reduction of 55% for houses with passive radon piping and sealing using bitumen felt (Arvela et al. 2010).

Activation of radon piping in SSD houses in Finland has resulted in typical reductions of 70 - 90% (STUK 2008).

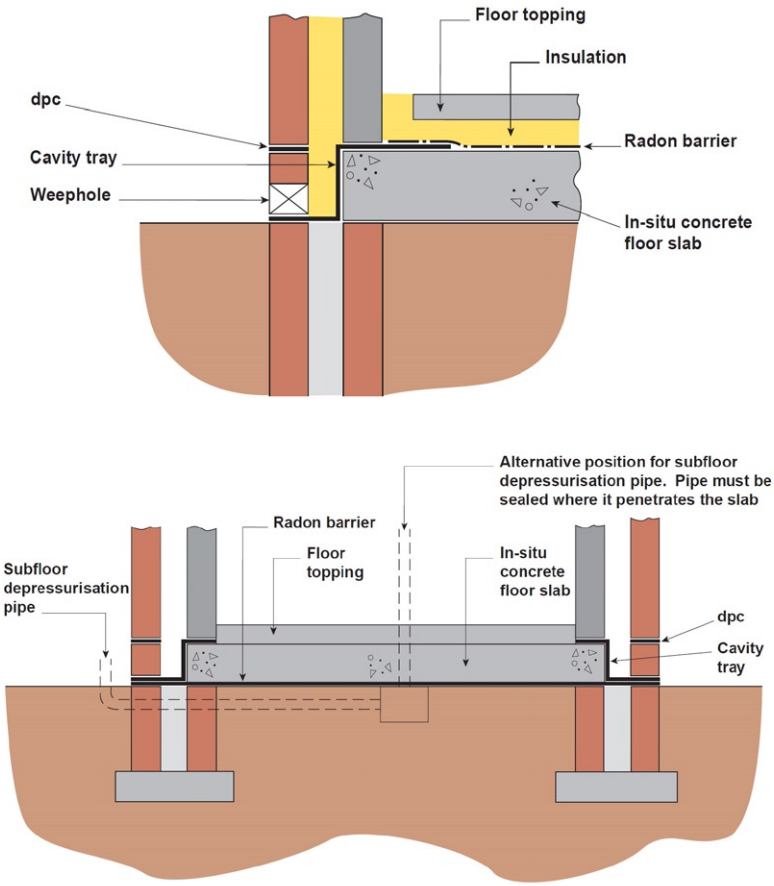

Fig. 14. BRE guidance for radon barrier and sub-floor depressurization in the case of a in-situ concrete ground floor (BRE 2007).

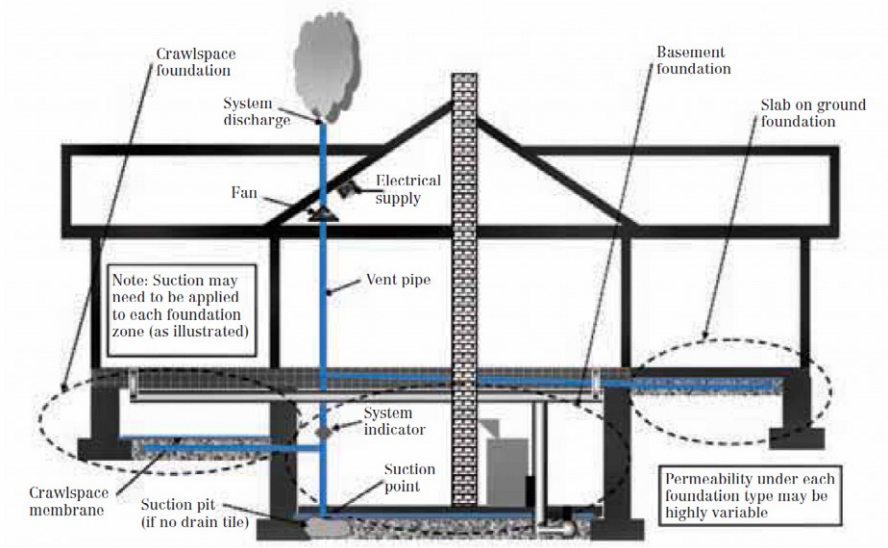

Fig. 15. Active soil depressurization in USA (USEPA 1993).

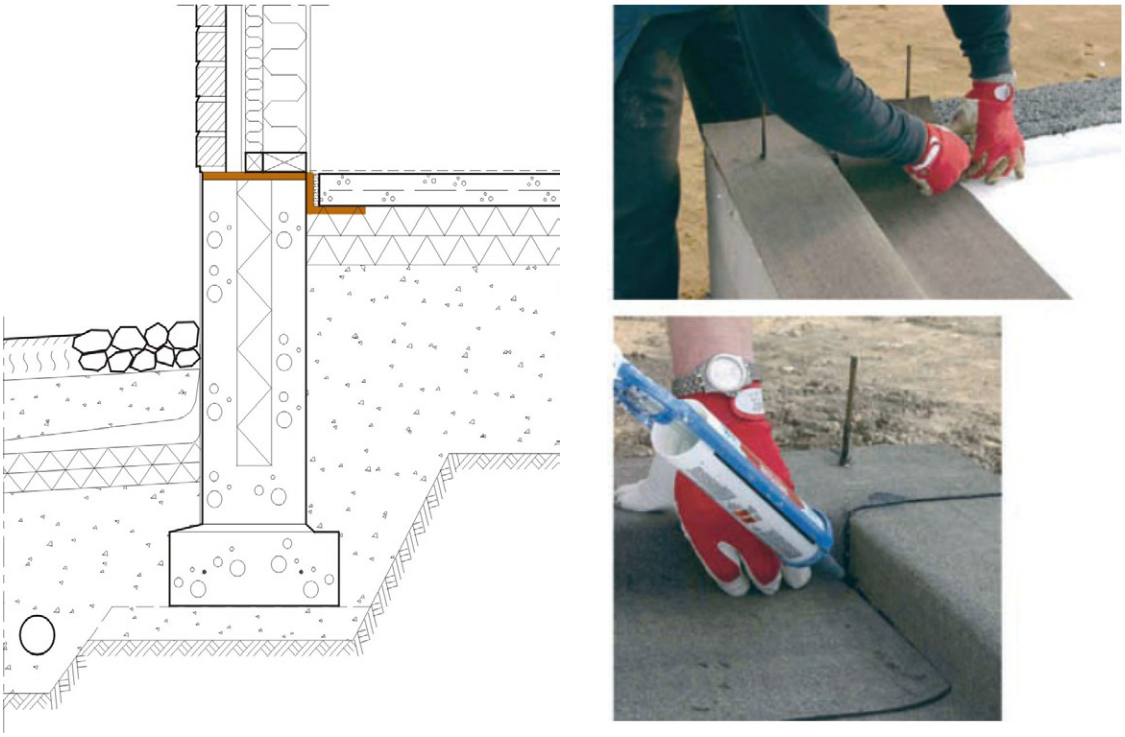

Fig. 16. Sealing of the gap between foundation wall and floor slab in the Finnish guide (Building Information Ltd 2003, Katepal Inc).

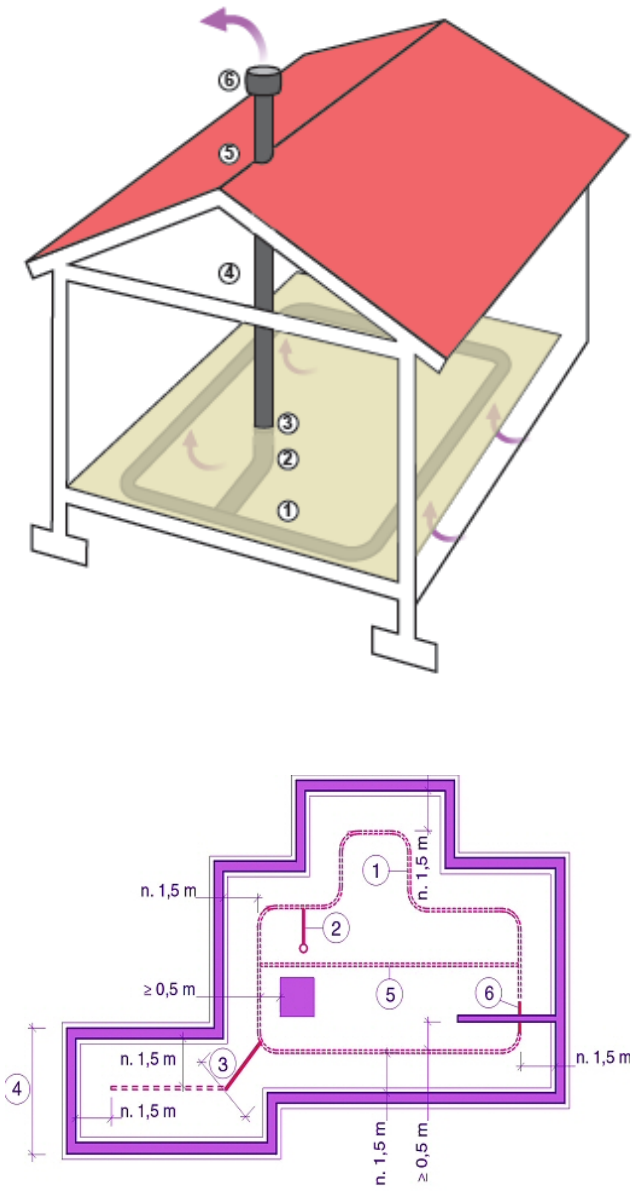

Fig. 17. Radon piping in the Finnish guide (STUK 2008, Building Information Ltd 2003).

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Questionnaire data of the EU RADPAR (Radon Prevention and Remediation) project (Directorate - General for Health and Consumers, DG SANCO) has been utilized in this presentation

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Radiation exposure of space and aircrew

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Abstract
Cosmic radiation and its secondaries created in interactions with planetary atmospheres, shielding structures and the human body constitute one of the most important hazards associated with space and air travel. Crew members are facing exposures to radiation levels that may easily exceed those routinely received by terrestrial radiation workers. To assess the significance of potential biological implications on the health of space and aircrew, it is necessary to discuss the characteristics of the cosmic-ray environment and its dependencies on altitude and geomagnetic latitude. Exposure of space and aircrew to cosmic radiation will be reviewed, and recommended dose limits for astronauts working in low-Earth orbit will be dealt with in comparison with radiation protection guidelines of aircrew personnel.

Introduction
Accomplishments in engineering over the past century have provided unprecedented opportunities for people to become mobile and travel rapidly on or near the surface of the Earth (White and Averner, 2001). Now that new technologies are at our hands to enable us to travel away from our home planet, we are about to become citizens of the universe. For this to happen requires the development of both a new understanding of the risks imposed by the potentially dangerous levels of cosmic radiation, extended weightlessness and psychological stressors, and a more effective means of coping with these hazards to the human organism. Ions of high charge and energy encountered in cosmic radiation have been shown to produce distinct biological damage compared with radiation on ground, leading to large uncertainties in the projection of cancer and other health risks, and obscuring evaluation of the effectiveness of possible countermeasures (Cucinotta and Durante, 2006). On a microscopic scale, it becomes apparent that these particles are likely to deposit their energy in a rather heterogeneous way. Although absorbed doses—averaged over a sufficiently large macroscopic mass element—might be small, there will be microscopic regions of extremely high local doses in close vicinity to the ion path.

Cosmic radiation environment
The radiation environment in space is characterized by a high degree of complexity and dynamics. It is mainly fed by solar and galactic sources (Fig. 1), with additional particles created in interactions with planetary atmospheres, shielding structures and the human body. Among these secondary radiations, neutrons are of foremost importance.

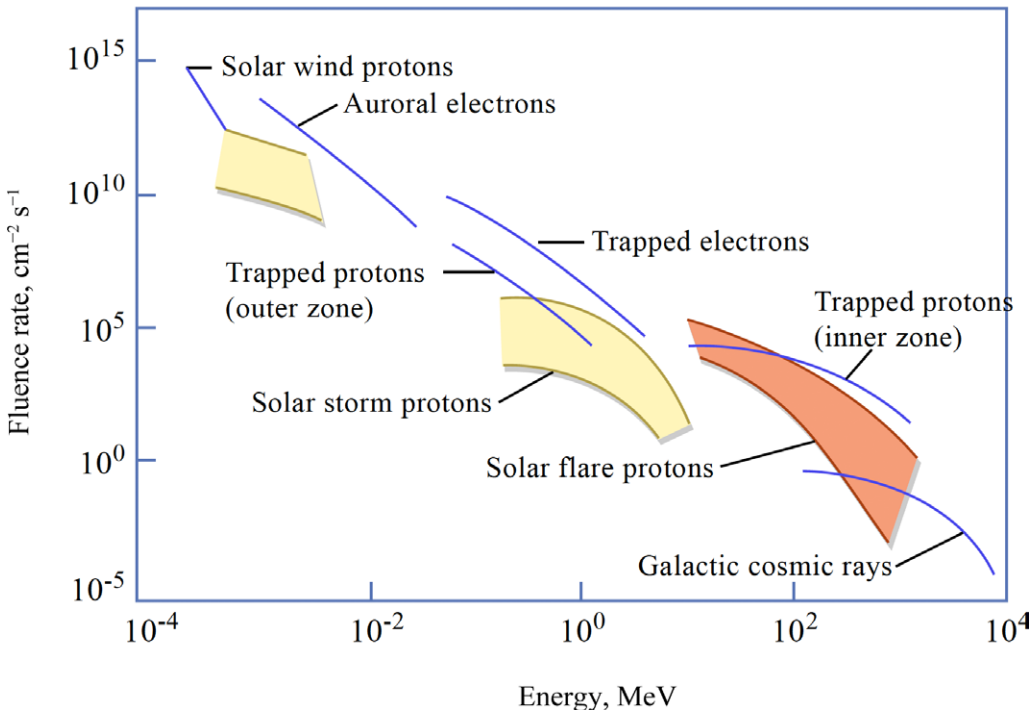

Fig. 1. Energy spectra of cosmic-ray contributors: energetic particles encountered in free space and low-Earth orbit cover a very broad range of energy and fluence rate (MIT OpenCourseWare).

Galactic cosmic radiation (GCR) originates from outside our solar system and is isotropic in distribution, *i.e.*, it arrives at any point in deep space with equal intensity from all directions. The GCR spectrum consists of all naturally occurring chemical elements with energies beyond 10^{20} eV (Lodders, 2003; Mewalt, 1988). Stellar flares, supernova explosions, pulsar spin-offs, or explosions of nascent galactic nuclei were believed to be the sources of GCR acceleration. However, there seems to be no credible mechanism, either inside or outside the galaxy, for accelerating particles to energies above 10^{20} eV (Schwarzschild, 1997). Astrophysicists developed plausible models for how ultra-high-energy cosmic rays might be produced, perhaps even involving new particle-physics phenomena or topological space-time defects left over from the Big Bang, but they still have no definite answers (Cronin *et al.*, 1997). The fluence rate of primary cosmic rays in the galaxy is ~1 cm^{-2} s^{-1}. Low-energy GCR particles consist of 92% protons and 6% helium nuclei, with the remainder being heavier ions with charges of $Z \leq 92$ (^{238}U). The incident fluence rate of cosmic rays with energies above 1 GeV is of the order of 10^{-2} cm^{-2} s^{-1} at the edge of the exosphere.

Solar cosmic radiation (SCR) comprises the flood of low-energy electrons and protons called the solar wind, which increases by factors of the order of 10^{6} during an active sun period to build into a torrential storm. This plasma, streaming out from the Sun's corona at velocities as high as 120 km s^{-1}, creates the interplanetary magnetic field (IMF), which varies according to the 11-year cycle of solar activity. GCR particles entering the heliosphere are scattered by IMF irregularities and undergo convection and adiabatic deceleration in the expanding solar wind (Heber *et al.*, 2009). The GCR intensity is thus anti-correlated with solar activity, which is usually determined from the number of sunspots (Lantos, 1993). Sporadically occurring solar particle events (SPEs)

originate from impulsive solar flares, coronal mass ejections or shocks in the interplanetary medium. The emitted protons have energies up to several hundred MeV and, during strong flares, their flux at the Earth's orbit can increase for some hundred percent during hours or days. It is these events, which are of particular concern for the possible manifestation of acute radiation syndrome effects such as nausea, emesis, haemorrhaging or, possibly, even death, since SPEs are still impossible to forecast and might be accompanied by significant dose enhancement (Townsend, 2005).

Energetic particles trapped in the geomagnetic field are confined via magnetic mirroring in two radiation belts, which surround the Earth. The inner belt, which extends from ~1–3 Earth radii in the equatorial plane, was discovered by J. A. Van Allen and co-workers[1] using data taken from Geiger-Müller counters flown on early U.S. satellites. It is mostly populated by protons with energies exceeding 10 MeV. The origin of these protons is thought to be the decay of albedo neutrons from the Earth's atmosphere. The inner belt is fairly quiescent. Particles eventually escape due to collisions with neutral atoms in the upper atmosphere above the Earth's poles. However, such collisions are sufficiently uncommon that the lifetime of particles in the belt range from a few hours to 10 years. Clearly, with such long trapping times only a small input rate of energetic particles is required to produce a region of intense radiation. The outer belt, which extends from ~3–9 Earth radii in the equatorial plane, consists mostly of electrons with energies below 10 MeV. The origin of these electrons is via injection from the outer magnetosphere. Unlike the inner belt, the outer belt is very dynamic, changing on time scales of a few hours in response to perturbations emanating from the outer magnetosphere. In regions not too far distant (*i.e.*, less than 10 Earth radii) from the Earth, the geomagnetic field can be approximated as a dipole field, which is tilted with respect to the Earth's rotational axis by an angle of ~11°. The intersection between the magnetic and rotational axis is located ~500 km more to the North, above the centre of the Earth. Because of this tilt and translation, the radiation belts are closest to the Earth's surface over the South Atlantic Ocean. This region is called the South Atlantic Anomaly (SAA) and is of great significance to space vehicles that orbit the Earth at several hundred kilometres altitude. These orbits take them through the anomaly periodically, each time exposing them for several minutes to increased radiation levels. The high SAA proton fluxes were explained to give rise to light flash phenomena in the eyes of astronauts (Casolino *et al.*, 2003).

Galactic cosmic rays and energetic particles generated in large solar flares finally interact with the Earth's atmosphere to produce in cascade-like reactions hadron, lepton and photon fields at aircraft altitude. The energy spectra of these secondary particles extend from the lowest possible energy to more than 10^{18} eV (O'Brien *et al.*, 1996). The total flux of ionizing particles in the upper atmosphere is fairly constant from 150–50 km altitude. Below 50 km the flux increases due to build-up of cascades and reaches the so-called Pfotzer maximum at about 15–20 km above sea level where absorption starts to dominate. The geomagnetic field deflects the incoming cosmic rays, depending on their rigidity, *i.e.*, momentum per unit charge, and angle of incidence. The vertical critical rigidity is zero at the magentic poles and at its maximum near the magnetic

[1] Van Allen was actually trying to measure the GCR flux in deep space, to see if it was similar to that measured on Earth. However, the flux of energetic particles detected by his instruments so greatly exceeded the expected value that it prompted one of his co-workers to exclaim, "My God, space is radioactive!"

equator. As a consequence, the primary (and secondary) cosmic-ray flux shows a distinct latitude effect. With respect to dose equivalent, atmospheric neutrons are the most important particles at aircraft altitude. They are produced as evaporation products of highly excited nuclei to form a peak around 1 MeV, and in peripheral collisions or charge exchange reactions of high-energy protons with a maximum flux around 100 MeV (Hajek *et al.*, 2004a). The dependence of neutron production on solar activity is most pronounced in polar regions, while the variation around the equator is just about 5%, since low-energy primaries are shielded by the Earth's magnetic field and high-energy particles undergo only slight solar modulation. The higher energy of primary cosmic rays entering the atmosphere around the equator causes the created neutrons to be able to penetrate deeper into the atmosphere, compared with pole-near latitudes. The maximum of the neutron flux is thus found at about 120 g cm^{-2} at the equator and at about 75 g cm^{-2} in polar regions. Considerable fluxes of neutrons are also produced when a strong solar flare hits the Earth.

Space crew exposure and radiation protection

Space travellers are facing exposures to radiation levels that may easily exceed those routinely received by terrestrial radiation workers. Missions in low-Earth orbit (LEO) are not exposed to the full intensities of the GCR and SPE spectra because of the protection afforded by the Earth's atmosphere and magnetic field. Hence, particle fluence rates are much lower than will be encountered in interplanetary missions—about a factor of three from the International Space Station (ISS) to deep space, where no protection from the magnetosphere or planetary bulk exists. The degree of protection is a function of spacecraft orbital inclination and altitude. For the 51.6° orbit of the ISS, typical dose equivalent rates are between 0.5 and 1.2 mSv d^{-1} (Berger, 2008; Hajek *et al.*, 2008), with ~75% coming from GCR and 25% coming from protons encountered in passages through the SAA region of the radiation belts (NCRP, 2006). For high-inclination space missions in LEO, only 25–30% of SPE protons are intercepted due to geomagnetic shielding, while the contribution of SPEs to the radiation load of astronauts is mostly negligible for low-inclination orbits (Benton and Benton, 2001). Outside the protection offered by the geomagnetic field, doses received from a major SPE in less shielded modules might reach lethal levels within a couple of hours. Hence, radiation shelters have to be provided to minimize the health risks for astronauts.

Radiation transport codes, which model the atomic and nuclear interactions of the cosmic-ray particles, are usually applied to describing how the external radiation fields are altered by passage through the spacecraft structure (Sihver, 2008). However, the high degree of complexity of both the shielding distribution and the generation of secondary charged and uncharged radiation makes it virtually impossible to simulate in detail the variation of the resulting particle fluence and energy spectra of the radiation field constituents within a space vessel. Unlike the situation for terrestrial exposures, the high costs of launching materials into space place limitations on spacecraft size and mass and preclude the purely engineering solution of providing as much additional shielding as needed to reduce radiation exposure to some desired level. Some model predictions indicate that some types of shielding materials may even give rise to secondary radiation environments that are more damaging than the unattenuated primary fields, which produced them.

Table 1. 10-year career limits for stochastic radiation effects applicable to missions in low-Earth orbit. Limits are expressed in effective dose (*E*). Recommendations by NASA and JAXA are age and gender specific (male/female).

NASA		Roscosmos	JAXA		CSA
Age, yrs	*E*, Sv	*E*, Sv	Age, yrs	*E*, Sv	*E*, Sv
25	0.7 / 0.4	1.0	25–29	0.6 / 0.6	1.0
35	1.0 / 0.6		30–35	0.9 / 0.8	
45	1.5 / 0.9		36–39	1.0 / 0.9	
55	2.9 / 1.6		≥ 40	1.2 / 1.1	

The development of radiation protection recommendations for astronauts reflects the current knowledge about radiation risks, which is based to a large extent on cancer incidence and cancer mortality in the atomic-bomb survivors of Hiroshima and Nagasaki. Since until now only few experimentally verified data on the biological effectiveness of heavy ions and the dose distribution within the human body exist, the concepts of terrestrial radiation protection are of limited applicability to human spaceflight, except for the principles of justification and optimization (ALARA). Radiation protection limits for astronauts are designed to prevent deterministic or non-cancer hazards and reduce the risk of stochastic effects to an acceptable level. Instead of applying the annual dose limits for workers on ground also to astronauts, whose careers are of comparatively short duration, the overall lifetime risk is used as a measure. The selection of dose limits for stochastic effects are related to the risk for fatal cancers (solid tumours and leukaemia) as well as for genetic effects. While radiation protection in the pre-Apollo era was concerned primarily with the avoidance of exposures, which might deteriorate the operational performance of astronauts, the first genuine radiation protection guidelines of the U.S. Space Science Board (NAS/NRC, 1970) allowed doubling of the spontaneous incidence of malignant tumours[2]. In 1989, the National Council on Radiation Protection and Measurements (NCRP) proposed age and gender specific dose limits for a 10-year career on the basis that a lifetime excess risk of cancer mortality of 3% was acceptable (NCRP, 1989). This risk was comparable with the risk in less safe but ordinary occupations, such as agriculture and construction. However, it is lower than the 5% lifetime risk that a radiation worker on ground would incur if the present annual protection limits were exhausted (20 mSv per year over 50 years). The increase of risk factors for fatal cancers per unit dose by UNSCEAR and BEIR V required reappraisal of the effective dose limits (Table 1), which were published in NCRP Report 132 (NCRP, 2000).

The Russian Federal Space Agency Roscosmos allows an annual limit of 500 mSv, and—in agreement with the Canadian Space Agency (CSA)—a career limit of 1 Sv, both independent of age and gender, since Russian studies yielded an increasing probability of non-cancer radiation effects with age that compensates the decreasing cancer risk (Roscosmos, 2004). The Russian career limit corresponds to an excess risk between 4.6% (at 30 years of age) and 2.4% (at 50 years of age). Like their U.S. analogue, the dose limits defined by the Japan Aerospace Exploration Agency

[2] In industrialized countries, the spontaneous cancer incidence is on average 20–25%.

(JAXA) depend on age and gender, but differ in the tolerated dose values and the age structure (Table 1). The associated excess risk is ~3%, but never exceeds 5%. The European Space Agency (ESA) based its radiation protection concept on the recommendations of the International Commission on Radiological Protection (ICRP, 1991) and the European Council Directive 96/29/Euratom (European Commission, 1996), both of which do not explicitly classify astronauts as radiation workers[3]. The limits applied to European astronauts are thus based on thresholds for deterministic radiation effects in dedicated organs and tissues (Straube *et al.*, 2010).

Dose limits for acute deterministic effects in the bone marrow, lens of the eye and the skin (Table 2) are expressed in gray equivalents (Gy-Eq), in which the organ absorbed dose is weighted by multiplication with the appropriate relative biological effectiveness (RBE) for a specific radiation quality and endpoint. The concept of Gy-Eq became necessary, since the radiation weighting factors used for stochastic effects do not apply to deterministic detriments. Considering the significant uncertainties in assessing RBE at low dose and dose rate, the values on which the dose limit recommendations are based have been determined at the threshold doses for the regarded deterministic effect (ICRP, 1989; Urano *et al.*, 1984).

For long-term missions outside Earth's magnetic field, the acceptable level of risk has not yet been defined, since there is not enough information available to estimate the risk of effects to the central nervous system and of potential non-cancer radiation health hazards (cataracts, cardiovascular diseases, etc.). Available data and pending questions have been compiled in NCRP Report 153 (NCRP, 2006), which will form the basis for developing radiation protection guidelines for missions into deep space.

Table 2. Recommended organ dose limits in Gy-Eq applicable to missions in low-Earth orbit. All limits are independent of age and gender.

Organ		NASA	Roscosmos	JAXA	ESA	CSA
Bone marrow, Gy-Eq	Acute	0.25	0.15	–	–	–
	30 d	0.25	0.25	–	0.25	–
	1 yr	0.5	0.5	0.5	0.5	–
	Career	–	–	–	–	–
Eye, Gy-Eq	Acute	–	–	0.5	–	–
	30 d	1.0	0.5	–	0.5	–
	1 yr	2.0	1.0	1.0	1.0	–
	Career	4.0	2.0	5.0	–	4.0
Skin, Gy-Eq	Acute	–	–	2.0	–	–
	30 d	1.5	1.5	–	1.5	–
	1 yr	3.0	3.0	4.0	3.0	–
	Career	6.0	6.0	20.0	–	6.0

[3] A task group appointed by the ICRP in 2006 shall develop recommendations for human space missions in low-Earth orbit and beyond.

Aircrew exposure and radiation protection

Aircraft passengers and crew are subject to elevated levels of secondary cosmic radiation produced in the atmosphere, the aircraft structure and its contents. From the beginning of the first commercial supersonic Concorde operations, measurements on board passenger aircraft became attractive and contributed to the vast pool of data available today. Total exposure on a given flight depends on the particular air route in terms of altitude (pressure rather than radar altitude) and geomagnetic latitude, as well as on solar activity and the duration of the flight. The dose rate increases with altitude and geomagnetic latitude, reaching a maximum at 15–20 km and a constant level above ~55°, respectively. As a rule of thumb, the effective dose from neutrons in polar regions is enhanced by a factor of ~6 compared with the equator, while the dose delivered by the directly ionizing component increases only by a factor of ~2. Commercial subsonic aircraft generally have cruising altitudes of 7–12 km. The effective dose rate at an altitude of 8 km in temperate latitudes is typically up to ~3 μSv h^{-1}, but decreases to only ~1–1.5 μSv h^{-1} near the equator (Fig. 2). At 12 km, the values are greater by about a factor of two. The dose for a return trans-Atlantic flight is typically 60–70 μSv (Hajek *et al.*, 2004b). The annual hours flown by crew members varies from individual to individual and from airline to airline, depending on policy. The average appears to be 300–900 hours per year. The annual effective doses received by aircraft crew usually lie within 2–4 mSv, with only few crew members receiving higher doses. At aircraft altitude and temperate latitudes, representative values of the main components of effective dose are neutrons 55%, electrons and positrons 20%, protons 15%, photons 5% and muons 5% (Bartlett, 2004). At sea level, the dominant component of effective dose is the muon component.

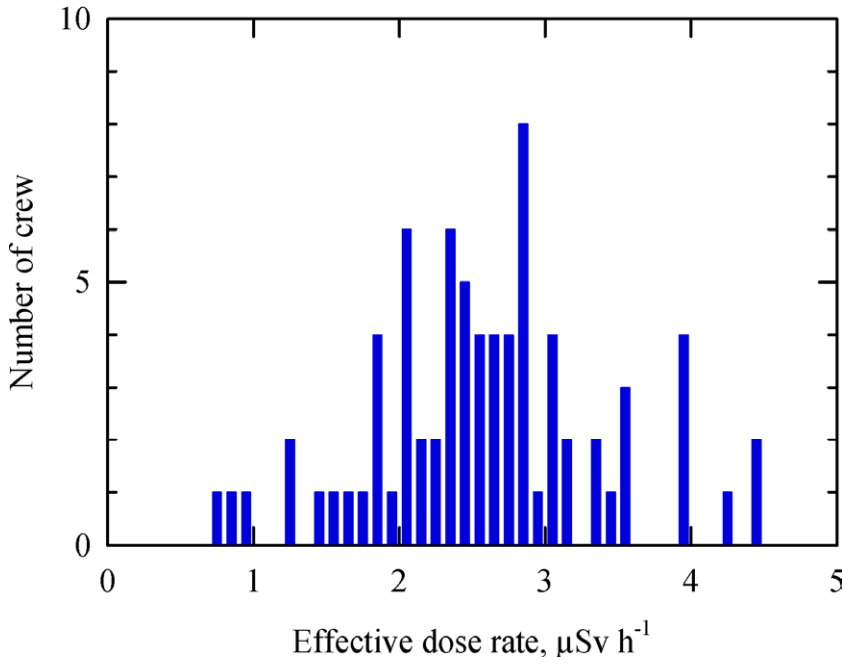

Fig. 2. Histogram of Tyrolean Airways crew radiation exposure on short- and medium-haul flights. The distribution of measured effective dose rates peaks between 2 and 3.5 μSv h^{-1}. For an average of 750 flight hours per year, effective dose can be estimated to result in 1.5–2.6 mSv.

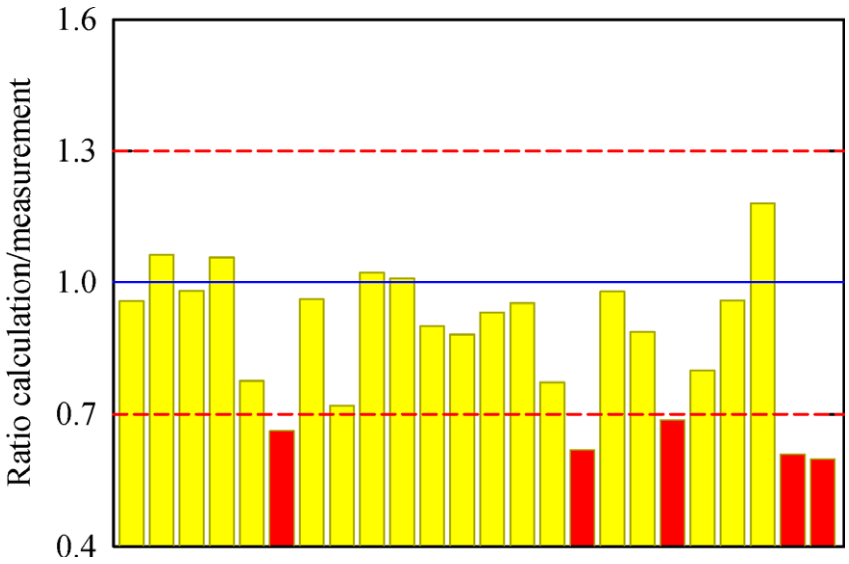

Fig. 3. Comparison of computed and measured route doses for 24 Tyrolean Airways flights. Agreement within ±30% is indicated, as required by Austrian regulations. There is generally good agreement between the results of calculations using CARI-6M and experimental determinations, except for very short-haul flights operated by propellant aircraft at low altitude.

There are a number of radiation transport codes and programmes to calculate dose rates and route doses in current use. The radiation transport codes take as input the cosmic radiation field at the top of the atmosphere and solve, either analytically or by Monte Carlo simulation, the radiation transport equations, which describe the interactions of each particle with the constituents of the atmosphere, in order to calculate the field at a given aircraft altitude and geographic location. The effect on particle trajectories of the Earth's magnetic field is included in approximations using tables of rigidity cut-offs. The programmes take account of the effects of IMF variation by applying an equivalent heliocentric electrostatic field. Generally, there is good agreement (Fig. 3) between the results of calculations and experimental determinations (Lindborg *et al.*, 2004).

Following ICRP recommendations (ICRP, 1991), the European Union (EU) introduced a revised Basic Safety Standards Directive (European Commission, 1996), which, *inter alia*, included the exposure to enhanced levels of cosmic radiation. The Directive requires account to be taken of the exposure of aircrew personnel liable to receive effective doses of more than 1 mSv per year. It further identifies the following protection measures (Bartlett, 2004): (i) to assess the exposure of the crew concerned; (ii) to take into account the assessed exposure when organizing working schedules with a view to reducing the doses of highly exposed crew; (iii) to inform the workers concerned of the health risks their work involves; and (iv) to apply the same special protection during pregnancy to female crew irrespective of the 'child to be born' as to other female workers. The EU Directive has already been incorporated into laws and regulations of the majority of the EU Member States and has been included in the aviation safety standards and procedures of the Joint Aviation Authorities (JAA). The preferred approach, supported by guidance from the European Commission and ICRP Publication 75 (ICRP, 1997), is that where the assessment of the exposure of aircraft

crew to cosmic radiation is necessary, doses can be computed from staff roster information, flight profiles and calculations of cosmic radiation dose rates as a function of altitude, geomagnetic latitude and solar modulation. The calculations are to be verified by measurements.

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Stakeholder involvement and engagement

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Abstract
One of the greatest challenges facing radiation protection professionals today is, how to include the society in radiation protection decision making. In response to this issue, internationally accepted guiding principles for radiation protection professionals on stakeholder engagement have been published by the International Radiological Protection Association (IRPA). IRPA goal is to ensure that consensus on directions for improvement of stakeholder involvement programmes is reached among radiation protection professionals, and that these guiding principles are taken into account during the development of future stakeholder involvement programmes.

Introduction
This work has been supported by Finnish Radiation and Nuclear Safety Authority, and parts of it have been published previously by International Radiation Protection Association and Nuclear Energy Agency within the Organisation for Economic Co-Operation and Development.

Stakeholder involvement and engagement are often used interchangeably when referring to public participation. In this article stakeholder involvement will refer to the policy and action taken with the stakeholders, whereas stakeholder engagement will refer to the process of how to engage with the stakeholder.

What is Stakeholder Involvement? Stakeholder involvement comprises of effective interaction between private and public companies, regulators and communities to make sure future activities and plans are discussed and understood by all parties. In decision making the stakeholder involvement is not intended to be stand alone process. Rather, it is designed to be an integral part of the decision-making and management processes, and it only has meaning if all parties have this intent from the outset. Whatever the level of involvement is, the distinction has to be maintained between the extent to which participants are responsible for the decision making and the level of interaction with stakeholders. Once this has been achieved the open and transparent approach of stakeholder engagement can help in establishing and maintaining effective relationships between all stakeholders (IRPA 2007).

History of Stakeholder involvement in Radiation Protection

Stakeholder involvement is not a new idea, but we are learning more and more of its application and benefits. Until the 1950s, the so called "managerial model" defined the relationships between the authorities and the public (Beierle and Crayford, 2002). In the "managerial model" the government administrators and experts in government agencies, are committed to pursuing the common good. In addition, they have been deliberate choices as possible so that the public interest and their policy have been used to produce the greatest benefit to most people in the long run. The signing of the Administrative Procedure Act in the United States in 1946, was a definite sign that the "managerial model" was reaching the end of the road. Soon other developed economies followed this trend with similar legislations. As the economies developed further and became more and more industrialised, the governments were faced with difficult decision making on matters related to environmental and health issues in particular (Beierle and Crayford, 2002, OECD/NEA, 2006).

In the 1970s, pluralism began to replace the "managerial model" as the primary approach to administrative decision making (Reich, 1985). According to this pluralist view, government regulators and politicians were not asked to have the impossible role of objective guardians and decision makers in the public interest, but rather be arbiters among different possible interests within the society. It was recognised that everlasting "common good" has no objective meaning, but rather that public interest has to be debated and arrived at by negotiation among different interested parties (Williams, 1995). This attitude was reflected in the 70s and 80s in many decisions where the role of public industrial (including nuclear) agencies was revised, with a clear separation of agencies having regulatory responsibilities and agencies aiming at promoting industrial development and applications (OECD/NEA, 2006). This separation of regulator and operator functions has since that time been a necessary element of any developed regulatory system (IAEA, 1996).

In the 1990s the "pluralist model" came under pressure as even more intense participatory perspective was expected by the stakeholders; with the aim to have consensus outcome rather than unsatisfying compromise, and from a desire to have long term sustainable decisions rather than immediate decisions from a better funded opposition (Beierle and Crayford, 2002, OECD/NEA, 2006). This "democratic model" stresses the importance of the act of public participation, not only in influencing decisions but also in strengthening public capabilities and social capital (Beierle and Crayford, 2002), emphasises interaction among often adversarial interests, but that interaction is viewed less as competitive negotiation than as a way to identify the common good and subsequently act on shared communal goals (Dryzek, 1997). From the public perspective, the act of participation *"makes people more aware of the linkages between public and private interests, helps them develop a sense of justice, and is a critical part of the process of developing a sense of community"* (Laird, 1993).

In today's regulatory environment stakeholder involvement provides the decision-framing, decision-aiding and decision making processes along the lines of the "popular-pragmatistic" model. The goal of radiological protection is, and has always been, to provide an appropriate standard of protection for man and environment. At the same time the justification principle requires, that consideration is given to the benefits arising practices, which give rise to radiation exposure, and that doses are optimised in

accordance to the optimisation principle (ICRP, 2007). These principles are the cornerstones of radiological protection, and they can not be sidelined in any radiation protection decision making. The implementation of these principles was first thought to be possible through semi-quantitative analytical methods, such as cost benefit analyses. While these approaches provide a great guideline it has to be complemented with a genuine public participation.

The result of the evolution of the radiological protection system, towards increasing public participation in environmental and health related decision-making, can be fully appreciated in the current ICRP Recommendations (ICRP, 2007), where a paragraph is devoted to the participation of stakeholders.

"This decision making process may often include the participation of relevant stakeholders rather than radiological specialists alone."

Stakeholders

To understand how to involve the stakeholder, it is first necessary to understand what a stakeholder is and what it will be defined as in this paper. In the simplest form "A stakeholder is a person or other actor with special concern and interest in an issue, and may be considered to be concerned either on the basis of self report or on the basis of observed activities", (Sjoberg, 2003). Significantly, stakeholders bring differing perspectives to the process, given the range of backgrounds and disciplines that stakeholders are drawn from. For instance, some internal and external stakeholders may have a reasonable degree of commonality of interest or a legal obligation within the subject under discussion. These are sometimes referred to as 'true stakeholders'. But other stakeholders become involved because they are affected by the decisions taken, or have a strong view on the conduct of companies and government organisations, even if their interests are very different from those of the organisation in question .

For radiation protection issues the following stakeholders can be identified:

- Regulatory Stakeholders ('true stakeholders'). All practitioners, who require a 'licence to operate' from defined stakeholders, such as radiation protection regulators, health and safety regulators or environmental regulators. In some countries, there are also legal requirements to consult statutory stakeholders with a regulatory or quasi-regulatory function in many contexts (especially planning-related).
- Campaign groups often see themselves as having a 'license to operate' or watchdog role, but they are also significant as opinion formers, able to influence other stakeholders particularly through the media.
- The media are sometimes considered to be stakeholders, but are more often considered separately alongside other opinion formers. There is usually no strong commonality of interest between the media and industry. The media has, almost by definition, the potential to exert considerable influence on other stakeholders, including the public.
- The local community is the main target in many stakeholder programmes. However the community cannot be treated as a single entity, relationships between practitioners and the above are contained within it. The people who live around affected by the decisions and the community are complex and

all the different types of stakeholder described community groups and local authorities that speak for them, have a wide range of inter-relationships and perspectives. There is usually no such a thing as 'the community view'.

- Industry and commerce are often neglected in many stakeholder programmes. Similarly local communities, it is impossible to identify industry a joint commerce view, as there a number of interests and opinions represented in industry and commerce.

To engage with the stakeholders and fully realise the benefits of involving stakeholders in the decision-making processes International Radiological Protection Association (IRPA) have developed Guiding Principles for Radiation Protection Professionals on Stakeholder Engagement.

Guiding Principles for Radiation Protection Professionals on Stakeholder Engagement

The need for Guiding Principles on stakeholder engagement were first identified by IRPA in 2004, when it was agreed that stakeholder involvement, should play an important and integral part in decision-making processes related to radiological protection. The aim of the guidance was to help radiation protection professionals to understand the objectives, requirements and demands of stakeholder involvement, encourage participation and provide a framework for establishing a constructive dialogue with other stakeholders.

According to the IRPA guidance radiological protection professionals should endeavour to:

1. Identify opportunities for engagement and ensure the level of engagement is proportionate to the nature of the radiation protection issues and their context.
2. Initiate the process as early as possible, and develop a sustainable implementation plan.
3. Enable an open, inclusive and transparent stakeholder engagement process.
4. Seek out and involve relevant stakeholders and experts.
5. Ensure that the roles and responsibilities of all participants, and the rules for cooperation are clearly defined
6. Collectively develop objectives for the stakeholder engagement process, based on a shared understanding of issues and boundaries.
7. Develop a culture which values a shared language and understanding, and favours collective learning.
8. Respect and value the expression of different perspectives.
9. Ensure a regular feedback mechanism is in place to inform and improve current and future stakeholder engagement processes.
10. Apply the IRPA Code of Ethics in their actions within these processes to the best of their knowledge.

Case Studies

Radiation protection regulators around the world are in a situation where they are regulating activities at NPPs and other facilities with radiation risks, which were constructed in a previous era without true stakeholder involvement in the associated decision making. However, with the recent developments in the nuclear industry, it is anticipated, that a variety of stakeholders will seek participation in decision making as new plants are developed and old ones decommissioned.

IRPA, ICRP, IAEA and OECD/NEA have all concluded that the expectations of stakeholders of a right to participate in decision making, is something that the radiation protection authorities should address, and in some countries already do. Decisions regarding such matters as policy, regulating radiation practices and siting and construction of NPPs, should be voluntarily opened up to all stakeholders from the closed domain of technical experts and government officials (IAEA, 2006, IRPA, 2007). Below are a few case studies demonstrating the application of stakeholder involvement in radiation protection, full details of these case studies can be seen in Appendix 1, case studies are taken from a OECD/NEA Expert Group on Stakeholder Involvement and Organisational Structures.

Stakeholder Involvement in mobile phones and radiation (Sweden, SSI)

The Swedish Statens strålskyddsinstitut (SSI) has responsibility for ionising and non-ionising radiation. This case study concerned debate around new mobile phone masts. A decision was taken at the European level and licenses issued in Sweden by the post and telecommunications authority for construction of a network for a 'third generation' of mobile phones. However, concerns amongst sections of the population over the health impacts of radiation from the masts associated with the network led to opposition to construction of masts. SSI's risk assessment concluded that the risk from the masts was essentially zero. To understand the views of those concerned by the masts, to explain its own view and to allow dialogue with other stakeholders, SSI organised three seminars around the topic. Although the seminars did not result in consensus, stakeholders discussed with each other and were able to improve their own understanding of other stakeholder roles and views.

Stakeholder Involvement in NPP Siting (Finland, STUK)

The Säteilyturvakeskus (STUK) has a role as an expert body and regulator for the new nuclear power plant at Olkiluoto in Finland. STUK's role was essentially one of providing technical expertise in an open fashion, in a context where local and national elected representatives had to explicitly approve construction of the power plant. STUK provided information on the regulatory process and its work in it, responding to address topical, yet potentially sensitive issues such as aircraft impact. It was also willing to interact with questioning stakeholders. The approach used was open but, broadly speaking, more reactive than proactive and seems to have resulted from a general presumption of openness in STUK. STUK's approach appeared to be effective and 'fit-for-purpose'.

Stakeholder Involvement in post-Chernobyl contamination monitoring (Belarus, CORE)

Generally radiological protection is concerned with 'normality' in 'normal' situations i.e. tasks are entrusted to professionals. However, the situation is somewhat different in the contaminated territories where it becomes even more important to balance radiological protection with other aspects of life. Indeed, 'integrate' is probably a better word, since the question is one of co-working with civil society, rather than arriving at a better informed decision or improving credibility. Thus, the question moves from 'regulation' to 'governance'. A lesson to be learned from experiences in Belarus is that for the public in general to share in governance of a topic, it is necessary for people to fully appreciate all the issues and feel like they have created a real partnership in the decision processes affecting their safety and living conditions.

Stakeholder Involvement in UK Nuclear Decommissioning Authority (UK, NDA)

The Nuclear Decommissioning Authority (NDA) is a public body set up to run decommissioning of United Kingdom civil nuclear facilities. The actual decommissioning is carried out by contractors. Drawing on experience in the nuclear industry and on the findings of a government commissioned report, stakeholder involvement was recognised as an important factor and included in the NDA from its inception. By re-invigorating previous site local liaison groups (run by the operators), the NDA set out to involve stakeholders in its work in a local and national framework. Local liaison groups include representatives from civil society, local municipalities, regulators and contractors as well as the NDA. One of the aims of introducing the national tier was to link local concerns to national, strategic considerations. Through founding legislation and its own policies, the NDA retains a firm commitment to stakeholder involvement in its work.

Stakeholder Involvement in Policy Formulation (USA, USNRC)

The United States Nuclear Regulatory Commission (USNRC) is a federal regulator tasked with nuclear regulation. In developing Below Regulatory Concern (BRC) policy USNRC consulted their stakeholders to achieve a sustainable decision.

This process began in earnest in the early 1990's, but it had limited success. USNRCs most recent effort to consult stakeholders on clearance policy was termed NUREG-1640 rulemaking (NRC, 2002). This programme built on the early BRC stakeholder programme, which started with good intentions, but it polarised the views of the stakeholders further it went along. Consequently four of the eight environmental and consumer groups that had been actively involved from the beginning of the BRC policy formulation refused to enter a proposed consensus building process and the programme ended in failure. The new process centred around a series of public meetings, in which the stakeholders were asked to participate. For their detriment USNRC had lost the trust and confidence of some of the important stakeholders during the first BRC policy formulation and had not done enough in the meantime to engage the public in any effective manner to regain the trust of the stakeholders. This scepticism and lack of trust again led to some national environmental and consumer advocacy groups to boycott the public meetings. Due to the stakeholder feedback USNRC decided to postpone the rulemaking and carry on with case by case clearance.

Conclusions

Stakeholder involvement is a key concept in modern approaches to decision making and it has received considerable attention from IRPA, ICRP IAEA and OECD/NEA (IRPA, 2004, 2008, ICRP, 1991, 2007, IAEA 2006, OECD 2003, OECD/NEA, 2001a, 2001b, 2003, 2004). It is worthwhile to mention the role played by IRPA in identifying the need for the Guiding Principles for Radiation Protection Professionals on Stakeholder Engagement. These principles were produced to help radiation protection professionals to understand the objectives, requirements and demands of stakeholder engagement, encourage participation and provide a framework for establishing a constructive dialogue with other stakeholders.

Taking into account the views of different stakeholders is sometimes considered a time-consuming process leading to solutions technically not optimised. As the case studies have shown stakeholder involvement is often essential in obtaining societal acceptance of decisions. It should be noted, however, that in some cases stakeholders will not wish to accept decision, and that for them the perfect solution is the "do nothing" option sustaining the status quo. Under these circumstances, the regulators will have to decide whether it is best to go forward or not. However, sometimes as highlighted in the case studies this type of arbitrary decisions will result in stakeholders seeking to influence the decisions through courts or even through direct action.

Significant benefits can be gained from the involvement of stakeholders. For radiation protection the most quoted are:

- Responds to shifts in societal attitudes to science, industry and government.
- Offers possibility of resolving tensions between economic and social concerns.
- Helps to prevent disputes and conflicts where it is deployed ex-ante.
- Helps to resolve disputes and conflicts where it is deployed ex-post.
- Increases the substantive quality and sustainability of decisions.
- Builds trust in institutions.
- Educates and informs the public.

To realise these benefits it has to be appreciated different stakeholders have different perspectives, perceptions, beliefs, interests and values. Having clear aims and objectives from the outset will assist in planning a dialogue process without conflict. Also it should be made clear who is responsible for making the final decision. In order to arrive at this point the roles of the regulators and stakeholders, should be well defined. To measure success, evaluation criteria must be developed to evaluate the process with the stakeholders that will be participating.

Even in the most inclusive and transparent stakeholder involvement programmes, there can be conflict between the equal opportunity to participate and influence both programme and outcomes for anyone who affected by the decision, and competent participation, i.e. construction of the most valid, considering both societal and technical aspects. To ensure this seen as an opportunity and a mutual learning process the programme has to be initiated early on, and efforts must be made to build trust and consensus among the stakeholders.

Once the *"optimum solution under the circumstances"* the involvement of local stakeholders must continue with the aim, of demonstrating to all parties the compliance with the objectives of the programme.

As the case studies have shown decision making is not always successful, even with the help of stakeholder involvement, but more than often stakeholder relations and legitimacy of radiation protection professionals can be improved through stakeholder involvement. This is an arduous path to take, but if the field of radiation protection is to involve the stakeholders in future decision making it is the only way to proceed.

To make this process a little simpler, IRPA have published the Guiding Principles for Radiation Protection Professionals on Stakeholder Engagement. These guiding principles should provide a good basis for running any stakeholder involvement programmes in the field of radiation protection.

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Decommissioning and waste management

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Abstract

Decommissioning of nuclear installations gives rise to large quantities of different types of radioactive material. Depending on its type, the overall mass of the controlled area of a large nuclear power plant has a mass of 100,000 to 400,000 Mg. The most important material streams are activated metals from the reactor pressure vessel and its components, activated concrete from the biological shield and the surrounding building structures, contaminated materials from systems, components and other structural elements, contaminated concrete from building structures, electrical installations, insulation material etc.

Decommissioning projects are in progress in a large number of countries around the world. In many countries, decommissioning has developed to be a mature industry. Techniques for dismantling and decontamination have been developed and are now available for all tasks and materials. This is also true for management of radioactive waste originating from decommissioning. Solidification, incineration, melting of metals, encapsulation and supercompaction are examples for techniques for treatment and conditioning of radioactive waste. The final disposal of radioactive waste from decommissioning is treated differently in various countries. Some countries use near-surface repositories for such wastes, which mainly consists of low-level waste, while others rely on deep geological repositories.

Only a small portion of the overall material from decommissioning has to be disposed of as radioactive waste. The vast majority of material can be cleared (that means released from radiological control). A thorough radiological characterisation is the prerequisite for clearance, for which, however, material from decommissioning of nuclear power plants has suitable properties. The radionuclides involved in material from nuclear power plants generally consist of a large portion of gamma emitters, so that the concept of nuclide vectors can be used. The measurements can be based on key nuclides like Co-60 and Cs-137, and the activity of nuclides that are not easy to measure can be derived from these values. Clearance procedures and clearance levels have been introduced by a large number of countries, making clearance are very important option in material management. Various measurement techniques are available for performing clearance.

Introduction

This paper addresses two topics that are closely related:

- decommissioning of nuclear installations and
- management of materials originating from decommissioning.

Decommissioning is the final phase in the lifecycle of nuclear installations after siting, design, construction, commissioning and operation. It is a complex process involving operations such as detailed surveys, decontamination and dismantling of plant, equipment and facilities, demolition of buildings and structures, site remediation, and the management of resulting waste and other materials. Decommissioning gives rise to a large amount of various types of material – a few 100,000 Mg for large NPP blocks – that can be treated and disposed of as radioactive waste or can be decontaminated and cleared (released) as non-radioactive material. This is the reason why the decommissioning phase of nuclear installations poses much larger challenges for the waste management than the operating phase.

This paper tries to address all aspects of this wide range of topics at least briefly: It first describes the decommissioning as a global task and as a process for a particular facility (restricting itself to nuclear power plants as the type of nuclear facilities from which the largest amounts of decommissioning waste arises), then lists the most important material streams originating from decommissioning and the related material quantities, and finally covers both management of radioactive waste (i.e. of material that has to be disposed of because of its activity) and clearance of material (i.e. the release of material that is either not contaminated above clearance levels or that complies with these levels after decontamination). As the topic is very wide, the information given in each section has to be necessarily succinct.

When analysing the development of decommissioning projects over the last two decades, it becomes clear that the vast majority of issues concerning techniques for decommissioning and dismantling, decontamination, waste management including treatment and packaging, measurement and characterisation, as well as clearance and release have been solved. Standard procedures and techniques are available off the shelf and can be deployed in all types of nuclear installations. Decommissioning has become a mature field of work. Nevertheless, each decommissioning projects needs its individual planning and adaptation of the techniques to its distinctive features. Waste management options that are available in a specific country will also influence the choice of techniques and procedures, even the choice of the decommissioning strategy. Therefore, both topics are closely interrelated, and this paper tries to highlight these interrelations.

Generally, it is assumed that the fuel elements would have been removed before start of decommissioning. There are, however, a number of projects where decommissioning has started with fresh and spent fuel on the site. In order to reasonably limit the scope, removal and storage of spent fuel is not dealt with in this paper. Likewise, operational waste that is still present at the beginning of decommissioning can usually be dealt with in the same way as during operation and is therefore also not included in this paper.

Decommissioning

Decommissioning of nuclear installations is not just the opposite of its construction. The aspects of radiation protection, management of partly highly activated and contaminated material and of large quantities of material for which the absence of contamination has to be verified, the necessity of remote operation in some areas where space might be tight and other features make decommissioning of nuclear installations a technically challenging and interesting field. In this chapter, decommissioning strategies that are viable for nuclear power plants and other nuclear installations are presented, followed by an overview of decommissioning projects and is decommissioning techniques.

Decommissioning strategies

There are two basic strategies that are widely used for decommissioning: immediate or early dismantling and deferred dismantling. A third strategy, entombment, leaves the facility and structures in place. This option, however, is rarely used. (IAEA, 2007)

Early Dismantling: Immediate or early dismantling is the strategy in which the equipment, structures, components and parts of a facility containing radioactive material are removed or decontaminated to a level that permits the facility to be released for unrestricted use (clearance) as soon as possible after permanent shutdown. In some cases, where unrestricted release is not feasible, the facility may be released from regulatory control with restrictions imposed by the regulatory body. The implementation of the decommissioning strategy begins shortly after permanent termination of operational activities for which the facility was intended, normally within a few years. Early dismantling involves the prompt removal and processing of all radioactive material from the facility for either long term storage or disposal. Non-radioactive structures may remain on-site. This is the preferred decommissioning strategy.

Deferred Dismantling: Deferred dismantling is the strategy in which the final dismantling of the facility is delayed and the facility is placed into long term storage where it is maintained in a safe condition. This strategy may involve some initial decontamination or dismantling, but a major part of the facility will remain for a certain time period in a caretaker mode. This time period might range from a few years to several decades, after which time the decommissioning process will be completed and the facility can be released from regulatory control. The deferred dismantling option is often used at multi-facility sites when one or more of the facilities are shut down while others continue to operate. This is especially true of facilities that share some common systems.

Entombment: Entombment is the strategy in which the radioactive contaminants are encased in a structurally long lasting material until the radioactivity decays to a level that permits release of the facility from regulatory control. The fact that radioactive material will remain on the site means that the facility will eventually become designated as a near surface waste disposal site and criteria for such a facility will need to be met.

Figure 1 shows an example for the early dismantling option, i.e. the transformation of a nuclear power plant site from the operational status to (nearly) green field conditions. Some buildings may be left in place, in this case buildings housing interim storage facilities that have to remain on site until the waste may be removed to a repository. The rest of the buildings of the controlled area as well as other have been cleared and removed.

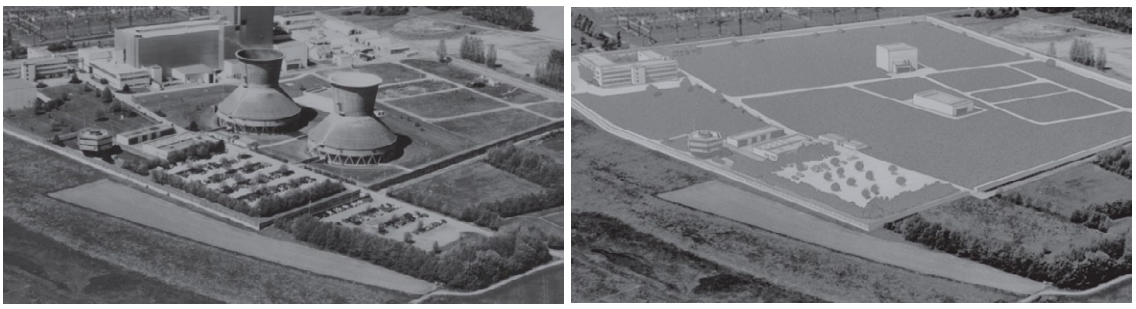

Figure 1. The site of NPP Würgassen (Germany). Left: before start of decommissioning (1995), right: after reaching green field conditions (artist's impression) (E.ON 2008).

Overview of Decommissioning projects

A large number of decommissioning projects exist worldwide. Table 1 gives an overview of shutdown power reactors in various countries that are under decommissioning (either early or deferred dismantling) or are awaiting decommissioning. There is an even larger number of research reactors that have also been permanently shut down or that are undergoing decommissioning. This overview makes it clear that decommissioning in many countries has become an industry handled by specialists rather than an experimental field that could be dealt with by the former operator of the facility alone.

Table 1. Shutdown reactors by country (IAEA 2010).

Country	Number of Units	Total MW(e)
Armenia	1	376
Belgium	1	10
Bulgaria	4	1632
Canada	3	478
France	12	3789
Germany	19	5879
Italy	4	1423
Japan	5	1618
Kazakhstan	1	52
Lithuania, Republic of	2	2370
Netherlands	1	55
Russian Federation	5	786
Slovak Republic	3	909
Spain	2	621
Sweden	3	1210
Ukraine	4	3515
United Kingdom	26	3301
United States of America	28	9764
Total	**124**	**37788**

There are a number of countries where a nuclear decommissioning industry has developed separately from the nuclear industry itself and where there is a good coexistence between operating nuclear installations and those undergoing decommissioning. As an example, Figure 2 shows an overview of NPPs in operation and undergoing decommissioning. The number of operating blocks and the number of blocks under decommissioning already equal each other.

Figure 2. Nuclear power plants in Germany in operation and undergoing decommissioning (Thierfeldt 2009).

Execution of decommissioning projects

Decommissioning projects are often implemented and executed using a phased approach. This means that the entire decommissioning project is structured into phases for which a license, permit or consent of the competent authority would be needed. The advantage of such an approach is that the next phase can be planned and the associated licensing procedure can be carried out while the previous phases are still in execution. Each phase can take advantage of experience gained in the previous phases. Figure 3 shows an example for decommissioning phases and their approximate duration (each phase divided into planning / licensing / implementation), while Figure 4 shows the corresponding illustration of the implementation of these phases for a boiling water reactor.

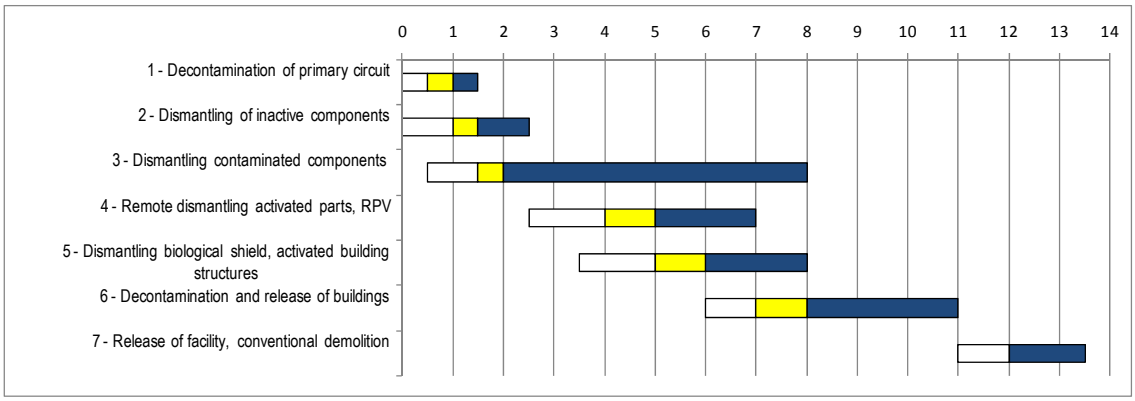

Figure 3. Example for decommissioning phases and their approximate duration for a large NPP.

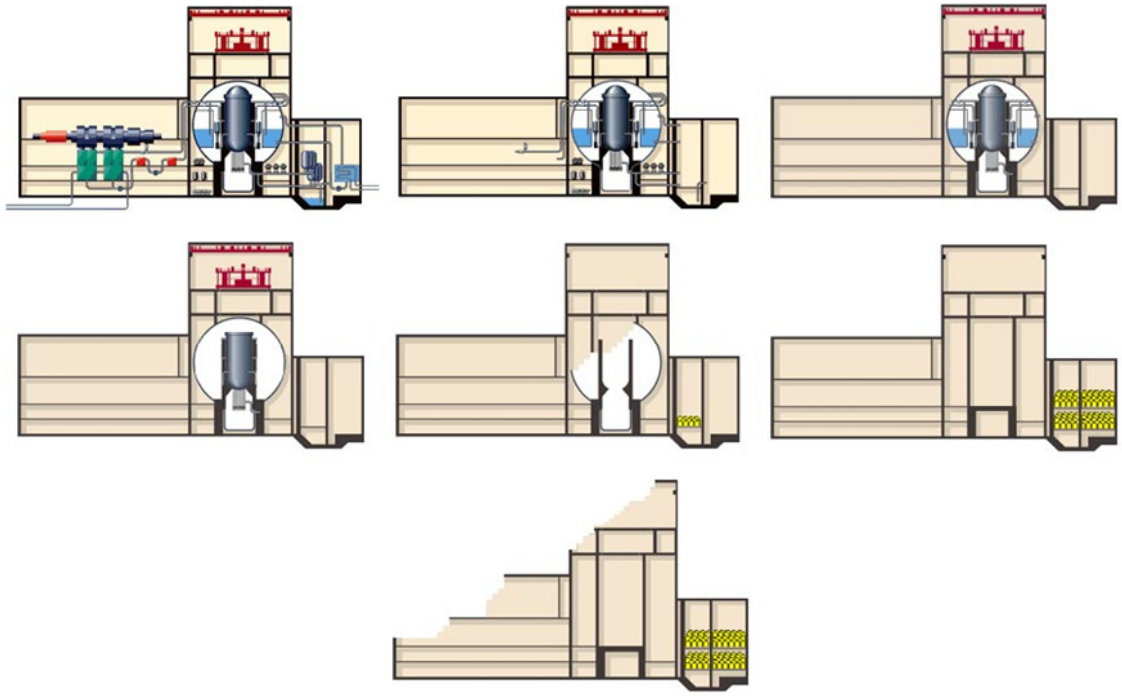

Figure 4. Illustration of decommissioning phases for a boiling water reactor (E.ON, 2008).

Decommissioning Techniques

For all the decommissioning strategies, the dismantling of external, unusable support systems and equipment will generally be performed using standard techniques. Much of this will be uncontaminated material that can be released from regulatory control and therefore needs to be segregated from contaminated material (see chapters on radioactive waste management and clearance).

There are differences in dismantling techniques between early and deferred dismantling. Higher dose rates resulting from high activation or contamination levels may require remote techniques for early dismantling of some facilities. High dose rates from activated components can generally be compensated by using additional shielding, e.g.

by working under water in the reactor spent fuel pool, and/or remotely operated techniques. For deferred dismantling, it is possible that some of the waste will have decayed to low levels and special techniques may not be needed during the dismantling phase. This will depend on the initial level of contamination of the specific structures, equipment or facilities to be decommissioned. It is desirable to plan for deferral periods if this will allow decay to low levels of contamination to simplify decommissioning. - Similar considerations apply to the entombment option: early preparation for entombment will require dealing with material with higher dose rates, while establishing entombment towards the end of a long deferral period will allow use of simpler techniques to prepare the facility as a near surface disposal site.

It can be reasonably assumed that most of the existing facility support equipment (e.g. lifting devices and bridges, utilities, water cleaning and filtration facilities, ventilation, sanitary facilities) will not be available after a prolonged deferred dismantling period without some type of refurbishment or replacement. This means that some advantages can be realised by selecting the immediate dismantling option in preference to deferred dismantling.

Decommissioning of facilities can be carried out with existing technology. There is no significant advantage in waiting for further progress in technology development, but in some particular cases (e.g. treatment/conditioning of special material such as graphite and the remote controlled tools needed for dismantling the reactor internals) some improvements may be expected and this could probably favour deferred dismantling in these particular cases. Nevertheless, there currently exists remote handling technology that allows the performance of this type of activity. (IAEA 2007).

Decommissioning techniques are generally divided into thermal techniques, mechanical techniques and other types. Dismantling and cutting techniques are available for metals, concrete, and all other purposes like cables, insulation etc. There is no single technique which could be used for all purposes. Thermal cutting techniques encompass various principles of heating and melting the material, like arc processes, in particular plasma arc processes, gas processes, laser beam processes, electro discharge machining and other processes, especially combinations of these techniques. An example is shown in Figure 5. Mechanical cutting techniques encompass grinding, various sawing techniques, shears, nibblers, milling cutters, diamond saws and diamond wire cutting etc., but also microwaves for concrete and hydraulic techniques. Examples are shown in Figure 6. A wider overview of decommissioning techniques can be found in (IAEA, 2008a).

Figure 5. Examples for thermal cutting techniques. Plasma cutting.

Figure 6. Examples for mechanical cutting techniques. Left: mechanical shears (BR3, Belgium), Right: diamond wire cutting for large components (HDB, Karlsruhe, Germany).

Materials arising from Decommissioning

This section deals with the various material types arising from decommissioning of nuclear installations. It can only present a short overview of the various material streams and their quantities, and it is necessary to restrict the overview to nuclear power plants, while research reactors and fuel cycle installations present different material compositions and waste streams. The largest part of all material streams will be going to clearance, while only a comparatively small portion will be disposed of as radioactive waste. Because of the completely different management and handling techniques for both parts, management of radioactive waste and clearance of material is dealt with in separate chapters.

Material types from decommissioning

The material types originating from decommissioning of light-water reactors are similar:

- activated and contaminated stainless steel from the reactor pressure vessel and its internals,
- contaminated stainless steel from the circuits, systems and components,
- carbon steel from components and structural elements,
- activated and contaminated concrete from the biological shield and adjacent concrete structures,
- contaminated concrete from building structures,
- electrical installations and cabling,
- insulation material,
- various other materials.

In gas-graphite reactors (GCR), there will also be activated and contaminated graphite.

Material quantities

It is clear that the material quantities that will arise from decommissioning depend on the size of the installation. There is, however, also a dependence on the type of the plant. In nuclear power plants with boiling water reactor (BWR), the secondary circuit and therefore the turbine hall and its components are part of the controlled area. The overall amount of material from BWRs is therefore larger than for a pressurised water reactor (PWR) of similar power rating. Furthermore, the specific type of construction also influences the material quantities. For example, PWRs of Russian type, VVER, have a higher amount of concrete than PWRs of a Western design, as VVERs do not have a full-pressure containment system like Western PWR, but a confinement system which consists of a large number of rooms by which overpressure could be gradually released in the case of an accident. This construction uses a higher amount of concrete per unit power.

Typical quantities of materials are provided in Table 2 for a 900 – 1300 MWe PWR and a 250 MWe GCR. The GCR has a higher mass per unit power than the PWR. In both cases the weight of the materials that are free of contamination and the weight of the building are not shown.

Table 2. Typical quantities of materials from decommissioning for a 900 – 1300 MWe PWR and a 250 MWe GCR (IAEA, 2008b).

Radioactive material	900 – 1300 MWe PWR	250 MWe GCR
activated / irradiated steel	650 Mg	3000 Mg
contaminated steel	3500 Mg	6000 Mg
graphite	-	2500 Mg
activated concrete	300 Mg	600 Mg
contaminated concrete	600 Mg	150 Mg

The weight of the building structures of the controlled area may amount to 200,000 Mg for large PWRs and 400,000 Mg for large BWRs. The structures include a few 10,000 Mg of reinforcement steel and other steel structures that are uncontaminated.

Activity levels and main contaminants

The activity levels, in particular the mass related activity (Bq/g) and the surface related activity (Bq/cm²), vary for the various material types, depending on the origin and history. Activated components of the reactor may reach specific activities of 10^9 Bq/g, the core region of the reactor pressure vessel may reach several 10^6 Bq/g, and activated concrete in the biological shield several 10^3 Bq/g. Surface contamination may range from some 10^5 Bq/cm² on highly contaminated areas down to a few Bq/cm² in less contaminated areas, nearing clearance levels. The main gamma emitting nuclides in LWR materials are Co-60 and Cs-137, causing the main part of the dose rate. Beta emitters like Fe-55, Ni-63 and Ni-59 as well as Sr-90 also contribute to the overall activity but not to the dose rate. Alpha emitters like Am-241 may be present in cases where (smaller or larger) fuel element defects have occurred during operation.

Management of Radioactive Waste

This chapter deals with the management of radioactive waste, while clearance of material is dealt with in the next chapter. First, a short overview of classification schemes for radioactive waste is given. Options for waste management and for decontamination are presented, followed by options for waste disposal.

Classification of radioactive waste

The main categories of decommissioning waste include:

(a) Low level radioactive waste;
(b) Intermediate level radioactive waste;
(c) High level radioactive waste;
(d) Hazardous, non-radioactive waste (chemicals, heavy metals, etc.);
(e) Non-radioactive and non-hazardous waste (cleared material that complies with clearance levels).

Each category of waste has its own unique concerns and specific management requirements.

The classification of radioactive waste is of particular importance when decommissioning nuclear facilities. The existence of guidelines for waste segregation and conditioning, in particular waste acceptance criteria for storage and disposal, can have a

significant impact on the planning of decommissioning activities, particularly cost estimates and the selection of decontamination and dismantling activities.

A widely used classification system based on dose rates identifies waste as one of the following: low level waste (LLW), intermediate level waste (ILW), and high level waste (HLW). However, this system serves mainly to support waste handling and storage activities. Other countries may have different classification schemes, like e.g. Germany where the only distinction is made between heat-generating waste (spent fuel, vitrified waste from reprocessing) and waste with negligible heat generation (all other radioactive waste above clearance levels). For the long term management of decommissioning waste, it may, however, be more desirable to employ a classification based on the activity levels and half lives of the radionuclides contained in the waste, as defined by the IAEA. Figure 7 shows the IAEA waste classification scheme together with the German classification scheme.

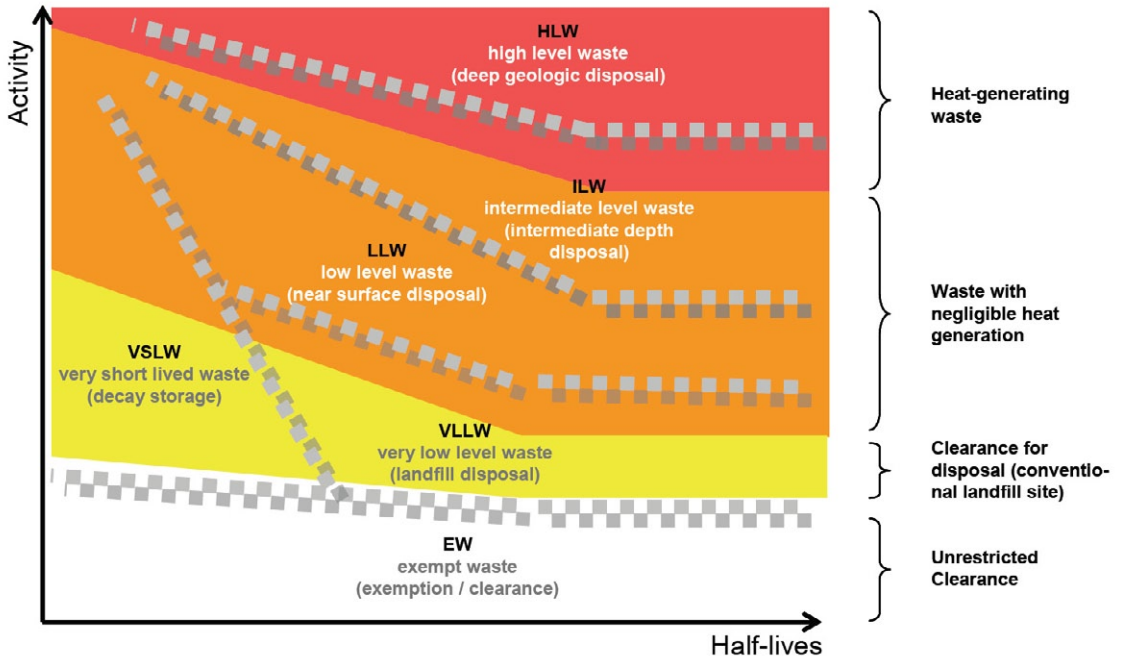

Figure 7. Waste classification according to IAEA and German classification scheme for comparison (German Government 2008).

Waste management options

Waste management starts with thorough characterisation of the material, taking into account the plant history including any incidents that might have led to contamination, activation calculations (for the reactor pressure vessel and the biological shield) and the results of a large number of samples and measurements. Characterisation of the plant is of paramount importance for a smooth implementation of decommissioning. It is, however, not possible to deal with this topic in this paper in any detail.

The basic options for waste management are depicted in Figure 8. In the process on the right starting with radioactive material the first decision is whether any treatment for reduction of activity (segregation of parts with higher and lower activity, decontamination) is technically and economically viable. If not, the material will end as ra-

dioactive waste and may need treatment and conditioning before being put into interim storage and eventually into final disposal. If treatment of the initial material is viable, the material has to be checked for compliance with release criteria (clearance levels). It may turn out that the material exceeds these values and has to be treated as radioactive waste, but in general it will be possible to bring the material to one of various release or clearance options (see next chapter). – The process for material for which contamination is only suspected is similar (branch on the left).

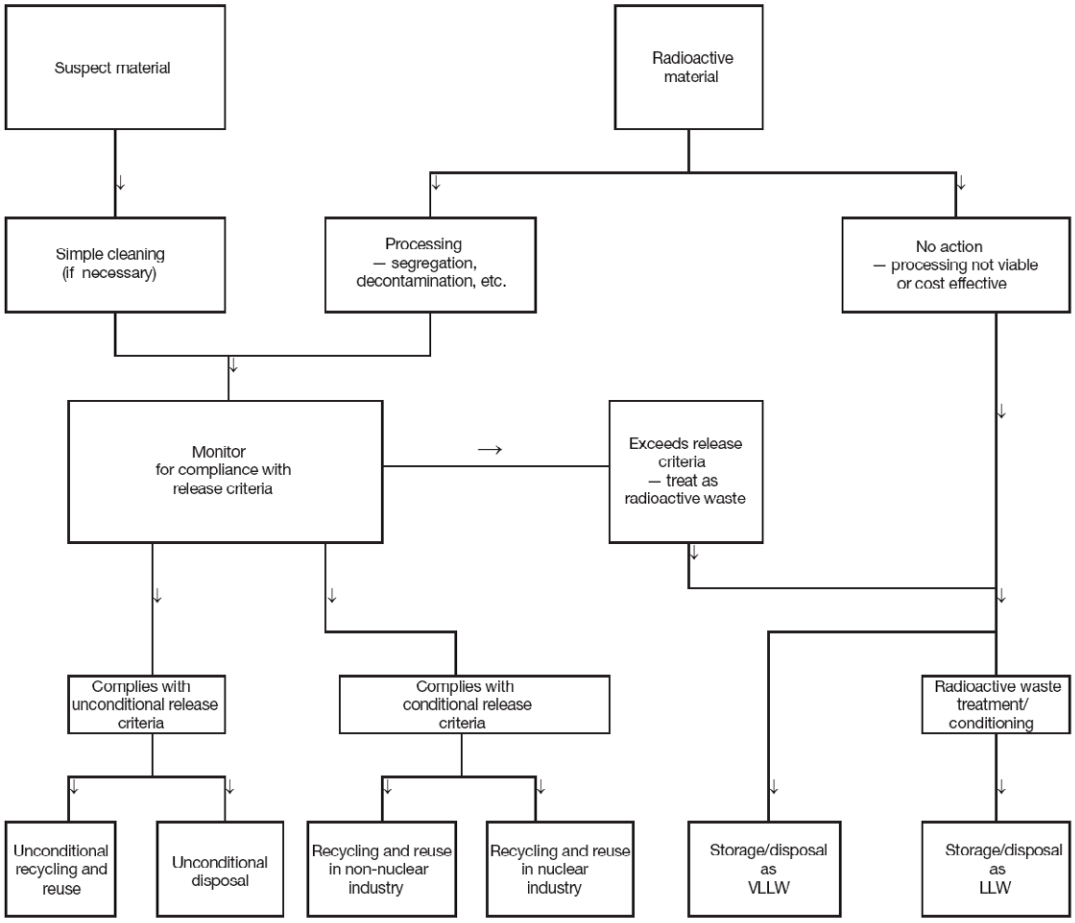

Figure 8. Management and disposition options for the segregation and characterisation of suspect and radioactive material (IAEA, 2008).

Decontamination

Decontamination techniques are based on the removal of surface contamination by mechanical, chemical or other procedures. Activation cannot be removed as it is a property of the entire volume of the material.

In many cases, light decontamination processes may be sufficient, like washing with water or a detergent, high-pressure water jetting or other commonly available processes. However, in many cases a fixed surface layer has to be removed, for which these techniques will not suffice. A widely used procedure for decontamination of metallic surfaces is grit blasting. Figure 9 shows an example of a grit blasting facility in a German

nuclear power plant. The mobile facility is placed in the controlled area and forms part of the material management process. Metallic surfaces are monitored prior to and after blasting to verify the effect of the decontamination method and to decide whether the procedure has been effective so that the material complies with clearance levels. If necessary, the procedure has to be repeated. The steel grit is cleaned and reused until it is fully worn and then forms secondary waste (i.e. waste that arises from processes in addition to the primary waste from the material of the nuclear facility itself).

Figure 9. Left: loading of material into the grit blasting caisson. Right: the grit blasting container from the outside (E.ON, 2008).

Table 3. Overview of decontamination techniques for various purposes (IAEA, 2001).

OBJECTIVE/TASK	COMPONENT/SURFACE TO BE DECONTAMINATED	METHOD EMPLOYED
Decontamination before dismantling Reduction of occupational exposure	Pipeline system	Chemical Mechanical
	Pool/tank	Hydro-jet Blast Strippable coating
Decontamination after dismantling Recycle of contaminated metal Reduction of radioactive waste	Pipes, components	Electropolishing Chemical immersion Blast Ultrasonic wave Gel
Decontamination of building Unconditional release of building Reduction of radioactive concrete waste	Concrete surface	Mechanical: • Scabbler • Shaver • Drill and spalling • Steel grit blast Thermal stress: • Microwave irradiation • Flame scarfing

There are numerous other decontamination techniques like chemical baths using acids or bases in combination with electrolytic processes, called electropolishing. By removing a thin surface layer, the contamination is also removed. A rather comprehensive overview is given in Table 3.

Treatment and conditioning techniques

For management of RW from decommissioning, standard technologies can be used which have been applied already during the operational phase. Usually, the technology must be improved and upgraded to be able to manage the decommissioning RW, mainly because of the much larger quantities. The capacity of the RW management technology must be adapted accordingly. In addition, fragmentation and decontamination of solid RW and new measurement tasks usually requires purchasing of special technological equipment. (Thierfeldt, 2006)

The different NPP types produced different RW either from operation or decommissioning. The NPPs with WWER reactors are usually equipped with tanks with big capacities for storage of liquid or semi liquid RW from NPP operations (concentrates, sludges, ion exchangers). This RW must be treated before the decommissioning activities star. Treatment of such kinds of RW could be difficult from the view of meeting the acceptance criteria for disposal and a special methods and equipment must be developed.

There is a wide range of waste treatment and conditioning techniques of which only a few can be presented in the following.

- Solidification refers to a range of processes and which additives are added to a given batch of waste. The waste is then converted to a single, solid form. Before solidification, the waste may take a variety of forms, liquid, slurry, sludge or dried solid particles.
- Incineration may be used if the waste is burnable and does not contain significant amounts of hazardous substances for which the incineration plant is not suitable. The most important property of incineration is its large volume reduction.
- Melting of metals in a specialised licensed facility will transform surface contamination into volume activity, a kind of decontamination process, and will also result in size reduction. The molten metal can be poured into containers with other radioactive waste thus making effective use of any unused space.
- Encapsulation may be used for small, loose materials, e.g. by fixing the material with cement, polyethylene or other material, depending on the acceptance criteria of the interim storage facility and the repository.
- Supercompaction may be used as an effective process for volume reduction of metallic or other compressible waste. Figure 10 shows a volume reduction facility that can be set up in the controlled area of nuclear installations, forming part of the material management process.

Figure 10. Volume reduction of 200 l drums (FAKIR supercompaction facility, GNS).

Disposal options

The large field of disposal options that exist in various countries can only be briefly touched upon in this paper. Various countries like UK, France or USA prefer near-surface disposal of most wastes resulting from decommissioning. The waste is put into waste containers that are filled with concrete and is then emplaced into trenches lined with concrete which are finally covered in concrete, creating a monolithic waste structure. This waste form will remain stable until the activity will have decayed to levels that are below regulatory concern (several decades or a few hundred years). The amount of long-lived nuclides is usually limited in such repositories.

Other countries like Germany prefer deep geological disposal also for decommissioning waste. The characteristics of the waste containers and the mode of emplacement have been derived from operational and safety analysis of the repository. There are usually fewer restrictions on the activity content of the waste, as the isolation of the waste inside a deep geological repository is superior to a near-surface disposal site.

Waste management and selection of decommissioning strategy

Different aspects of waste generation and waste management can have an impact on the selection of a decommissioning strategy. Among the most important aspects are: the overall national waste management strategy, the amount of waste, the types and categories of waste (both radioactive and non-radioactive) and the facilities needed to process, handle, store and dispose of the waste. The absence of a waste management policy for any of these waste categories will introduce uncertainties in the decommissioning strat-

egy selection process. If no disposal route exists for a particular waste category, the only option may be to store the waste on-site in regulated storage facilities.

Lack of a disposal facility is in itself insufficient reason for not performing immediate dismantling. The waste can be placed into an interim storage facility that will also require decommissioning eventually, once a final disposal scheme for the waste is decided upon. If the nuclear programme within a country is very limited and the type of facility to be decommissioned is amenable to entombment, then this may be the preferred option. As stated previously, an entombed facility has to be considered as a near surface waste disposal site and needs to meet the regulatory requirements for such a facility. This means, for example, that no or only limited amounts of long lived radionuclides are allowed in an entombed facility. If the country has not prepared the necessary infrastructure for a low level waste disposal facility, then entombment may not be feasible. However, the facility could be placed into a short deferred dismantling mode until such requirements can be put into place. (IAEA, 2007)

A country specific analysis of the influence of waste management techniques on decommissioning strategy selection among EU member states yielded three basic types of approaches (Thierfeldt, 2007):

- High influence on strategy selection: dismantling is postponed until a viable waste route becomes available, either in a prolonged form of early decommissioning which is drawn out over a suitably period of time or by choosing the deferred decommissioning option. Examples are NPP Barsebäck (Sweden), the four Italian nuclear power plants, in particular the NPP Latina and the NPP Vandellós 1 (Spain).
- Medium influence on strategy selection: dismantling is commenced although no repository is available. However, in order to manage the radioactive waste streams from decommissioning and to provide a storage even after dismantling of the nuclear installation will have been completed, an interim storage facility is created (e.g. in an existing building at the site which is converted into a storage facility or in a building newly constructed for this purpose). In some cases, the characteristics of this interim radwaste storage facility may have influence on the waste management options and the conditioning techniques, e.g. on the use of waste containers because of limited shielding provided by the waste building, on the mass of waste packages because of limited lifting equipment capacity etc. Examples are the NPPs Würgassen and Stade (Germany) and NPPs in the UK.
- Low or negligible influence on strategy selection: disposal routes for radioactive waste are available, either in the form of an existing repository or of centralised radwaste storage facilities, and consequently waste acceptance criteria are also established. In such a case, dismantling can commence without consideration to the creation of new waste disposal facilities or disposal routes. Examples are mainly decommissioning projects from countries with existing repositories like France

Clearance of Material

Clearance is of central importance for the material management in decommissioning projects. The topic is so broad that the analysis in this paper necessarily has to be very short. The vast majority of material from decommissioning of nuclear installations is either free of contamination (especially the large volumes of concrete from building demolition) or can be brought to sufficiently low levels by decontamination. If it can be demonstrated that the material, the buildings or the site of the facility complies with clearance levels, it can be released from regulatory control.

Clearance levels

Clearance is part of the EURATOM Basic Safety Standards (European Communities, 1996). Therefore, a number of countries have implemented clearance regulations with various degrees of complexity and using different clearance levels. The European Commission has also given guidance on clearance levels and the implementation of clearance for metal scrap, buildings, building rubble and unconditional clearance (European Commission, 1998, 2000a, 2000b). The IAEA has issued recommendations on unconditional clearance as well (IAEA, 2004). Table 4 shows a small comparison of values that are used or have been recommended for use in various countries. This table shows that there for some countries there are large differences for certain values which derive from the radiological models used. The values in many international studies that have been explicitly derived for unconditional clearance, however, are generally close together.

Table 4. Comparison of clearance levels for unconditional clearance for key radionuclides from studies and the regulatory framework of various countries and from international recommendations (values rounded to 1 significant digit; "-" means not analysed or no value given) (Thierfeldt, 2005).

Country/ Organis.	Regulatory framework/ study / recommendation	Clearance level in [Bq/g] for				
		Co 60	Cs 137	Sr 90	Am 241	H 3
Germany	Radiation Protection Ordinance Ann. III Tab. 1 Col. 5	0.1	0.5	2	0.05	1,000
Netherlands	report Timmermans et al. (recycling of steel)	100	100	10,000	100	1E7
Sweden	report Elert et al. (minimum of all clearance options for metals)	0.2	8	4	70	-
UK	report DETR	0.04	0.2	3	0.6	600
UK	Radioactive Substances Act	0.4	0.4	0.4	0.4	0.4
USA	report NUREG 1640	0.04	0.04	1	-	-
EU	RP 122 part 1	0.1	1	1	0.1	100
IAEA	Safety Guide RS-G-1.7	0.1	0.1	1	0.1	100
OECD/NEA	report of Task Group (additional surface specific CL apply)	0.2	0.7	94	50	-

A study on EU member states (Thierfeldt, 2006) showed that the existence of regulations for clearance will not necessarily depend on the size of the nuclear programme. Instead, most countries have implemented clearance regulations as soon as

there is a substantial amount of material which need not be treated as radioactive waste and which has to be dealt with. A number of countries have followed one or more of the recommendations issued by the European Commission when implementing clearance in national regulations. While clearance regulations have been introduced in all countries except France, substantial differences pertain to the detailed design of these regulations, mainly the clearance options and the clearance levels which are in place. The comparison also reveals that the existence of a repository for radioactive waste is obviously not related to the implementation of clearance regulations in a particular country.

Measurement techniques for clearance

A large number of measurement techniques suitable for various clearance options is available which cannot be enumerated and outlined here. In cases where hard to measure radionuclides (alpha emitters, pure beta emitters etc.) form part of the nuclide vector and have to be taken into account, correlation with easy-to-measure nuclides are used. Usually a correlation between hard gamma emitters which have a high abundance in the nuclide vectors (e.g. ^{60}Co, ^{137}Cs), so-called key nuclides, is used. Correlation factors to other nuclides like ^{90}Sr, ^{241}Am etc. are determined from radiological measurements. The overall radionuclide contents of material can then be calculated from measuring the activity of the key nuclides and calculate the activities of the other nuclides via the correlation factors.

The applicability of direct measurements of gamma emitting nuclides as well as of the correlation method depends to a large extent on the percentage that these gamma emitting nuclides have in the nuclide vector. Because of the radioactive decay and because of the relatively short half-life of ^{60}Co (5.3 a) and the long half-lives of hard-to-measure nuclides (30 a for ^{90}Sr, 432 a for ^{241}Am), this percentage is continuously decreasing. Some installations, especially those with a high abundance of alpha emitting nuclides in the contamination, have taken this fact into account when determining the duration of the safe enclosure period. (Thierfeldt, 2006)

Examples for the three most important measurement techniques for material from NPPs are shown in the following figures. Figure 11 shows measurements with surface contamination monitors on metallic components, a technique which is indispensable for metallic material. These instruments are sensitive for beta and/or alpha radiation, so that the nuclide vector has to be known. Larger quantities (several 100 kg up to 1 Mg) of metallic material and bulk material like building rubble can be measured in bulk monitors of various sizes as shown in Figure 12. These facilities are sensitive for gross gamma radiation, so that the nuclide vector has to be known. Finally, building surface can be monitored using in-situ gamma spectrometry as shown in Figure 13. Here, the gamma emitting nuclides are measured individually.

Figure 11. Measurements with surface contamination monitors on metallic components (EON, 2008).

Figure 12. Measurements with a bulk monitor for large quantities of material (up to 1 Mg) (RADOS).

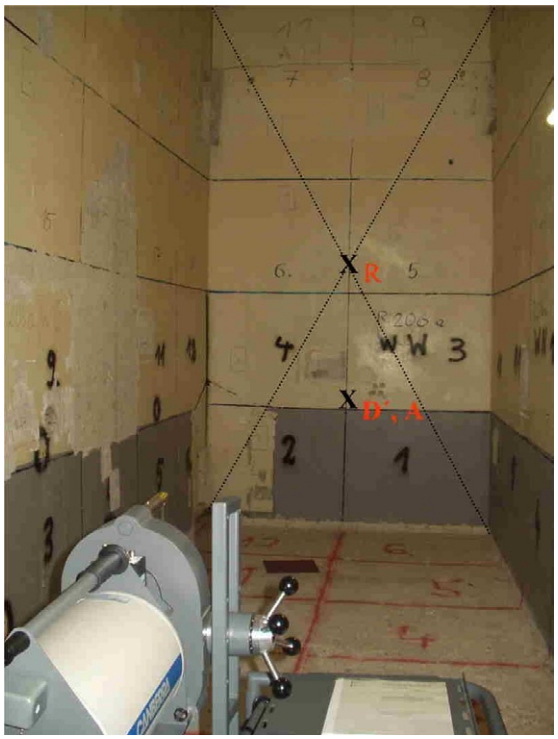

Figure 13. Measurement of a room with in situ gamma spectrometry (Naber, 2007).

In summary, the implementation of clearance or of an equivalent way of dealing with material with extremely low activity is a necessary prerequisite for carrying out active decommissioning. The recommendations on clearance issued by the European Commission (RP 89, 113, 122/I) or the use of similar clearance levels have proven to be viable. International harmonisation of clearance levels between European countries is a goal that has often been called for, but which has been achieved already to a large extent. The recommendations on clearance issued by the European Commission have played an important role as guidance. It is obviously not necessary to have exact harmonisation of the values of all clearance levels for all radionuclides, but to use clearance levels in the same order of magnitude for the most important radionuclides

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Non-ionising radiation

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Abstract
Technological development in modern societies cause an ever increasing exposure of people to non-ionising radiation in all situations of life. Examples include new technologies for diagnosis and treatment in medicine, wireless communication technologies at home and in the office, and high power devices for material processing in industry. The health implications of such exposures have been investigated quite intensively, but remain an issue of public concern. The relevant mechanisms and health effects are different throughout the non-ionising electromagnetic spectrum. Nerve excitation and tissue heating are well established examples. Cancer induction especially in children or from very long chronic exposure is still under investigation. The International Commission on Non-Ionizing Radiation Protection has issued guidance on limiting exposure to static, low, and high frequency electric and magnetic fields, that are regularly reviewed.

Introduction
This refresher course addresses the current scientific evidence of health related risks of exposures to static and low frequency (LF) electric and magnetic fields and high frequency (or radiofrequency (RF)) electromagnetic fields. It illuminates different aspects relevant for setting guidelines on limiting exposure. The following overview is based primarily on reviews conducted by the International Commission on Non-Ionizing Radiation Protection (ICNIRP) (ICNIRP 2003, ICNIRP 2009a) and the World Health Organization (WHO 2006, WHO 2007).

Sources
Exposure to non-ionising radiation can result from both, natural and man made sources. Static electric fields occur naturally in the atmosphere. Beneath thunderclouds values of up to 3 kV m^{-1} can occur. High electric fields from man made sources are the result of either charge separation as a result of friction or the use of high voltages. Common examples are direct current (DC) power transmission that can produce static electric fields of up to 20 kV m^{-1}, rail systems using DC with fields of up to 300 V m^{-1} inside the train, and video display units that can create electric fields of around 10 - 20 kV m^{-1} at a distance of 30 cm. Local fields of up to 500 kV m^{-1} may be generated while walking on non conducting carpets.

The static geomagnetic field varies over the Earth's surface between about 35 - 70 µT. Magnetic flux densities of up to 2 mT have been reported inside electric trains and in developmental magnetic levitation systems. Workers are exposed to larger fields of up to around 60 mT from electrolytic reduction processes of alumina, and electric arc welding produces around 5 mT at 1 cm from the welding cables. Much larger magnetic fields are used in medical diagnosis. The static magnetic field of MRI scanners in routine clinical systems range from 0.2 - 3 T. In research applications, higher magnetic fields up to 9.4 T are used for whole body patient scanning. In terms of exposure, the magnetic flux density at the operator's console is typically about 0.5 mT, and higher. Strong fields are also produced in high-energy technologies such as thermonuclear reactors, magneto hydrodynamic systems and superconducting generators.

The dominant exposure to LF electric and magnetic fields is normally that associated with the generation, transmission and use of electricity at the power frequencies of 50 or 60 Hz. Maximum field strengths of up to several kV m^{-1} might be encountered at ground level below power lines operating at 100 kV or greater. Common sources are the various electrical appliances and tools used in industry and at home. The power frequency electric fields in conventional homes supplied with electricity range up to several hundred volts per metre. Power frequency magnetic fields in homes arise from the flow of current in electrical appliances and circuits both inside and outside the home, and from currents in conducting pipe work and the ground. The strongest LF magnetic fields in homes are usually those generated by electrical appliances. Transmission lines can produce maximum magnetic flux densities of up to a few tens of µT during peak demand, however mean levels are usually no more than a few µT. The magnetic fields encountered in electrified rail systems vary considerably because of the large variety of possible arrangements of power supply and traction. Operators in the vicinity of induction furnaces and equipment heaters may receive some of the highest exposure levels found in industry. Maximum flux densities of 1 - 60 mT have been reported. Incidental exposures to LF electric and magnetic fields can be incurred in a variety of occupational settings. Examples for typically magnetic flux densities that can occur are a few µT from underfloor heating, a few mT from tape erasers, a few mT from motorised equipment, several hundred µT from arc furnaces, and several mT from resistance welding.

At lower radio frequencies, below 30 MHz, the background electromagnetic radiation is mainly due to lightning discharges during thunderstorms. At high radio frequencies, above 30 MHz, the natural EM-fields originate from very broadband blackbody radiation from the warm earth and from extraterrestrial processes, mainly from the sun and the extraterrestrial microwave background radiation from the whole sky. The power density of the radiation component emitted by the warm surface of the ground at 300 K temperature (27 °C) is a few $mW \cdot m^{-2}$.

Man-made high frequency sources produce local field levels many orders of magnitude above the natural background. The antennas of broadcast stations are the most powerful continuous sources of high frequency electromagnetic energy intentionally radiated into free space. The cellular mobile telephone industry has undergone rapid growth resulting in a wide use of wireless communication devices in all parts of modern society today. The mobile phone networks consist of a system of adjoining 'cells', each with its own base station that sends and receives radio signals.

The base stations have power outputs ranging from tens of Watt in macrocells down to a fraction of a Watt in picocells. The exposure levels from these mobile telecommunications base stations are very low. Strong high frequency fields are encountered in medicine. Here high frequency fields are used for example for tissue heating (hyperthermia, diathermia), ablation, telemetry, and diagnostic purposes (MRI). High power electromagnetic sources are also used for material processing in industrial and domestic environments, for electronic article surveillance, for identification, and various RADAR applications. Microwave fields are also considered as carrier for wireless transport of electrical energy.

Mechanisms and effects

The three established physical mechanisms through which static magnetic fields interact with living matter are magnetic induction, magneto-mechanical, and electronic interactions. Magnetic induction originates through electrodynamic interactions with moving electrolytes. Static fields exert Lorentz forces on moving ionic charge carriers and thereby give rise to induced electric fields and currents. In accordance with Faraday's law of induction electric fields and currents within the body may also result from exposure to time varying magnetic fields, or from movement in a static magnetic field. The two types of mechanical effects that a static magnetic field can exert on biological objects are magneto-orientation and magneto-mechanical translation. In a static field, paramagnetic molecules experience a torque that orients them in a configuration that minimizes their free energy within the field. In the presence of gradients, static magnetic fields produce a net translational force on both diamagnetic and paramagnetic materials. Magnetic fields affect the recombination of radical pairs involved in an intermediate state of certain metabolic reactions.

Concerning magnetic field effects, different levels of biological organization at a cellular level have been investigated, including cell-free systems and various cell models using both bacteria and mammalian cells. Only a few in vitro studies on genotoxicity have been performed. Overall there is little convincing evidence from cellular and cell-free models for biologically harmful effects of exposure to static magnetic fields with flux densities up to several T. A large number of animal studies on the effects of static magnetic fields have been conducted (Saunders 2005). The most consistent responses seen in behavioural studies suggest that the movement of laboratory rodents in magnetic fields of about 4 T and higher may lead to aversive responses and conditioned avoidance. A well-established effect of exposure of animals to static magnetic fields greater than about 0.1 T is the induction of blood flow potentials in and around the heart and other major vessels of the central circulatory system. Exposures to static fields of up to 1 T have not been demonstrated to have an effect on foetal growth or postnatal development in mice. It is not possible to draw any conclusions from animal studies on possible genotoxic or carcinogenic effects of static magnetic fields. Detailed physiological studies in humans evaluating various physiological parameters including body (sublingual) temperature, respiratory rate, pulse rate, blood pressure, and finger oxygenation levels have not shown any pronounced effects of exposure to magnetic fields up to 8 T (Chakeres et al. 2003). Behavioural studies carried out on subjects situated in the proximity of MR systems with exposures up to 7 T have suggested that there may be a transient negative

influence of exposure on eye-hand coordination and visual contrast sensitivity associated with head movement in the field (de Vocht et al. 2006, 2007a, 2007b). Overall, current information does not indicate any serious health effects resulting from the acute exposure of stationary humans to static magnetic fields up to 8 T. It should be noted, however, that such exposures can lead to potentially unpleasant sensory effects such as vertigo and transient decrements in the performance of some behavioural tasks during head or body movement. The few available epidemiological studies have mostly been conducted on workers exposed to moderate static magnetic fields of up to several tens of mT either working in aluminium smelters or chloralkali plants or as welders. Overall, the few available epidemiological studies on static magnetic fields have methodological limitations and leave a number of issues unresolved concerning the possibility of risk of cancer or other outcomes from long-term exposure. These studies do not indicate strong effects of static magnetic field exposure of the level of tens of mT on the various health outcomes studied, but they would not be able to detect small to moderate effects.

Time varying electric fields external to the body induce a surface charge on the body; this results in induced currents in the body, the distribution of which depends on exposure conditions, on the size and shape of the body, and on the body's position in the field. Surface charges may cause well-defined biological responses, ranging from perception to annoyance. The physical interaction of time-varying magnetic fields with the human body results in induced electric fields and circulating electric currents. Induced electric fields inside the body can be calculated by computational methods using anatomically and electrically realistic models of the body, which have a high degree of anatomical resolution. The responsiveness of electrically excitable nerve and muscle tissue to electric stimuli including those induced by exposure to LF electric and magnetic fields has been well established for many years. Myelinated nerve fibres of the peripheral nervous system have been stimulated by pulsed electric fields induced during volunteer exposure to the switched gradient magnetic fields of magnetic resonance (MR) systems. The most robustly established effect of electric fields below the threshold for direct nerve or muscle excitation is the induction of magnetic phosphenes, a perception of faint flickering light in the periphery of the visual field, in the retinas of volunteers exposed to LF magnetic fields. The retina as part of the CNS is regarded as an appropriate, albeit conservative model for induced electric field effects on CNS neuronal circuitry in general.

The experimental data available so far do not indicate that LF electric and/or magnetic fields affect the neuroendocrine system in a way that would have an adverse impact on human health. It has been hypothesized that exposure to LF fields is associated with several neurodegenerative diseases. For Parkinson's disease and multiple sclerosis the number of studies has been small and there is no evidence for an association between LF exposure and these diseases. For Alzheimer's disease and amyotrophic lateral sclerosis (ALS) more studies have been published. Overall, the evidence for the association between LF exposure and Alzheimer´s disease and ALS is inconclusive. Experimental studies of both short-term and long-term exposure indicate that hazardous cardiovascular effects associated with LF fields are unlikely to occur at exposure levels commonly encountered environmentally or occupationally.

Epidemiological studies have not shown an association between human adverse reproductive outcomes and maternal or paternal exposure to LF fields. There is limited evidence for an increased risk of miscarriage associated with maternal magnetic field exposure, but overall, the evidence for an association between LF fields and developmental and reproductive effects is very weak. A considerable number of epidemiological studies, carried out particularly during the 1980s and 90s, indicated that long term exposure to weak LF magnetic fields, orders of magnitude below the recommended limits, might be associated with childhood leukaemia. In contrast to the epidemiological evidence of an association between childhood leukaemia and prolonged exposure to power frequency magnetic fields, the animal cancer data, particularly those from large-scale lifetime studies, are almost universally negative.

Exposure to electromagnetic fields at frequencies above about 100 kHz can lead to significant absorption of energy and temperature increases. In general, exposure to a uniform (plane-wave) electromagnetic field results in a highly non-uniform deposition and distribution of energy within the body, which must be assessed by dosimetric measurement and calculation. As regards absorption of energy by the human body, electromagnetic fields can be divided into four ranges:

- frequencies from about 100 kHz to less than about 20 MHz, at which absorption in the trunk decreases rapidly with decreasing frequency, and significant absorption may occur in the neck and legs;
- frequencies in the range from about 20 MHz to 300 MHz, at which relatively high absorption can occur in the whole body, and to even higher values if partial body (e.g., head) resonances are considered;
- frequencies in the range from about 300 MHz to several GHz, at which significant local, non uniform absorption occurs; and
- frequencies above about 10 GHz, at which energy absorption occurs primarily at the body surface.

There are several theoretical hypotheses describing potential non-thermal mechanisms for biological effects from low level exposure. Some have been tested experimentally, but so far there has been no compelling evidence that they might plausibly account for any significant health effect.

Over the last 30 years there have been many in vitro studies on potential cellular effects of high frequency electromagnetic exposure. These studies gave insight into the basic mechanisms by which effects might be induced in more complex animal or human organisms. Interpretation of the experimental results is, however, limited by anomalous cell behaviour generated by the culture conditions and other factors which limit the extrapolation to humans. The studies conducted so far have not provided consistent evidence of biological effects under non-thermal electromagnetic exposure conditions. More recently, studies have been carried out using powerful high-throughput screening techniques capable of examining changes in the expression of very large numbers of genes. Such studies often showed a limited number of alterations where some genes were up- and others down-regulated. These advances in molecular studies are promising, but not yet decisive in risk evaluation.

Animal studies further support the conclusion that the most consistent and reproducible responses to acute high frequency exposure result from the induced heating. These studies established that, in general, an increase in body temperature

elicits several cardiovascular changes including increased blood flow to the skin, increasing skin thermal conductance, and increased cardiac output, primarily due to an increase in heart rate, in order to maintain arterial pressure within the normal range. Deficits in learned behaviours, particularly the disruption of ongoing operant behaviours, occur mainly when core temperatures are increased by about 1°C or more. Similar rises in body temperature also result in significantly enhanced plasma corticosterone or cortisol levels in rodents and primates and transient changes in immune function and hematology, generally consistent with the acute responses to non-specific stressors. In addition, high frequency radiation can cause increased embryo and foetal losses, increased incidence of foetal malformations and anomalies, reduced foetal weight at term and impair male fertility at exposure levels that are sufficiently high to cause a significant increase in temperature. High frequency induced cataract also remains a well-established thermal effect of exposure in anaesthetised rabbits. However, primates appear less susceptible to cataract induction, and opacities have not been observed following either acute or prolonged exposures. The results of recent carcinogenicity studies in animals are rather consistent and indicate that carcinogenic effects are not likely at specific power absorption rates (SAR) of up to 4 W kg^{-1} averaged over the whole body, even for long-term exposure. To date, there is no consistent evidence of relevant effects at non-thermal exposure levels in animals.

The most consistent effects of acute exposure of human subjects to electromagnetic fields are the thermoregulatory responses to the induced heating. Volunteer studies indicate that exposed subjects can accommodate whole body heat loads of up to several 4 W kg^{-1} with minimal changes in core temperature. Increased skin blood flow and profuse localized sweating minimise skin temperature rises in response to high local peak SAR. Most volunteer studies have investigated the effects of electromagnetic exposures characteristic of mobile phone use, usually to the head, on a number of physiological parameters including brain electrical activity and blood flow, cognition, and more generally on the endocrine and cardiovascular systems. Some evidence suggests that such exposure may affect the spontaneous EEG in volunteers, but results overall are variable and inconsistent. The small changes seen in brain electrical activity and possibly in regional cerebral blood flow may not have any functional significance. Despite the large number of studies concerned with the issue, no consistent effects of electromagnetic fields on cognitive performance have been found so far. There is no robust evidence of any effect of mobile phone type radiation on children or adolescents. A wide range of subjective symptoms including headaches and migraine, fatigue, and skin itches have been attributed to various electromagnetic sources both at home and at work. However, the evidence from double-blind provocation studies suggests that the reported symptoms are not causally related to EMF exposure.

Results of epidemiological studies to date give no consistent or convincing evidence of a causal relation between electromagnetic exposure and any adverse health effect. On the other hand, these studies have too many deficiencies to rule out an association. A key concern across all studies is the quality of assessment of electromagnetic exposure, including the question of whether such exposure was present at all. Another general concern in mobile phone studies is that the time periods that have been examined to date are necessarily short. The implication is that if a longer

time period is required for a health effect to occur, the effect could not be detected in these studies. Another gap in the research is children. No study population to date has included children, with the exception of studies of people living near radio and TV antennas.

Recommendations

ICNIRP is currently in the process of reviewing and revising its guidance concerning the limitation of exposure in the whole non-ionising radiation range. Guidelines on static magnetic fields have recently been published, revised guidelines on LF fields (1Hz – 100 kHz) have already undergone broad consultation and guidelines on radiofrequency fields (up to 300 GHz) will be reviewed synchronously to the health risk assessment process of the World Health Organization.

The exposure guidelines developed by ICNIRP are intended to protect against the adverse health effects of NIR exposure. Because adverse consequences of NIR exposure can vary across the entire range from trivial to life threatening, a balanced judgement is required before deciding on exposure guidance. Annoyance or discomfort may not be pathological per se but, if substantiated, can affect the physical and mental well being of a person and the resultant effect is considered to be a potential health hazard. ICNIRP seeks to define what is meant by adverse effects in its specific scientific reviews and guidelines. Examples are provided in Table 1.

Table 1. Relevant mechanisms of interaction, adverse effects, and biologically effective physical quantities used in different parts of the electromagnetic field spectrum.

field type	mechanism, interaction	adverse effect	effective quantity
static electric field	surface electric charge	annoyance of surface effects, shock	external electric field strength
static magnetic field	magnetic induction	annoyance of sensory effects, cardiovascular and CNS effects	external magnetic flux density
LF electric field	surface electric charge; induction of electric fields	annoyance of surface effects, shock; stimulation of PNS and CNS	external electric field strength; internal electric field strength
LF magnetic fields	induction of electric fields	stimulation of PNS and CNS	internal electric field strength
RF electromagnetic field	induction of electric fields; absorption of energy within the body; surface absorption of energy; thermo-acoustic wave propagation.	excessive heating, electric shock and burn; excessive surface heating; annoyance from microwave hearing effect	Specific energy absorption rate; Power density; Specific energy absorption

In general, separate guidance is given for occupational exposures and exposure of the general public. The limits for occupational exposure in the ICNIRP guidelines apply to those individuals who are exposed as a result of performing their regular or assigned job activities.

For static magnetic fields ICNIRP recommended that occupational exposure of the head and trunk should not exceed a spatial peak magnetic flux density of 2 T, to prevent vertigo, nausea and other potentially annoying sensations (ICNIRP 2009b). For work applications for which exposures above 2 T are deemed necessary, exposure up to 8 T can be permitted if the environment is controlled and appropriate work practices are implemented to control movement-induced effects. When restricted to the limbs, maximum exposures of up to 8 T are acceptable. Exposure of the general public should not exceed 400 mT (any part of the body). However, because of potential indirect adverse effects practical policies need to be implemented to prevent inadvertent harmful exposure of people with implanted electronic medical devices and implants containing ferromagnetic materials, and injuries due to flying ferromagnetic objects, and these considerations can lead to much lower restriction levels. The limits recommended for occupational and general public exposures to static magnetic fields are summarized in Table 2.

Table 2. Limits of exposure[a] to static magnetic fields.

Exposure characteristics	Magnetic flux density
Occupational	
Exposure of head and of trunk	2 T
Exposure of limbs	8 T
General public	
Exposure of any part of the body	400 mT

[a] ICNIRP recommends that these limits should be viewed operationally as spatial peak exposure limits.

Concerning time varying electric and magnetic fields, ICNIRP (ICNIRP 1998) recommended two classes of guidance: Basic restrictions that are based directly on established health effects. Depending upon the frequency of the field, the physical quantities used to specify these restrictions are internal electric field strength (E), current density (J), specific energy absorption rate (SAR), and power density (S). Reference levels are provided for practical exposure assessment purposes to determine whether the basic restrictions are likely to be exceeded. The derived quantities are electric field strength (E), magnetic field strength (H), magnetic flux density (B), power density (S), and currents flowing through the limbs (I).

For exposure to LF electric and magnetic fields, basic restrictions are provided on current density to prevent effects on nervous system functions. In the frequency range from a few Hz to 1 kHz, for levels of induced current density above 100 mA m^{-2}, the thresholds for acute changes in central nervous system excitability and other acute effects such as reversal of the visually evoked potential are exceeded. In view of safety considerations, occupational exposure should be limited to fields that induce current densities less than 10 mA m^{-2}. For the general public an additional factor of 5 is applied. ICNIRP is currently revising its guidance on limiting exposure to LF electric and magnetic fields. Different modifications with respect to the 1998 guidelines are under

consideration. Protection from effects on central and peripheral nervous system are discussed. Field effects caused in the retina are considered a model for effects in the brain. With regard to the basic restriction, the induced electric field seems to be closer related to the biological effects and thus might replace the induced current density in future recommendations. In deriving reference levels, the most recent advances in numerical dosimetry will have to be considered.

For exposure to high frequency electromagnetic fields, the 1998 guidelines have been reconfirmed recently (ICNIRP 2009c). It is the opinion of ICNIRP that the scientific literature published since the 1998 guidelines has provided no evidence of any adverse effects below the basic restrictions and does not necessitate an immediate revision of its guidance on limiting exposure to high frequency electromagnetic fields. The biological basis of such guidance remains the avoidance of adverse effects such as "work stoppage" caused by mild whole body heat stress and/or tissue damage caused by excessive localized heating. there has been considerable advance in dosimetric investigations in terms of precision and resolution (Lin 2007). A special concern was raised with regard to numerical computations using anatomical models of human bodies which might influence the derivation of reference levels from the basic restrictions. Some published studies showed that in the frequency ranges of body resonance (~100 MHz) and from 1 to 4 GHz for bodies shorter than 1.3 m in height (corresponding approximately to children aged 8 y or younger) at the recommended reference level the induced SARs could be up to 40% higher than the current basic restriction under worst-case conditions. However, this is considered negligible compared with the large reduction factor of 50 (5,000%) for the general public. The basic restrictions recommended for occupational and general public exposures to high frequency electromagnetic fields are summarized in Table 3.

Table 3. Limits of exposure to high frequency electromagnetic fields.

Exposure characteristics	Frequency range	Current density mAm^{-2}	Whole body average SAR $W\ kg^{-1}$	Localised SAR (head/trunk) $W\ kg^{-1}$	Localised SAR (limbs) $W\ kg^{-1}$
Occupational					
	100 kHz-10 MHz	f/100	0,4	10	20
	10 MHz-10 GHz		0,4	10	20
General public					
	100 kHz-10 MHz	f/500	0,08	2	4
	10 MHz-10 GHz		0,08	2	4

Basic restrictions for frequencies between 10 and 300 GHz limit power density to 50 W m^{-2} (occupational exposure) and 10 W m^{-2} (general public).

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