From: Robert Meck -RES
To: ISCORS RECYCLE SUBCOMMITTEE
Date: Mon, Apr 21, 2003 11:28 AM
Subject: IAEA GUIDANCE ON CLEARANCE--U.S. Member State comments

Dear Colleagues:

Attached are the files for our review of the subject guidance. Formal comments are due to the IAEA by August 14, 2003. Because of the time it takes to coordinate comments and get concurrence, I propose that we meet to gather and review our comments in mid- or late-June. Please state your preference for which week and which days of the week are feasible. I'll arrange for a meeting room here at NRC. However, if the majority of the subcommittee finds it more convenient to meet in downtown DC, then I'm open to another subcommittee member making such arrangements.

In the meantime, I'll put the electronic versions of the tables in the latest IAEA draft into a spreadsheet for comparisons. I expect that the final NRC dose assessments (NUREG-1640) will be available soon. I'll include those in the spreadsheet, too. If other agencies wish for their assessments to be included, please contact me. There are other levels such as the Reg. Guide 1.86 and ANSI N13.12 levels that would be useful for inclusion, also. It will be several weeks before I can complete this task. However, I plan to get the comparisons to you in advance of our meeting in June.

Of course, your comments and feed-back are welcome.

Best regards,

Bob

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B-26

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723-J9.03.1 Circ.

The Secretariat of the International Atomic Energy Agency presents its compliments to the Ministries of Foreign Affairs of Member States of the Agency and has the honour to request that they draw the attention of the appropriate Governmental authorities to review the following draft safety standard

Radioactivity in Material not requiring Regulation for Purposes of Radiation Protection

This document is submitted in order to provide Member States and their experts the opportunity for a simultaneous review and evaluation of the document. The English version is enclosed.

Any proposed changes to this document resulting from the review by Member States will be taken into account in the finalization of the safety standard.

Comments on the document should be provided in accordance with the guidance given in the attached Explanatory Note.

A supporting draft Safety Report is also submitted for your information. Any significant comments directly affecting the draft Safety Guide will be considered by the Agency in finalizing this draft Safety Report.

The Secretariat of the International Atomic Energy Agency avails itself of this opportunity to renew to the Ministries of Foreign Affairs the assurances of its highest consideration.

Attachments

Statement by the Commission on Safety Standards

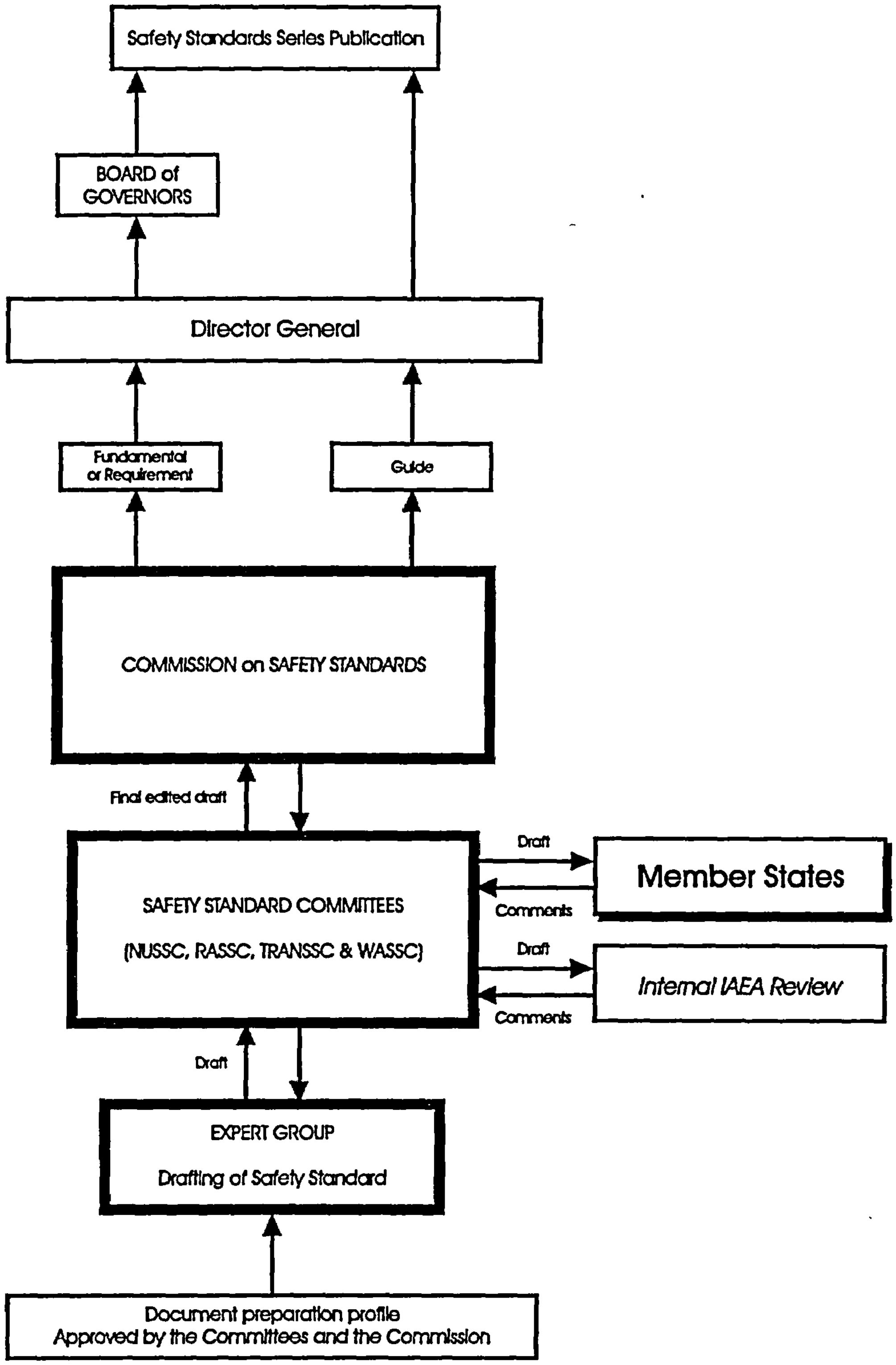
The IAEA's safety standards are prepared and reviewed in accordance with a uniform process. To this end, the Commission on Safety Standards and four Committees (NUSSC, RASSC, WASSC and TRANSSC) with harmonized terms of reference were established in 1996. The Commission has a special overview role with regard to the Agency's safety standards and provides advice to the Director General on the overall programme on regulatory aspects of safety.

The uniform preparation and review process involves: organizing expert group meetings; arranging at different stages of preparation for the internal review of draft texts; submitting documents to the relevant Committee(s) for review; submitting draft texts to the Agency's Member States for comment; and submitting the final edited draft of the safety standards[1] for endorsement by the Commission before publication (see attached flow chart).

The Commission on Safety Standards stresses the importance of Member States' comments to the preparation and review process for safety standards. The Agency's safety standards should not only be of the requisite quality but should represent the consensus view of the Member States and should address the issues of importance to the Member States. While the Commission, the Committees and the Secretariat strive to provide safety standards that satisfy these criteria, the review of draft standards in the Member States is an essential stage in obtaining the broadest possible technical consensus and the highest possible quality and relevance.

Member States are also encouraged to provide the IAEA with feedback on the use of the published safety standards. The full text of recent safety standards and the status of safety standards in preparation are posted on the IAEA's Web site [www.iaea.org/ns/coordinet]. The responsible IAEA officer is Mr. A. Karbassioun of the Department of Nuclear Safety. He may be contacted for further information in connection with this subject at (0043)-1-2600-22696 or through e-mail at a.karbassioun@iaea.org

[1] Safety Guides are published under the authority of the Director General. Safety Fundamentals and Safety Requirements publications require the approval of the Board of Governors, after endorsement by the Commission.



SAFETY STANDARDS PREPARATION PROCESS

SHORT EXPLANATORY NOTE

This draft Safety Guide is intended to assist Member States in clarifying what activities need to be regulated for radiation safety purposes. It has been prepared within the Agency's Safety Standards programme, and its development has included Consultants Meetings, Technical Meetings and meetings of the Safety Standards Committees (RASSC, WASSC and TRANSSC).

Objective

The objective of this Safety Guide is to specify levels of activity concentration in material below which regulation for the purposes of radiation protection should not be required.

History

The enclosed documents have drawn upon the Secretariat's earlier work related to defining the scope of regulation, including a 1999 draft Safety Guide containing criteria for the clearance of material from practices, as defined in the Basic Safety Standards. The Safety Guide was in a final draft stage ready for Member States review prior to the 2000 General Conference.

The 44th General Conference in 2000 adopted a resolution (GC(44)/RES/15) which requested the Agency's Secretariat "*to develop, using the Agency's radiation protection advisory mechanisms and in collaboration with the competent organs of the United Nations and with the specialized agencies concerned, during the next two years and within available resources, radiological criteria for long-lived radionuclides in commodities, particularly foodstuffs and wood, and to submit them to the Board of Governor's for its approval*".

In considering the implementation of the resolution, concern was expressed within the Secretariat that the establishment of criteria for commodities could create confusion because of the activity concentrations already in existence in the BSS for exemption and the additional levels being generated for clearance. It was also noted that activity concentrations exist for free trade in foodstuffs during the first year following an accident and for drinking water quality. A meeting of senior experts was held in Chilton, United Kingdom to discuss a systematic approach that would provide a single set of values for determining when material would require regulatory control. In the pursuing 18 months, four Consultants Meetings and four Technical Meetings were held with the aim of producing a single set of values that would at the same time meet the requirements of the General Conference resolution. The earlier draft Safety Guide developed for clearance was used as a foundation for this effort.

In March 2002, the Waste Safety Standards Committee (WASSC) and the Radiation Safety Standards Committee (RASSC) agreed that a draft Safety Guide (DS161) should be sent to Member States for comment. The Committees also requested the associated Safety Report, which provides the basis for the activity concentrations presented in the draft Safety Guide, be provided as background information.

Almost 300 comments were received from Member States concerning the draft Safety Guide. On consideration of these comments it was found to be necessary to introduce significant modifications to the documents. They were then resubmitted to TRANSSC, RASSC and WASSC for their review. Because of the modifications and because of further changes introduced by the Committees themselves, RASSC and WASSC at their meeting in March 2003 requested that the draft documents again be sent to Member States for review and further comment.

Issues

During the RASSC, WASSC and TRANSSC meetings of early 2003, a number of issues related to the former draft were identified and discussed. The current document reflects the position reached by the Committees concerning these issues, some of which are indicated below. It may assist the Member States when commenting on the draft Safety Guide to note the following points.

- *Purpose of the document*

The purpose of this Safety Guide is to assist Member States in defining as simply as possible a boundary of applicability of radiation protection requirements. This is of benefit to both the regulator and industry, since the use of it should avoid unnecessary regulatory oversight of everyday activities that are of no regulatory concern. The Safety Guide deals only with material containing radionuclides and sets activity concentration levels below which regulatory control of that material, or of the activities within which they are used, should not be required.

An important potential application of the specified levels of activity concentrations is in facilitating the transboundary movement of material containing only trace amounts of radionuclides.

- *Relationship of this document with the Basic Safety Standards*

The principle of placing bounds on the scope of regulation is not new. The Basic Safety Standards employs the concept of *exclusion* to avoid the requirements applying to exposures that cannot realistically be controlled. The BSS requirements do not apply to exposures that are *unamenable* to control, such as exposures from ^{40}K in the body or from unmodified concentrations of most raw material. In most cases, the concept of exclusion is applied to naturally occurring sources of exposure and there has been a desire for a more explicit demarcation of what is unamenable to control. This Safety Guide provides further guidance by specifying activity concentration levels for naturally occurring radionuclides in material below which regulation for their use is not warranted. These levels have been established on the basis of comparisons with activity concentrations commonly found in soils around the world.

In the case of 'artificial' radionuclides (i.e., those produced as a consequence of human activities), the BSS concepts of *exemption* and *clearance* have been used to establish the relevant activity concentration values based on doses to individuals that are considered trivial. The activity concentration values in this Safety Guide differ from those of Table I of the BSS because they relate to bulk quantities of material; those in the BSS are relevant to moderate quantities.

In discussion of this draft Safety Guide by RASSC and WASSC, it was suggested that some further guidance might be required on how the activity concentration levels should be implemented in a regulatory context. The Secretariat has included further text to take account of this request. Nevertheless, if Member States feel that further guidance is necessary or appropriate, it would be greatly assist the Secretariat if specific suggestions could be made.

This Safety Guide is supported by a Safety Report which describes the method used to derive the activity concentrations contained in Section 4 and Table I of the Safety Guide. The Safety Report is attached as part of the package being sent to Member States.

Comments on the scope, approach and content of the enclosed Safety Guide are requested. Comments of an editorial nature will be considered; however, it should be noted that the document would be comprehensively edited by the Secretariat prior to publication.

The comments should be made in English, should refer to the paragraph number in the document being reviewed, and when appropriate should provide alternative text. Please use the attached comment form for documenting all comments.

Any comments should be received by the Secretariat by **14 August 2003.** The responsible IAEA officer is Mr. D. W. Reisenweaver of the Department of Nuclear Safety and Security. He may be contacted for further information in connection with this subject at +43-1-2600-22852 or through e-mail at d.reisenweaver@iaea.org.

The Secretariat will take due account of the comments and produce a revised version of the draft Safety Guide for consideration of the WASSC/RASSC Committees at their next meetings.

RADIOACTIVITY IN MATERIAL NOT REQUIRING REGULATION FOR PURPOSES OF RADIATION PROTECTION, DS161

COMMENTS BY REVIEWER Reviewer: Page.... of.... Country/Organization: Date:				RESOLUTION			
Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection

DS161
April 2003

IAEA SAFETY STANDARDS SERIES

Status: Approved by RASSC and WASSC on 27 March 2003 for 2nd round submittal to Member States for comment.

Radioactivity in Material not requiring Regulation for Purposes of Radiation Protection

DRAFT SAFETY GUIDE DS161

INTERNATIONAL
ATOMIC ENERGY AGENCY
VIENNA

April 2003

(Front inside cover)

IAEA SAFETY RELATED PUBLICATIONS

IAEA SAFETY STANDARDS

Under the terms of Article III of its Statute, the IAEA is authorized to establish standards of safety for protection against ionizing radiation and to provide for the application of these standards to peaceful nuclear activities.

The regulatory related publications by means of which the IAEA establishes safety standards and measures are issued in the **IAEA Safety Standards Series.** This series covers nuclear safety, radiation safety, transport safety and waste safety, and also general safety (that is, of relevance in two or more of the four areas), and the categories within it are **Safety Fundamentals, Safety Requirements** and **Safety Guides.**

Safety Fundamentals (blue lettering) present basic objectives, concepts and principles of safety and protection in the development and application of nuclear energy for peaceful purposes.

Safety Requirements (red lettering) establish the requirements that must be met to ensure safety. These requirements, which are expressed, as 'shall' statements, are governed by the objectives and principles presented in the Safety Fundamentals.

Safety Guides (green lettering) recommend actions, conditions or procedures for meeting safety requirements. Recommendations in Safety Guides are expressed as 'should' statements, with the implication that it is necessary to take the measures recommended or equivalent alternative measures to comply with the requirements.

The IAEA's safety standards are not legally binding on Member States but may be adopted by them, at their own discretion, for use in national regulations in respect of their own activities. The standards are binding on the IAEA in relation to its own operations and on States in relation to operations assisted by the IAEA.

Information on the IAEA's safety standards programme (including editions in languages other than English) is available at the IAEA Internet site

www.iaea.org/ns/coordinet

or on request to the Safety Co-ordination Section, IAEA, P.O. Box 100, A-1400 Vienna, Austria.

OTHER SAFETY RELATED PUBLICATIONS

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

Reports on safety and protection in nuclear activities are issued in other series, in particular the **IAEA Safety Reports Series,** as informational publications. Safety Reports may describe good practices and give practical examples and detailed methods that can be used to meet safety requirements. They do not establish requirements or make recommendations.

Other IAEA Series that include safety related sales publications are the **Technical Reports Series**, the **Radiological Assessment Reports Series** and the **INSAG Series**. The IAEA also issues reports on radiological accidents and other special sales publications. Unpriced safety related publications are issued in the **TECDOC Series**, the **Provisional Safety Standards Series**, the **Training Course Series**, the **IAEA Services Series** and the **Computer Manual Series**, and as **Practical Radiation Safety Manuals** and **Practical Radiation Technical Manuals**.

FOREWORD
by Mohamed ElBaradei
Director General

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

The following bodies oversee the development of safety standards: the Commission for Safety Standards (CSS); the Nuclear Safety Standards Committee (NUSSC); the Radiation Safety Standards Committee (RASSC); the Transport Safety Standards Committee (TRANSSC); and the Waste Safety Standards Committee (WASSC). Member States are widely represented on these committees.

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

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

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

The physical protection of fissile and radioactive materials and of nuclear power plants as a whole is mentioned where appropriate but is not treated in detail; obligations of States in this respect should be addressed on the basis of the relevant instruments and publications developed under the auspices of the IAEA. Non-radiological aspects of industrial safety and environmental protection are also not explicitly considered; it is recognized that States should fulfill their international undertakings and obligations in relation to these.

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

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

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

EDITORIAL NOTE

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

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

The English version of the text is the authoritative version.

[For editions in other languages, add on the copyright page a disclaimer for the translation if necessary.]

CONTENTS

1. INTRODUCTION

BACKGROUND

1.1. The *International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources* (the BSS) [1] specify the basic international requirements for protection of health against exposure to ionizing radiation (hereinafter termed radiation), and for the safety of radiation sources (including their security). The BSS are based on the estimates of the detrimental effects attributed to radiation exposure provided by the United Nations Committee on the Effects of Atomic Radiation (UNSCEAR) [2], as well as on recommendations of the International Commission on Radiological Protection (ICRP) [3] and are intended to provide the basis for the regulation of both '*practices*'[1] and '*interventions*'[2]. The BSS presume the existence of a national infrastructure for radiation safety and a complementary document [4] establishes the basic requirements for the legal and governmental infrastructure that is necessary in order to implement the BSS effectively. An essential component of this infrastructure is the existence of a competent national regulatory body that has the authority to establish regulations. Such regulations shall, inter alia, define the scope of situations to be regulated for purposes of radiation protection. There should also be provision for notification and authorization of practices and sources within practices, and for exemption from the requirements for practices, subject to the criteria defined in the BSS.

1.2. Humans incur radiation doses from exposure to radionuclides, which can cause either direct irradiation from outside the body or be taken into the body and irradiate from within. Some radionuclides are primordial or are created by the continuous interaction of cosmic rays with the atmosphere, and they are usually referred to as 'naturally occurring[3]'. Others have been produced by artificial means.

1.3. Naturally occurring radionuclides are ubiquitous in the environment, although their activity concentrations vary considerably. Uranium or thorium may be extracted from ores containing relatively high concentrations and, where this is done, the BSS clearly regard the

[1] A practice is defined as any human activity that introduces additional sources of exposure or exposure pathways or extends exposure to additional people or modifies the network of exposure pathways from existing sources, so as to increase the exposure or the likelihood of exposure of people or the number of people exposed.
[2] An intervention is defined as any action intended to reduce or avert exposure or the likelihood of exposure to sources which are not part of a controlled practice or which are out of control as a consequence of an accident.
[3] The term naturally occurring radionuclides is defined as those radionuclides that occur in significant quantities on Earth and usually refer to ^{40}K, ^{235}U, ^{238}U, ^{232}Th and their radioactive decay products.

situations as falling under the requirements for practices. The position regarding ores and other materials with above average concentrations of the naturally occurring radionuclides in the Earth's crust is however undefined.

1.4. Radionuclides of artificial origin are produced and used within practices. As such, the provisions for exemption and clearance given in the BSS apply. In addition, many of these radionuclides are widely spread in the environment as a result of, for instance, fallout from the testing of nuclear weapons in the atmosphere and from routine or accidental releases from past and current practices.

1.5. As a result of the widespread presence of radionuclides in the environment, a certain amount of radioactivity, of natural or artificial origin, is always present in material[4], (including goods, merchandises, consumer products, buildings, soil and, in general, in any *'commodity'*[5]). Not everything that contains radioactivity should therefore need to be regulated. The specification of the radionuclide content in material requiring regulation for purposes of radiation protection is essential for defining the scope of the relevant regulations, the implication being that material containing an amount of radioactivity higher than a prescribed level will require regulation. There are however, different approaches that can be used to determine the scope of application of regulations, very often determined by already established national practices.

OBJECTIVE

1.6. The objective of this Safety Guide is to provide guidance to national authorities, including regulatory bodies, and operating organizations on specific levels of activity concentrations for both naturally occurring radionuclides and those of artificial origin below which regulation of the material for the purposes of radiation protection in accordance with the BSS should not be required, irrespective of the amounts involved. These activity concentrations of radionuclides in material may be derived using different methodological approaches, in the case of naturally occurring radionuclides, from the concept of exclusion, and in the case of radionuclides of artificial origin, from the concepts of exemption and clearance. For this reason, the use of a generic label for these activity concentrations will be

[4] The term material is defined as the matter from which a thing is made, the elements or constitute parts of a substance.
[5] Commodities are any article or raw material. that can be bought or sold.

avoided. Guidance is also provided on how these levels should be applied in a regulatory context.

SCOPE

1.7. The activity concentrations developed in this document apply to all material including those materials with elevated levels as a consequence of technological processing (also see paragraph 3.2. of this Guide).

1.8. The activity concentrations developed in this Guide are a practical application of the concepts of exclusion, exemption and clearance established in the BSS. Exclusion is, by definition, outside the scope of the Standards and this Guide provides quantitative guidance on the provision for this in the BSS. Exemption is from the requirements for practices of the BSS. Clearance is similar to exemption, but specifically relates to the removal of radioactive material within authorized practices from any further control by the regulatory body. Bulk quantities may be involved in clearance and for this reason regulatory bodies may wish to adopt more stringent activity concentration levels than those given in the BSS for exemption. This Guide provides activity concentrations that may be used by regulatory bodies for determining when controls over bulk quantities of material that are part of authorized practices are not required.

1.9. The activity concentrations in this document do not apply to:

- foodstuffs, drinking water, animal feed and any material intended for use in food or animal feed. Specific levels for drinking water are contained in [5] and specific levels for foodstuffs (applicable up to one year after an accident) are found in [6];
- radon, as action levels are provided in the BSS; and
- potassium-40 in the body, which is already excluded from the BSS.

1.10. It is not within the scope of this Safety Guide to calculate activity concentration levels for radionuclide concentrations for foodstuffs and drinking water. However, the activity concentration levels for radionuclides of artificial origin were based on a set of typical exposure scenarios for all material causing external irradiation as well as inhalation and ingestion of radioactive material, including foodstuff and drinking water exposure pathways. It should be emphasized that the ingestion of foodstuffs and drinking water used in the derivation of nuclide-specific levels for all material does not have any relation to activity concentration levels already specified for foodstuffs and drinking water.

1.11. The activity concentration levels in this document are not intended to be applied to the control of radioactive discharges of liquid and airborne effluents from authorized practices, or to radioactive residues in the environment. Guidance on authorization of liquid and airborne effluents discharges and reuse of contaminated land is provided elsewhere [7][8].

STRUCTURE

1.12. The Safety Guide is structured as follows: Section 2 describes the conceptual approach used to derive the activity concentrations based on the concepts given in the BSS. Section 3 presents the basis for deriving the levels, and is supported by a Safety Report XXX [9] describing the methodology used. Section 4 provides the activity concentration levels. Section 5 provides guidance on the application of the activity concentrations to practices and Section 6 provides guidance application to trade. Section 7 describes how to account for mixtures of radionuclides and Section 8 provides guidance on averaging procedures. Section 9 identifies concerns with the intentional dilution of material.

2. THE CONCEPTUAL APPROACH

GENERAL

2.1. In this section, the conceptual approach for establishing a set of activity concentrations above which regulatory control for purposes of radiation protection might be needed and below which regulatory control would not be needed is discussed.

2.2. The International Commission on Radiological Protection (ICRP) has recognized the importance of limiting the scope of the system of radiological protection. The ICRP [3] has stated that "*everyone in the world is exposed to radiation from natural and artificial sources ... any realistic system of radiological protection must therefore have a clearly defined scope if it is not to apply to the whole of mankind activities*".

2.3. The International Basic Safety Standards (BSS) [1] establish the requirements for protection against the risks associated with radiation exposure. The BSS covers both practice and intervention situations and includes the concepts of *exclusion, exemption* and *clearance.* These concepts and the relation between them are briefly presented below.

EXCLUSION

2.4. The BSS glossary defines "excluded" as "*outside the scope of the standards*". The BSS expands on this by stating "*any exposure whose magnitude or likelihood is essentially unamenable to control through the requirements of the Standards is deemed to be excluded from the Standards.*" [1, para. 1.4.]

2.5. Examples of excluded exposures given in the BSS are: exposure from ^{40}K in the body, from cosmic radiation at the surface of the earth and from unmodified concentrations of radionuclides in most raw material. All of these examples involve natural sources of radiation although there is no explicit requirement to limit the concept to such sources of exposure. In particular, regulatory bodies may wish to apply it to exposures from artificial radionuclides that are now widespread in the environment due to past practices and accidents.

EXEMPTION

2.6. The BSS use the concept of exemption only within the context of practices. Exemption determines *a priori* which practices, sources and radioactive material may be freed from the requirements for practices, and hence regulatory control, based on meeting certain criteria. In essence, it can be considered as a generic authorization for practices granted by the regulatory body, which, once issued, releases persons from the requirements that would otherwise apply.

2.7. Exemption should be granted if the regulatory body is satisfied that the practices or sources within practices meet the exemption criteria or the exemption levels specified in Schedule I of the BSS, or other exemption levels specified by the regulatory body on the basis of the exemption criteria. Exemption should not be granted to permit practices that would otherwise not be justified. The grounds for exemption are that the source gives rise to small (trivial) individual effective doses (of the order of 10 μSv or less in a year) and the protection is optimized, i.e. regulatory provisions will produce little or no improvement in dose reduction (this is indicated by a collective effective dose committed by one year of performance of the practice of no more than about 1 man·Sv, or an assessment for the optimization of protection that shows that exemption is the optimum option).

2.8. The levels in Schedule I of the BSS were derived by establishing a set of exposure scenarios and using them to derive activity concentrations and total quantities of radionuclides that correspond to the dose criteria for exemption of practices. These derived radionuclide-specific levels are based on moderate quantities of material. Their use allows exemption from the requirements of the BSS if the criteria are met, except that the practice should be justified, i.e. exemption should not be invoked to allow frivolous or unwarranted usage of radionuclides. A footnote to Schedule I of the BSS indicates that exemption for bulk amounts of materials with activity concentrations lower than the guidance exemption levels given in that Schedule may require further consideration by the regulatory body.

CLEARANCE

2.9. The BSS also use the concept of *clearance* only within the context of practices. While exemption is used as a part of a process to determine the nature and extent of application of the system of radiation protection and regulatory control, *clearance* is intended to establish which sources under regulatory control can be removed from this control. Like exemption, it can be considered as a generic authorization, granted by the Regulatory Body, for that component of a practice that applies to the release of radioactive material.

2.10. Clearance is defined in the BSS glossary as the "*removal of radioactive materials or radioactive objects within authorized practices from any further control by the Regulatory Authority*". Furthermore, the BSS state that clearance is subject to clearance levels that are defined as "*values, established by the Regulatory Authority and expressed in terms of activity concentrations and/or total activity, at or below which sources of radiation may be released from regulatory control*". A footnote indicates that clearance of bulk amounts of material with activity concentrations lower than the guidance exemption levels specified in Schedule I of the BSS may require further consideration by the regulatory body.

3. BASIS FOR DERIVING ACTIVITY CONCENTRATION LEVELS

3.1. Two approaches are used to establish activity concentrations. The first applies broadly to radionuclides of natural origin; the second applies broadly to radionuclides of artificial origin. These approaches are intended to be consistent with the underlying approach given in the BSS whereby exclusion relates primarily to the former, while exemption and clearance relate primarily to the latter, although in neither case are these relationships exclusive. For instance, exposures from some radionuclides of artificial origin should be implicitly or explicitly excluded from regulatory control, such as fallout from the atmospheric testing of nuclear weapons. Similarly, some material contaminated by radionuclides of natural origin, if within a practice, should be a candidate for exemption or clearance, as appropriate. A full discussion of the methodological approaches used is given in the supporting Safety Report [9].

RADIONUCLIDES OF NATURAL ORIGIN

3.2. Exclusion, as described in the BSS, relates to the amenability of exposures to regulatory control rather than to the actual magnitude of those exposures. Amenability to control is a relative concept; it is a matter of reasonableness and implies recognition of the cost of exercising regulatory control and the benefit to be gained by so doing. The examples of excluded exposures given in the BSS include those from "*unmodified concentrations of radionuclides in most raw materials*" [1, footnote 2]. The reference to unmodified concentrations points to the fact that processing some raw material, which may have typical concentrations of radionuclides of natural origin, may lead to products or waste that have higher values or give rise to exposures that should not be excluded from regulatory control. The reference to exposure from most raw material suggests that exposure from some raw material themselves should not be subject to exclusion. Thus, whatever the cause of the exposure - through enhancement of the radionuclide content during processing or simply because the material has an intrinsically relatively high radionuclide content - the regulatory body should recognize that there are some industries handling or using naturally occurring radioactive material where attributable exposures warrant consideration and control. This Guide therefore provides quantitative guidance on the phrase "*unmodified concentrations of radionuclides in most raw materials*". It is noted that some consideration of the occupational exposures that might result from such material has already been given in another Safety Guide [10].

3.3. The activity concentrations for naturally occurring radionuclides have been selected from a consideration of the worldwide distribution of the activity concentrations given by UNSCEAR [2]. Doses to individuals as a consequence of the use of these activity concentrations are unlikely to exceed about 1 mSv in a year, excluding the contribution from the emanation of radon, which is dealt with separately in the BSS. Situations involving the contamination of the water pathway may require a case-by-case evaluation of possible doses. In this context, it is noted that WHO has issued guidelines for drinking water, which include levels for naturally occurring radionuclides [5].

RADIONUCLIDES OF ARTIFICIAL ORIGIN

3.4. The primary radiological basis for establishing levels for exemption and clearance of bulk amounts of material is that effective doses to individuals should be of the order of 10μSv in a year or less. In order to avoid treating this dose as a limit, which would necessitate the use of extremely cautious models, an additional criterion was used which is that there should be a low probability of the effective dose to any individual approaching 1mSv in any particular year. Consideration was also given to doses to the skin; an equivalent dose criterion of 50 mSv in a year was used for this purpose.

3.5. Many studies undertaken at national or international levels have derived radionuclide specific levels for the clearance of solid material [11 - 13]. The results presented in this document draw upon the extensive experience gained in undertaking these studies and independent calculations performed under the auspices of the Agency [9]. The calculations are based on the evaluation of a selected set of typical exposure scenarios for all material encompassing external irradiation, dust inhalation and ingestion (direct and indirect). As stated in paragraph 1.10, foodstuffs and drinking water pathways were taken into account to address the radiological consequences as appropriate. The selected levels were the lowest values obtained from the scenarios.

3.6. For a number of short-lived radionuclides, the calculations lead to levels that are higher than the exemption levels given in the BSS. This is due to the fact that the scenarios focus on the transport, trade, use, or deposition of materials outside the facilities in which they arise (i.e., reactors, accelerators, laboratories), and account was taken of the lapse of time involved before the start of the exposure. The models on which the exemption levels are based, consider the direct handling of the material within these facilities and consequently do not allow for any radioactive decay of the radionuclides before the exposure starts. For these radionuclides, the levels chosen were the exemption levels of the BSS.

4. ACTIVITY CONCENTRATION LEVELS FOR MATERIAL

4.1 The activity concentrations for radionuclides of natural origin derived using the first approach discussed in paragraphs 3.2 and 3.3 are given in Table I.

TABLE I. ACTIVITY CONCENTRATION LEVELS FOR RADIONUCLIDES OF NATURAL ORIGIN

Radionuclide	Concentration Level (Bq/g)
Radionuclides in the ^{235}U decay series	0.05
^{40}K	5
All other naturally occuring radionuclides	0.5

These levels have been determined on the basis of the worldwide distribution of radioactivity concentrations for these radionuclides. Consequently, they are valid for the natural decay chains in secular equilibrium, i.e., ^{238}U, ^{235}U and ^{232}Th, with the value given being applied to the parent of the decay chain. The values can also be used individually for each decay product in the chains or the head of subsets of the chains, such as ^{226}Ra.

4.2. Those for radionuclides of artificial origin derived using the second approach discussed paragraphs 3.4 to 3.6 are given in Table II.

4.3 The details of the calculations that led to these values are contained in Safety Report XXX [9].

TABLE II. ACTIVITY CONCENTRATION LEVELS FOR RADIONUCLIDES OF ARTIFICIAL ORIGIN

Radionuclide	Concentration Level (Bq/g)		Radionuclide	Concentration Level (Bq/g)		Radionuclide	Concentration Level (Bq/g)	
H-3	100		As-73	1000		Cd-109	1	
Be-7	10		As-74	10	*	Cd-115	10	
C-14	1		As-76	10	*	Cd-115m	100	
F-18	10	*	As-77	1000		In-111	10	
Na-22	0.1		Se-75	1		In-113m	100	*
Na-24	1	*	Br-82	1		In-114m	10	
Si-31	1000	*	Rb-86	100		In-115m	100	*
P-32	1000		Sr-85	1		Sn-113	1	
P-33	1000		Sr-85m	100	*	Sn-125	10	
S-35	100		Sr-87m	100	*	Sb-122	10	
Cl-36	1		Sr-89	1000		Sb-124	1	
Cl-38	10	*	Sr-90	1		Sb-125	0.1	
K-42	100		Sr-91	10	*	Te-123m	1	
K-43	10	*	Sr-92	10	*	Te-125m	1000	
Ca-45	100		Y-90	1000		Te-127	1000	
Ca-47	10		Y-91	100		Te-127m	10	
Sc-46	0.1		Y-91m	100	*	Te-129	100	*
Sc-47	100		Y-92	100	*	Te-129m	100	
Sc-48	1		Y-93	100	*	Te-131	100	*
V-48	1		Zr-93	10	*	Te-131m	10	
Cr-51	100		Zr-95	1		Te-132	1	
Mn-51	10	*	Zr-97	10	*	Te-133	10	*
Mn-52	1		Nb-93m	10		Te-133m	10	*
Mn-52m	10	*	Nb-94	0.1		Te-134	10	*
Mn-53	100		Nb-95	10		I-123	10	
Mn-54	0.1		Nb-97	10	*	I-125	1000	
Mn-56	10	*	Nb-98	10	*	I-126	10	
Fe-52	10	*	Mo-90	10	*	I-129	0.1	
Fe-55	1000		Mo-93	10		I-130	10	*
Fe-59	1		Mo-99	10		I-131	10	
Co-55	10	*	Mo-101	10	*	I-132	10	*
Co-56	0.1		Tc-96	1		I-133	10	*
Co-57	1		Tc-96m	1000	*	I-134	10	*
Co-58	1		Tc-97	10		I-135	10	*
Co-58m	10000	*	Tc-97m	100		Cs-129	10	
Co-60	0.1		Tc-99	1		Cs-131	1000	
Co-60m	1000	*	Tc-99m	100	*	Cs-132	10	
Co-61	100	*	Ru-97	10		Cs-134	0.1	
Co-62m	10	*	Ru-103	10		Cs-134m	10	*
Ni-59	100		Ru-105	10	*	Cs-135	100	
Ni-63	100		Ru-106	0.1		Cs-136	1	
Ni-65	10	*	Rh-103m	10000	*	Cs-137	0.1	
Cu-64	100	*	Rh-105	100		Cs-138	10	*
Zn-65	0.1		Pd-103	1000		Ba-131	10	
Zn-69	1000	*	Pd-109	100		Ba-140	1	
Zn-69m	10	*	Ag-105	10		La-140	1	
Ga-72	10	*	Ag-110m	0.1		Ce-139	1	
Ge-71	10000		Ag-111	100		Ce-141	100	

Radionuclide	Concentration Level (Bq/g)	
Ce-143	10	
Ce-144	10	
Pr-142	100	*
Pr-143	1000	
Nd-147	100	
Nd-149	100	*
Pm-147	1000	
Pm-149	1000	
Sm-151	10000	
Sm-153	100	
Eu-152	0.1	
Eu-152m	100	*
Eu-154	0.1	
Eu-155	1	
Gd-153	10	
Gd-159	100	*
Tb-160	1	
Dy-165	1000	*
Dy-166	100	
Ho-166	100	
Er-169	1000	
Er-171	100	*
Tm-170	100	
Tm-171	1000	
Yb-175	100	
Lu-177	100	
Hf-181	10	
Ta-182	0.1	
W-181	10	
W-185	1000	
W-187	101000	
Re-186	1000	
Re-188	100	*
Os-185	1	
Os-191	100	
Os-191m	1000	*
Os-193	100	
Ir-190	1	
Ir-192	1	

* indicates half life less than 1 day

Radionuclide	Concentration Level (Bq/g)	
Ir-194	100	*
Pt-191	10	
Pt-193m	1000	
Pt-197	1000	*
Pt-197m	100	*
Au-198	10	
Au-199	100	
Hg-197	100	
Hg-197m	100	
Hg-203	10	
Tl-200	10	
Tl-201	100	
Tl-202	10	
Tl-204	1	
Pb-203	10	
Bi-206	1	
Bi-207	0.1	
Po-203	10	*
Po-205	10	*
Po-207	10	*
At-211	1000	
Ra-225	10	
Ra-227	100	
Th-226	1000	
Th-229	0.1	
Pa-230	10	
Pa-233	10	
U-230	10	
U-231	100	
U-232	0.1	
U-233	10	
U-236	10	
U-237	100	
U-239	100	*
U-240	100	*
Np-237	1	
Np-239	100	
Np-240	10	*
Pu-234	100	*

Radionuclide	Concentration Level (Bq/g)	
Pu-235	100	*
Pu-236	1	
Pu-237	100	
Pu-238	1	
Pu-239	1	
Pu-240	1	
Pu-241	100	
Pu-242	1	
Pu-243	1000	*
Pu-244	0.1	
Am-241	1	
Am-242	1000	*
Am-242m	1	
Am-243	1	
Cm-242	10	
Cm-243	1	
Cm-244	10	
Cm-245	1	
Cm-246	1	
Cm-247	0.1	
Cm-248	1	
Bk-249	100	
Cf-246	1000	
Cf-248	10	
Cf-249	0.1	
Cf-250	1	
Cf-251	1	
Cf-252	10	
Cf-253	100	
Cf-254	1	
Es-253	100	
Es-254	0.1	
Es-254m	10	
Fm-254	10000	*
Fm-255	100	*

5. APPLICATION OF ACTIVITY CONCENTRATIONS TO PRACTICES

RADIONUCLIDES OF NATURAL ORIGIN

5.1. In this document, the concept of exclusion has been used in the context of exposures from natural sources of radiation. The activity concentrations of such radionuclides in material should be used to define that material that should be outside or inside, as the case may be, the scope of regulatory control. If the activity concentration is below the activity concentration values in Table I for the radionuclides in question, then the handling and use of the material should be regarded as outside the scope of regulatory control. If the activity concentration is above the activity concentration levels in Table II, the regulatory body should decide on the extent to which the regulatory requirements for practices set out in the BSS [1] should be applied.

5.2. In addition, the activity concentrations of radionuclides of natural origin should be used to determine when material within a practice can be released from regulatory control. This applies irrespective of the amount of material involved.

5.3. The way in which these levels should be incorporated into national regulatory requirements will depend on the particular approach to be adopted. One approach might be to use these levels in the actual definition of the scope of the regulations. Another might be to use the levels to define radioactive material for the purposes of the regulations.

RADIONUCLIDES OF ARTIFICIAL ORIGIN

5.4. In this document, the concepts of exemption and clearance have been applied to bulk amounts of material containing radionuclides of artificial origin. These concepts relate specifically to practices that are considered by the regulatory body to be justified[3].

5.5. The BSS, in Schedule I, indicate that radioactive substances from an authorized practice or source whose release to the environment has been authorized, are exempted from any new requirements of notification, registration or licensing unless specified by the

[3] It should be noted that the justification principle applies to practices as a whole and not separately to its component parts such as the disposal of waste. Thus the means whereby material that is contaminated as a consequence of a practice is disposed is a matter of optimization of protection, rather than justification. One of the purposes for which the activity concentrations have been established is to permit material, in bulk quantities, to be 'exempted' or 'cleared' from a practice without further consideration..

regulatory body. Since exemption and clearance are in essence generic authorizations, this provision of the BSS means that 'exempted' or 'cleared' material should be allowed to be used without any further restriction, that is, material that has been exempted or cleared should not re-enter the system of protection for practices, unless specifically required to do so by the Regulatory Body. In particular, any subsequent use of the material within a practice should not necessitate application of the principle of justification from a radiation protection point of view.

5.6. The way in which these levels should be incorporated into national regulatory requirements will depend on the particular approach to be adopted. Any of the approaches proposed in paragraph 5.3 for the naturally occurring radionuclides might be used. However, it is noted that many regulatory bodies have adopted the activity concentrations given in Schedule I of the BSS into their national requirements. Where that is the case, one possibility would be to express the levels in a specific regulatory instrument in which the requirements relating to exemption and clearance of bulk amounts of material are given.

6. APPLICATION OF ACTIVITY CONCENTRATIONS TO TRADE

6.1. The terms 'practice' and 'intervention' were intended to assist regulatory and national bodies in determining those situations that should be under some form of control, particularly regulatory control. If the activity concentrations developed in this Guide are used as indicated in the previous section, then there should be no need to give attention to intervention. In particular, national and international trade in commodities containing radionuclides with activity concentrations below the activity concentrations values in Table I and II should not be subject to regulatory controls.

6.2. Compliance with the activity concentrations given in this Guide should be verified at the first point of entry into trade. Thus, authorities in exporting countries should ensure that systems are in place to prevent unrestricted trade in material containing higher levels of radioactivity. Authorities in importing countries should ensure that any monitoring that is undertaken at borders and elsewhere, such as scrap recycling plants, to detect for the presence of 'orphan sources'[4] should take account of the activity concentrations given in this Safety Guide in order to prevent unnecessary restrictions on the movement and use of material. In general, however, it should not be necessary for each country to set up its own routine measurement programme solely for the purpose of monitoring commodities, particularly if there is confidence in the controls exercised by the producing country.

6.3. In cases where there are reasonable grounds for believing that the activity concentrations may be exceeded, arrangements should be made to determine the actual levels either by obtaining the information from the supplier or by measurement. In such cases, measurements should be performed using appropriate techniques and equipment capable of measurement of activity concentrations at the specified levels. In general, countries should coordinate their regulatory strategies and implementation with neighboring countries, including monitoring programmes for commodities, in order to avoid unnecessary hindrance to trade at boundary transfer points.

[4] Orphan source means a source which poses sufficient radiological hazard to warrant regulatory control but is not under regulatory control, either because it has never been under regulatory control, or because it has been abandoned, lost, misplaced, stolen or transferred without proper authorization.

7. MIXTURES OF RADIONUCLIDES

7.1. For material containing a mixture of radionuclides, either of natural or artificial origin, comparison with the activity concentration levels should be undertaken as indicated by the formulas below:

for each naturally occurring radionuclide

$$\frac{C_{natural}}{Activity\ concentration} \leq 1$$

where $C_{natural}$ is the concentration (Bq/g) of the naturally occurring radionuclide in the material or, for those radioactive decay chains in secular equilibrium, it is the concentration of the parent radionuclide, and the activity concentration is the level specified for the relevant naturally occurring radionuclide (or for those in secular equilibrium, the parent nuclide);

for radionuclides of artificial origin

$$\sum_{i=1}^{n} \frac{C_{i(artificial)}}{Activity\ concentration_i} \leq 1$$

where $C_{i(artificial)}$ is the concentration (Bq/g) of the i^{th} radionuclide of artificial origin in the material, *Activity concentration*$_i$ is the level for that radionuclide in the material and n is the number of radionuclides present.

7.2. If both (1) and (2) are satisfied and are less than or equal to 1, then the material should not be subject to regulatory control. If the result of either equation is greater than one, the requirements of the BSS [1] should be applied to the material. This type of relationship should be used by the regulatory bodies in their specific guidance on application of the BSS [1] to account for situations where multiple radionuclides are present in mixtures.

8. AVERAGING

8.1. When applying the above activity concentrations, the regulatory body should consider methodologies for sampling, averaging, monitoring, and detection of radionuclides. In doing this, the regulatory body should recognize that these activity concentrations were derived for large quantities and therefore the averaging should be done accordingly.

Consideration should also be given to surface contamination levels that would equate to the specified dose criteria. The Agency is currently preparing guidance on these issues.

9. DILUTION OF MATERIAL

9.1 Deliberate dilution of material, as opposed to dilution that takes place in normal operations when radioactivity is not a consideration, in order to meet the activity concentration levels given in section 4 should not be permitted without the prior approval of the regulatory body.

10. SUMMARY

10.1. A summary of extracts of relevant parts from the BSS [1], from the Safety Guide "Occupational Radiation Protection" [11] are given in the Annex along with a summary of the overall guidance on exclusion, exemption and clearance for material.

REFERENCES

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ANNEX – SCOPE OF REGULATORY CONTROL FOR PRACTICES

Practices are defined in the Glossary of the BSS as follows:

> *Any human activity that introduces additional sources of exposure or exposure pathways or extends exposure to additional people or modifies the network of exposure pathways from existing sources, so as to increase the exposure or the likelihood of exposure of people or the number of people exposed.*

This definition does not unequivocally indicate which sources of exposure should be included and which should be excluded. However, the BSS provides the following statements, which provide further clarification:

201. The practices to which the Standards shall apply include:

- *(a) the production of sources and the use of radiation or radioactive substances for medical, industrial, veterinary or agricultural purposes, or for education, training or research, including any activities related to that use which involve or could involve exposure to radiation or radioactive substances;*
- *(b) the generation of nuclear power, including any activities in the nuclear fuel cycle which involve or could involve exposure to radiation or radioactive substances;*
- *(c) practices involving exposure to natural sources specified by the Regulatory Authority as requiring control; and*
- *(d) any other practice specified by the Regulatory Authority.*

202. The sources within any practice to which the requirements for practices of the Standards shall apply include:

- *(a) radioactive substances and devices that contain radioactive substances or produce radiation, including consumer products, sealed sources, unsealed sources, and radiation generators, including mobile radiography equipment;*
- *(b) installations and facilities which contain radioactive substances or devices which produce radiation, including irradiation installations, mines and mills processing radioactive ores, installations processing radioactive substances, nuclear installations, and radioactive waste management facilities; and*
- *(c) any other source specified by the Regulatory Authority.*

204. The exposures to which the requirements of the Standards apply are any occupational exposure, medical exposure or public exposure due to any relevant practice or source within the practice, including both normal exposures and potential exposures.

205. Exposure to natural sources shall normally be considered as a chronic exposure situation and, if necessary, shall be subject to the requirements for intervention, except that[5]*:*

- *(a) public exposure delivered by effluent discharges or the disposal of radioactive*

[5] At the time of the endorsement of the Standards, the available quantitative recommendations of the ICRP for protection against exposure to natural sources were confined to radon. It was therefore decided that the General Obligations for practices concerning protection against natural sources will be that exposure to natural sources, which is normally a chronic exposure situation, should be subject to intervention and that the requirements for practices should be generally limited to exposure to radon, the exposure to other natural sources being expected to be dealt with by exclusion or exemption of the source or otherwise at the discretion of the Regulatory Authority.

waste arising from a practice involving natural sources shall be subject to the requirements for practices given in the BSS, unless the exposure is excluded or the practice or the source is exempted; and

(b) *occupational exposure of workers to natural sources shall be subject to the requirements for practices given in this section if these sources lead to:*

   (i) *exposure to radon required by or directly related to their work, irrespective of whether the exposure is higher or lower than the action level for remedial action relating to chronic exposure situations involving radon in workplaces*[6]*, unless the exposure is excluded or the practice or the source is exempted; or*

   (ii) *exposure to radon incidental to their work, but the exposure is higher than the action level for remedial action relating to chronic exposure situations involving radon in workplaces; unless the exposure is excluded or the practice or the source is exempted: or*

   (iii) *exposure specified by the Regulatory Authority to be subject to such requirements.*

The Safety Guide on Occupational Radiation Protection [11] elaborates these requirements as follows:

2.20. The term 'radioactive substance' is not specifically defined in the BSS; it should be noted in particular that the term is not qualified by reference to artificial radionuclides only. Thus, the BSS are intended to apply to naturally occurring radionuclides that have been extracted from ones, irrespective of the use to which those radionuclides are put. Sealed and unsealed sources containing naturally occurring radionuclides such as radium-226 should therefore be treated as being within a practice.

2.20. From para. 2.5(b)(i) of the BSS, it is clear that the mining and milling of radioactive ores should be treated as practices. All exposures in these situations including those from radon, should be subject to the requirements for practices, irrespective of whether the concentrations of radon in air are above the action level specified in the BSS.

2.20. Paragraph 2.5(b)(ii) of the BSS should be taken to mean that exposures to radon in workplaces other than those covered in para. 2.5(b)(i) should be subject to the requirements for occupational exposure if the radon concentration exceeds the action level. This does not, however, apply if the exposure has been excluded or the practice or source has been exempted. Examples of workplaces where exposure to radon is adventious and the levels are likely to exceed the action level include mines (other than those intended to produce radioactive ores), spas and aboveground workplaces in radon prone areas.

2.20. Action levels apply to chronic exposure situations, which are described in Appendix VI of the BSS. The primary purpose of an action level is to define the circumstances under which remedial or protective action should be undertaken. In the case of adventious exposure to radon, the procedure should be for the regulatory authority to identify or determine, by means of a survey or otherwise, those workplaces with radon concentrations above the action level. Consideration should then be given to whether the concentrations can reasonably be reduced below the action level. Where sufficient reduction in concentrations cannot be achieved, the requirements for practices should be applied. Thus, at this stage the numerical value of the action level has a conceptually different significance than that initially given to it. It is no

[6] See BSS Schedule VI, Guidelines for Action Levels in Chronic Exposure Situations, para. VI-3.

longer to be used as the basis for a decision on intervention, but as the basis for a decision to consider the exposures to be arising from a practice.

On the basis of this guidance and the additional guidance given in this Guide (DS-161), the following are regarded as coming under the requirements for practices in the BSS, subject to the provision for exemption:

- All devices containing radioactive sources;
- All mines producing radioactive ores;
- All workplaces with radon concentrations above the action level;
- All workplaces where the activity concentrations of naturally occurring radionuclides in material that is being handled or used are above the levels specified in body of this Guide (DS-161).

The provision for exclusion in the BSS is as follows:

104. Any exposure whose magnitude or likelihood is essentially unamenable to control through the requirements of the Standards is deemed to be excluded from the Standards[7].

Schedule I of the BSS contains the following provision relating to authorized practices or sources.

> *(I.6) Radioactive substances from an authorized practice or source whose release to the environment has been authorized, are exempted from any new requirements of notification, registration or licensing unless otherwise specified by the Regulatory Authority.*

This provision also applies to exemption and clearance, which are effectively generic authorizations.

On the basis of this provision and the additional guidance given in the body of this Guide (DS-161), the following should be excluded from the regulatory requirements for practices:

- exposure from ^{40}K in the body;
- exposure from cosmic radiation at the surface of the earth;
- exposure from radon below the action level;
- exposure from materials containing activity concentrations of the naturally occurring radionuclides below the levels for given in the body of this Guide (DS-161); and
- exposure from materials containing activity concentrations of the radionuclides of artificial origin below the levels for given in the body of this Guide (DS-161).

[7] Examples are exposure from ^{40}K in the body, from cosmic radiation at the surface of the earth and from unmodified concentrations of radionuclides in most raw materials.

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Derivation of Activity Concentration Levels for Material not Requiring Regulation

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ATOMIC ENERGY AGENCY
VIENNA

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CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

Regulatory systems for radiation protection are intended to ensure the protection of people from harm arising from exposure to ionizing radiation. However, there are some human activities involving exposure to radiation that do not warrant regulatory control. Such circumstances arise when the resources that would need to be expended in regulating the activity would be excessive in relation to any benefit that might ensue in terms of reduced risk. The scope of legal instruments for regulatory control should therefore be defined so as to include only activities for which regulation is warranted. This Safety Report supports the Safety Guide on *Radioactivity in Material not requiring Regulation for Purposes of Radiation Protection* [1].

This document deals with all material[1] to include commodities[2] for which regulatory control in accordance with the *International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources (the BSS)* [2] should be applied. It includes the removal of control of material containing very low levels of radioactivity originating from regulated practices[3], i.e., industrial installations (nuclear fuel cycle and others), hospitals, and research institutes, and of material from interventions[4]. It also addresses naturally occurring radioactive material (NORM) that should be considered for regulation.

The document derives activity concentration[5] levels for deciding if a certain material should come under the regime of the BSS [2]. These levels are derived in such a way that they are valid for all types of material containing radionuclides of artificial or natural origin except foodstuffs and drinking water. Because the levels are applicable to a whole range of material, they have been derived on the basis of several scenarios and assumptions:

(a) The main basis for the derivation of the activity concentration levels for artificial radionuclides is a set of radiological scenarios referring to external irradiation, dust inhalation and ingestion (direct and secondary ingestion) which are deemed to encompass all typical exposure situations for all material types except NORM. Those scenarios relate the activity concentration in the material to individual doses. The scenarios are determined by taking existing radiological studies (e.g., those used for deriving clearance and exemption levels) and using them to develop a framework of generalized scenarios. The approach to envelop the worldwide variety of situations that may be found in Member States necessarily requires a degree of conservatism. In order to cover various exposure scenarios, more than one scenario has been considered for each pathway to reflect the range of material characteristics and exposed individuals. Each scenario therefore contains a set of parameter values and represents a range of exposure situations.

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1 The term *material* is defined as the matter from which a thing is made, the elements or constitute parts of a substance.

2 The term *commodity* is any article or raw or material that can be bought or sold.

3 A *practice* is defined as any human activity that introduces additional sources of exposure or exposure pathways or extends exposure to additional people or modifies the network of exposure pathways from existing sources, so as to increase the exposure or the likelihood of exposure to people or the number of people exposed.

4 An *intervention* is defined as any action intended to reduce or avert exposure or the likelihood of exposure to sources which are not part of a controlled practice or which are out of control as a consequence of an accident.

5 *Activity concentration* is the amount of a radionuclide per unit mass or volume of a material.

A scenario-based approach was not used in the case of naturally occurring radioactive material (NORM). Instead the activity concentration levels applicable to NORM were derived using a pragmatic approach that places greater emphasis on optimization of protection. This involved consideration of the worldwide distribution of the concentration of naturally occurring radionuclides in environmental material.

2. RADIOLOGICAL BASIS FOR ACTIVITY CONCENTRATION LEVELS

For each artificial radionuclide in material, the activity concentration level has been determined such that individual effective doses to the public and workers[6] would be on the order of 10 μSv/a and having only a very low probability of approaching an individual dose of 1 mSv/a. A dose of 10 μSv/a corresponds to a trivial level of risk [2].

While no activity concentration levels have been derived in this Safety Report for foodstuff and drinking water, the water and food pathways have been taken into account in the scenarios for artificial radionuclides to address the radiological consequences from these pathways. Specific levels for foodstuffs have been developed by the Codex Alimentarius Commission [3] and for drinking water by the World Health Organization [4].

The calculations of the activity concentration levels for artificial radionuclides are based on the evaluation of a selected set of typical exposure scenarios for all material, encompassing external irradiation, dust inhalation and ingestion (direct and indirect). The resulting activity concentration levels were derived based on these scenarios as the lower value obtained from:

I. The use of realistic parameter values applying an effective dose criterion of 10 μSv/a.

II. The use of low probability parameter values applying an effective dose criterion of 1mSv/a and a skin equivalent dose limit of 50 mSv/a.

The derived results from the scenario calculations are sufficient to ensure an adequate degree of protection in both occupational and public exposure situations.

If radionuclide-specific activity concentration values for naturally occurring radionuclides are derived on the basis of the same radiological criteria, the values will, in many cases, be lower than concentrations that occur in many natural environmental material. Thus, many human activities previously unregulated from a radiological standpoint, such as construction of houses from natural building material or even the use of land in many areas, could be subject to regulation. Establishing levels for natural radionuclides that invoke such widespread regulatory consideration, in circumstances where in many cases it is unlikely to achieve any improvement in protection, is not an optimum use of regulatory resources. Therefore, derivation of activity concentration levels for naturally occurring radionuclides is based on a methodology that places greater emphasis on optimization of protection, including regulatory resources.

[6] Worker is taken here to mean those workers who could be inadvertently exposed to ionizing radiation while at work, such as foundry or landfill workers.

The objective in defining naturally occurring radioactive substances that should be regulated is to identify that material of significant radiological risk where regulation can achieve real improvements in protection. At the same time, the number of materials involved should not be so great as to make regulation essentially unmanageable. The application of a dose criterion of 10 μSv/a is not practical for NORM. In selecting levels for material that contains NORM, a major issue is that high levels that would exclude the majority of natural material in the environment would also allow a number of situations such as release of phosphate slags to be excluded without further considerations. Conversely, selecting a low value would trigger an unnecessary application of the BSS. Therefore, the activity concentration levels were derived from consideration of the worldwide distribution of concentrations of naturally occurring radionuclides from an independent source (8).

Activity concentration levels for naturally occurring radionuclides are the total of the background and any added radioactivity. Doses to individuals as a consequence of the use of these levels are unlikely to exceed about 1mSv in a year, excluding the emanation of radon and in situations of large volumes contaminating water pathways. This situation could require case-by-case evaluation of possible doses.

3. GENERAL APPROACH FOR DERIVING ACTIVITY CONCENTRATION LEVELS

3.1. CHOICE OF RADIONUCLIDES AND DOSE COEFFICIENTS

The radionuclides for which activity concentration levels are calculated are those for which exemption levels exist in the BSS [2]. This set contains those nuclides that are most relevant to nuclear installations like nuclear power plants or fuel cycle facilities and the application of radionuclides in research, industry and medicine, including short-lived nuclides. A number of additional radionuclides are also considered because of their practical relevance in some cases (e.g., ^{41}Ca, ^{79}Se). Radionuclides of natural origin (^{40}K and the decay chains of ^{238}U, ^{235}U, ^{232}Th) are also included.

A number of radionuclides that are considered in this document decay into unstable short-lived radionuclides. The way in which decay products are treated is discussed in section 3.2. of this document.

In general, dose coefficients are used to calculate (annual) doses from a given activity. More specifically, dose coefficients are used for the following exposure pathways:

- External exposure: The dose from external irradiation is caused by photons from gamma emitting radionuclides absorbed by the human body. Therefore, the relationship between dose and radioactivity is complicated, depending not only on the radionuclide, but also on the geometry in which the radioactivity is distributed, on shielding effects, on self-absorption effects and on the distance and direction to the source. Dose coefficients for external irradiation are expressed as dose rate (μSv/h) per activity content of the source (Bq/g). For this Report, suitable dose coefficients are calculated for each radionuclide and each exposure geometry. These dose coefficients are presented in Appendix II, Table II-III.

  The exposure scenarios consider adults and children of an age between one and two years, which are the most critical age groups for external exposure. A correction of the dose coefficients calculated for adults is required for children to take account of the higher effective dose as compared to adults in the same exposure situations (i.e., for the same air kerma). The factor applied is estimated from Figure 12 in [5], comparing the effective dose per unit air kerma for different age groups in an isotropic irradiation geometry. For the relevant range of photon energies above 100 keV, the ratio between children of 1 year of age and adults is about 1.2. This factor is being used in the scenario calculations for children.

- Inhalation exposure: Dose coefficients for inhalation are contained in Appendix II, Table II-IV. The dose coefficients relate the individual effective dose (in Sv) to the inhaled quantity of radioactivity (in Bq).

- Ingestion exposure: Dose coefficients for ingestion are also contained in Appendix II, Table II-V. The dose coefficients relate the individual effective dose (in Sv) to the ingested quantity of radioactivity (in Bq).

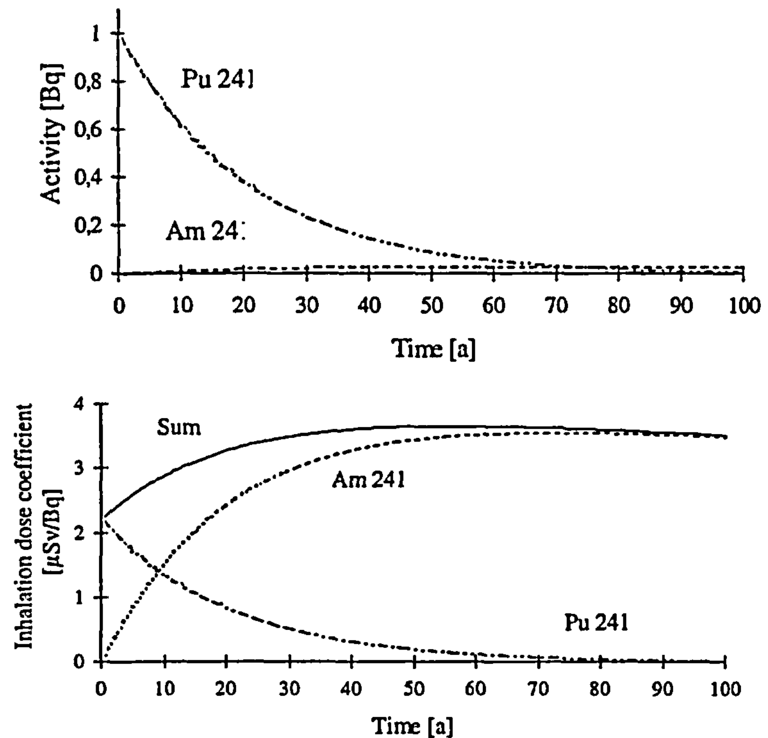
- Skin exposure: Dose coefficients for the skin relate the skin equivalent dose to the concentration of radionuclides on the skin. Skin dose coefficients are listed in [6] and are taken conservatively for a skin surface weight of 4 mg/cm^2. These dose coefficients are contained in Appendix II, Table II-VI.

3.2. DECAY CHAINS AND PROGENY INGROWTH

For radionuclides possessing daughter radionuclides that have a non-negligible dose coefficient in comparison to the parent radionuclides, dose coefficients are calculated as the weighted sum of parent and daughter radionuclides. Weighting is done by using the activity ratios given in Appendix I for the daughter radionuclides indicated. This ensures that the effect of the daughter radionuclides is properly accounted for in the dose calculations.

A number of the radionuclides considered in this document decay into unstable short-lived radionuclides. These daughter radionuclides also contribute to the dose caused by the parent radionuclide after release from regulatory control. For daughter radionuclides with short half-lives, an equilibrium situation with the parent nuclides is reached in a very short time, like for the pair $^{137}Cs/^{137m}Ba$ within 30 minutes or for the pair $^{90}Sr/^{90}Y$ within 20 days. However, there are some important daughter radionuclides with longer half-lives, which yield a high dose contribution, like $^{241}Pu/^{241}Am$. In Fig 1 (a) the activity as a function of time is shown for an initial quantity of 1 Bq of ^{241}Pu. The activity maximum of the daughter radionuclide ^{241}Am occurs at about 70 years at which time the total activity represents only a fraction of the initial activity. In Fig 1 (b) the inhalation dose coefficient is plotted for material in which the initial activity of ^{241}Pu is 1 Bq. In contrast to the activity, the dose coefficient increases over time reaching a maximum at around 60 years although at this time the total activity has decreased to less than 0.1 Bq. This demonstrates that if material containing those radionuclides remains together for a prolonged period of time, the scenarios occurring many years after being released from regulatory control can lead to higher doses than those calculated for the first year after its release due to the ingrowth of daughter radionuclides. Therefore, the relevant progeny is accounted for in the calculations.

(a)



(b)

FIG. 1. Development of activity and dose coefficient of the radionuclide pair $^{241}Pu/^{241}Am$ over time.

The dose contribution from daughter radionuclides is included in the calculations in order not to underestimate doses. This is ensured by adding the dose coefficients of the daughter radionuclides to the dose coefficients of the parent radionuclides, using the appropriate weighting factors for the dose coefficients of the daughter radionuclides. The weighting factors for the daughter nuclides are taken as the maximum activity ratio that the respective daughter radionuclides will reach during a time span of 100 years as illustrated in Fig 2 where the point of maximum activity of the daughter radionuclide is marked. A time span of 100 years is necessary to ensure that material, which does not exceed the activity concentrtion levels at a certain time, will also do so at any later point of time within a reasonable time frame.[7]

[7] This approach does not take account of the fact that in situations like the $^{241}Pu/^{241}Am$ example given in Figure 1, the parent nuclide already has decayed to a large extent when the daughter nuclide reaches its activity maximum. Consequently, the dose factor for the mixture of parent and daughter nuclide will be overestimated in such situations (by a factor of about 1.7 in the example). However, an approach avoiding this potential overestimation would be complicated in particular when several daughter nuclides are involved. Therefore, the approach presented is considered appropriate, satisfying the overall goals of the dose assessments presented here not to underestimate doses and to the extent possible use simple and concise models.

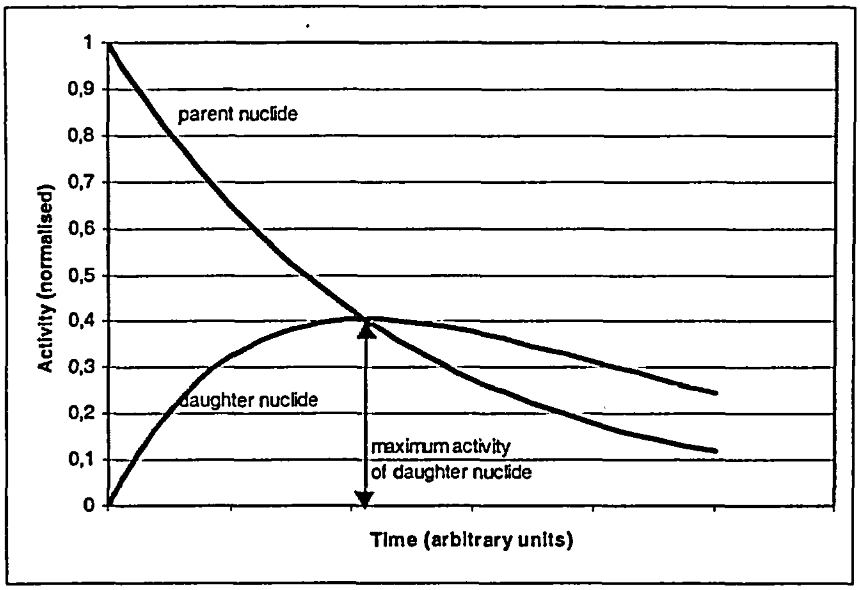


FIG. 2. Activity of arbitrary parent and daughter radionuclide with time. The point of maximum activity of the daughter radionuclide is marked.

The time at which the activity of the first decay product is at a maximum is derived as follows:

If the activity of the progeny as a function of time is designated as $A_2(t)$, then,

$$A_2(t)=A_1(0)\lambda_2\frac{(e^{-\lambda_1 t}-e^{-\lambda_2 t})B_2}{\lambda_2-\lambda_1}$$

$A_2(t)$ = activity of daughter at time t

$A_1(0)$ = initial activity of parent

λ_1 = decay constant of parent

λ_2 = radioactive decay constant

B_2 = branching ratio of daughter

setting the derivative with respect to time to zero

$$\frac{dA_2(t)}{dt}=\frac{A_1(0)\lambda_2}{\lambda_2-\lambda_1}(\lambda_2 e^{-\lambda_2 t}-\lambda_1 e^{-\lambda_1 t})B_2=0$$

solving for t, one obtains

$$t_{\max}=\frac{\log\left(\frac{\lambda_2}{\lambda_1}\right)}{\lambda_2-\lambda_1}$$

$t_{\max}$ = time of maximum

The weighting factors that are calculated in this way are provided in Appendix I of this document.

As the activity concentration levels derived in this document already take into account dose contributions from daughter radionuclides, it is also possible to provide a list of those daughter radionuclides that are fully accounted for in the activity concentration levels of the parent radionuclide. The following set of criteria is convenient in order to define when this is the case for a particular daughter radionuclide:

1. The half-life of the daughter radionuclide must be shorter than that of the parent radionuclide.

   AND

2. The half-life of the daughter radionuclide is less than 1 day OR

3. The half-life of the daughter radionuclide is less than 10% of the half-life of the parent radionuclide AND the half-life of the daughter radionuclide is less than 10 years.

This means that a daughter radionuclide needs not be treated separately if criterion 1 is fulfilled together with at least one of the criteria 2 and 3. Table I provides a list of parent and daughter radionuclides that fulfill the above criteria. For decay chains (i.e., more than one daughter radionuclide), the process of including daughter radionuclides in this way is carried on until a radionuclide is reached which fails to meet the criteria. All daughter radionuclides up to this point are then taken into account in the dose calculations. The parent radionuclides are marked with the sign "+" to indicate that the derived activity concentration level also includes daughter radionuclides. When applying the activity concentration levels, the daughter radionuclides listed in Table I need not be considered separately.

TABLE I. LIST OF DAUGHTER RADIONUCLIDES THAT ARE TAKEN INTO ACCOUNT WITH THE PARENT RADIONUCLIDE

Parent Radionuclide	Daughter Radionuclides						
Fe-52+	Mn-52m						
Zn-69m+	Zn-69						
Sr-90+	Y-90						
Sr-91+	Y-91m						
Zr-95+	Nb-95m						
Zr-97+	Nb-97m	Nb-97					
Nb-97+	Nb-97m						
Mo-99+	Tc-99m						
Mo-101+	Tc-101						
Ru-103+	Rh-103m						
Ru-105+	Rh-105m						
Ru-106+	Rh-106						
Pd-103+	Rh-103m						
Pd-109+	Ag-109m						
Ag-108m+	Ag-108						
Ag-110m+	Ag-110						
Cd-109+	Ag-109m						
Cd-113m	In-113m	Cd-113					
Cd-115+	In-115m						
Cd-115m+	In-115m						
In-114m+	In-114						
Sn-113+	In-113m						
Sn-121m	Sn-121						
Sb-125+	Te-125m						
Te-127m+	Te-127						
Te-129m+	Te-129						
Te-131m+	Te-131						
Te-132+	I-132						
Cs-137+	Ba-137m						
Ce-144+	Pr-144	Pr-144m					
Pm-146	Sm-146						
U-232sec	Th-228	Ra-224	Rn-220	Po-216	Pb-212	Bi-212	Tl-208
U-240+	Np-240m	Np-240					
Np-237+	Pa-233						
Pu-244+	U-240	Np-240m	Np-240				
Am-242m+	Np-238						
Am-243+	Np-239						
Cm-247+	Pu-243						
Es-254+	Bk-250						
Es-254m+	Fm-254						

3.3. ARTIFICIAL RADIONUCLIDES

The sequence of calculations for deriving the activity concentration levels for all material containing artificial radionuclides, except foodstuffs and drinking water, proceeds along the following lines:

- selection of radionuclides for which the calculations are carried out;
- definition of suitable scenarios and parameter values;
- calculation of annual doses relating to the unit specific activity (i.e., 1 Bq/g) for each radionuclide;
- identification of the limiting scenario for each set of calculations, i.e., the one that gives the highest dose;
- derivation of the radionuclide specific activity concentration levels by dividing the reference dose level (10 μSv/a, 1 mSv/a, or 50 mSv/a, as appropriate) by the annual dose calculated for 1 Bq/g for the limiting scenario for that nuclide; and
- application of rounding procedures to the activity concentration levels.

The rounding[8] to powers of ten is similar to the approach followed for the exemption levels. It implies that the radiological models do not possess such a level of accuracy that a higher precision of the result would be justified.

For the artificial radionuclides, several evaluations were considered as described below. The scenarios described in this section serve to determine whether material should fall within the scope of the BSS [2]. They are designed to be applicable to all material types in large quantities. They are not, however, suitable to treat large amounts of NORM that is dealt with in section 3.4.

Examination of a large number of scenarios from around the world revealed that the limiting cases for a significant number of radionuclides could be reduced to a few scenarios. Within these scenarios, different exposure pathways may account for the total exposure. These relevant exposure pathways are summed up for each scenario to yield the total dose.

On a radionuclide by radionuclide basis, the dominant scenario depends on a few parameters, such as exposure time, concentration of the radionuclide used in the exposure pathway(s), timing of the scenario with respect to radioactive decay, etc. Based on these observations from specific and detailed scenarios, the following scenarios are used in the calculation of activity concentration levels:

- Scenario WL

  A worker is exposed from contaminated material dumped on a landfill. Exposure pathways encompass external irradiation from the material, the inhalation of contaminated dust, and the inadvertent ingestion of contaminated material (e.g., via hand-to-mouth pathway).

[8] If the calculated values lie between 3×10^{x} and $3\times10^{x+1}$, the rounded value is 10^{x+1}. This type of near-logarithmic rounding was preferred in order to err by the same factor rather than by a factor 2 upwards and 5 downwards in conventional rounding.

- Scenario WF

A worker in a foundry where contaminated metal is smelted. External exposures arise if the worker stays within the vicinity of piles of contaminated material. In addition, the worker is exposed to dust released from the material during the transport and melting process. This dust can be inhaled and inadvertently ingested.

- Scenario WO

A worker who comes into contact with contaminated material on a regular basis (e.g., a truck driver). He is exposed externally from the material (e.g., from the truckload). This scenario also covers the exposure from a large piece of equipment cleared from regulatory control and is re-used in a workplace.

- Scenarios RL-C and RL-A

This scenario considers individuals living near a landfill or an other facility (RL-C = child, RL-A = adult), being exposed through contaminated dust released at the landfill or facility. In addition, it is assumed that the residents harvest foodstuff in a private garden on ground that has become contaminated through the deposition of contaminated material.

- Scenario RF

Since the exposure situation with respect to contaminated dust could be different near a foundry to the residential scenario (RL), another scenario of a child being exposed to contaminated dust released by a foundry is considered. Unlike scenario RL, covering a general situation including landfills, no food consumption is considered here, because the presence of contaminated material offsite is already covered by scenario RL.

- Scenario RH

Contaminated material (building rubble, slag, fly ash) may be used in the construction of buildings as concrete aggregate or cement substitute. This will lead to an external exposure of the building residents, which is addressed in this scenario. Other possible uses of material cleared from nuclear facilities in private homes are also covered by this scenario (e.g., the use of steel plates for the cladding of walls).

- Scenario RP

If contaminated material is used for covering public places there will be external exposure and the inhalation and ingestion of contaminated dust for residents (e.g., playing children). This exposure situation is covered in this scenario.

- Scenario RW

The presence of contaminated material may lead to a release of radionuclides into a groundwater aquifer. This may affect downstream wells. As a consequence, this may lead to the ingestion of contaminated drinking water or of contaminated foodstuff produced in a private garden if the well water is used for irrigation. If the contaminated groundwater discharges into a river, the additional pathway of fish consumption has to be considered.

The identified scenarios encompass all reasonable situations worldwide without specifying a specific situation. The scenarios are not intended to account for worst-case scenarios, outlier scenarios or scenarios that apply to a very few individuals. In this way the scenarios are not bounding.

Development of the scenarios is approached by the examination of the parameters of the dominant exposure pathways, and the parameters are adapted to ensure worldwide applicability to a variety of situations. Care is taken to ensure that the parameter values are internally consistent within a particular scenario.

The limiting scenario may be different for different countries, because of different exposure geometries, working hours, sizes of transportation vehicles, etc. Thus, different sets of parameters could be chosen in different countries but the linkage of all relevant parameters needs to be taken into account in developing the scenarios. There are balancing effects between sets of parameters; while one parameter may be higher in one set than in another, other parameters may be lower and compensate for the higher parameter. The enveloping parameter set has been chosen carefully to avoid over-conservatism. The most restrictive parameters are not necessarily all gathered into the enveloping scenario.

A number of scenarios are required which cover all relevant aspects of external irradiation, inhalation, and ingestion in such a way that any exposure situation, that is reasonable to assume, would not lead to higher doses. Whereas the exact parameter values may be material specific, the general categories of scenarios and formulae are common to all material.

For each scenario, two distinct approaches have been used:

- The first approach is to make the calculations with realistic scenario parameter values using an effective dose criterion of 10 μSv/a.

- The second approach is to use a set of low probability scenario parameter values using an effective dose criterion of 1 mSv/a and a skin equivalent dose limit of 50 mSv/a.

The approach applied differs from the derivation of clearance values or exemption levels made by other organizations [7], where only the predominant exposure pathway and not the sum of all exposures within a exposure situation is taken as the basis for comparison to the dose criterion. The reason for adopting this different approach is twofold:

- The original derivation of the 10 μSv/a criterion was based on a dose of 100 μSv/a that was considered acceptable as a trivial risk. But since an individual may be exposed to several exposure sources over different pathways, the criterion was divided by ten accounting for this possible multiple exposures. The derivation of activity concentration levels presented here, however, also is based on the 1 mSv/a public dose criterion for the low probability parameter assumptions. In this case, no allowance can be made for multiple exposure pathways affecting one individual because the dose criterion refers to the overall exposure of a member of the public. Therefore, the sum of all exposures affecting one individual in a specific situation has to be considered.

- The scenarios have been defined combining only those exposure pathways that will occur simultaneously in a particular situation with a high probability. For example, a landfill worker dealing with contaminated material will, in most cases, be affected by external exposure as well as by dust inhalation and ingestion.

Therefore, it is considered prudent to base the derivation of the activity concentration levels on the sum of exposure pathways having a high probability of affecting an individual simultaneously.

The situation could also occur that the different defined scenarios affect one individual. For example, the landfill worker may happen to live in a house constructed with contaminated material. A further combination of these exposures to yield the hypothetical maximum exposure to an individual is not considered appropriate:

- For realistic parameters used in the scenarios, comparison is made with the 10 μSv/a criterion, allowing for possible multiple exposures as discussed above. Consequently, the activity concentration levels based on realistic parameters implicitly take account of the possibility of such unlikely but possible multiple exposures.

- Comparing exposures to the 1 mSv/a dose criterion, on the other hand, involves low probability assumptions for each scenario. Therefore, the assumption that one individual is exposed by two different scenarios, having only a small probability of occurrence as such, plus the further assumption that in both scenarios the low probability parameters are adequately describing the situation has only a negligible overall probability of occurrence. It is therefore reasonable to assume that for one individual, at a maximum, one exposure scenario will correspond to the low probability parameters. This scenario then dominates the assessment based on the 1 mSv/a dose criterion, and the possible simultaneous exposure through another scenario contributing only 10 μSv/a is not of consequence.

3.4. SHORT-LIVED RADIONUCLIDES

According to the overall concept outlined in the Safety Guide, the activity concentration levels should be lower than or equal to the exemption levels given in the BSS, because the activity concentration levels define the entry level into the regime of the BSS while the exemption levels are criteria within the scope of the BSS for exemption from this regime for material with small activity concentrations and total activities. This condition is satisfied by the results of the defined scenarios for most of the radionuclides, but not for all of them.

The calculated activity concentration levels are higher than the exemption levels for a number of radionuclides with short half-lifes. The reason for this lies in the fact that the scenarios used to determine the activity concentration levels are focusing on the handling (transport, trade, use, or deposition) of the material outside the facilities in which they arise (i.e., reactors, accelerators, laboratories), because these facilities will be under regulatory control in any case. As a consequence, the scenarios used for the activity concentration levels always consider a decay time before the start of the exposure (see Section 4.2), which is assumed to be at least one day (or considerably longer for some scenarios). The calculations on which the activity concentration levels in the BSS are based do not consider decay times because they also cover the direct handling of the material in the facilities where the material arises.

In order to cover the direct handling of the material in the derivation of the activity concentration levels, scenarios could be added in analogy to those used for the BSS. However, this would not add any new information. Therefore, it is concluded to define the activity concentration levels as the minimum of the scenario results presented and the exemption levels given in the BSS. This assures that the case of direct handling of the material is adequately reflected in the activity concentration levels also for the short-lived radionuclides.

3.5 NATURAL RADIONUCLIDES

Scenarios were not used for calculating activity concentration levels for naturally occurring radionuclides. Rather, they were based on consideration of worldwide distribution of concentrations of naturally occurring radionuclides.

3.6. MIXTURES

To apply the activity concentration levels to a material containing a mixture of radionuclides (either artificial or naturally occurring), the concentrations should be determined as follows:

(1) artificial

$$\sum_{i=1}^{n} \frac{C_{i(artificial)}}{Activity\ concentration_i} \langle 1$$

(2) naturally occurring — For each radionuclide*:

$$\frac{C_{natural}}{Activity\ concentration} \langle 1$$

* In case of secular equilibrium, all $C_{natural}$ of a chain are equal.

where $C_{i(artificial)}$ is the concentration (Bq/g) of artificial radionuclide is in the material, *activity concentration*$_i$ is the activity concentration level for the artificial radionuclide in that material and n is the number of radionuclides in the mixture. For equation 2, $C_{natural}$ is the concentration (Bq/g) of naturally occurring radionuclide in the material or for those materials in secular equilibrium, it is the concentration of the parent nuclide, and *activity concentration* is the activity concentration level for the naturally occurring radionuclide (or for those in secular equilibrium, the parent nuclide).

If both (1) and (2) are satisfied and are less than or equal to 1, then the material should not be attributed to radiation protection considerations. If either sum is greater than one, the requirements of the BSS [2] should be applied to the material as given in section 2 of this document. This type of relationship should be used by national regulatory bodies in their specific guidance on application of the BSS [2] to account for situations where multiple radionuclides are present in mixtures.

It is worth noting that this is a conservative approach since the pathways of exposure of the critical group of exposed individuals is not necessarily the same for each nuclide, because of partitioning or separation of nuclides by processes. In many cases it will be useful to identify a measurable indicator nuclide within the spectrum and apply correspondingly, a sum-index as defined above.

3.7. AVERAGING PROCEDURE

When applying the derived activity concentrations, the regulatory body should consider methodologies for sampling, averaging, monitoring, and detection of radionuclides. In doing this, the regulatory body should recognize that these activity concentrations were derived for large quantities and therefore the averaging should be done accordingly. Consideration should also be given to surface contamination levels that would equate to the specified dose criteria. The Agency is currently preparing guidance on these issues.

3.8. EFFECTS OF PARTICLE SIZES

The activity concentration levels are based on the average activity concentration in a material. For material exhibiting a particle size distribution (e.g., building rubble, soil, ashes), the average activity concentration is not necessarily identical with the activity in certain particle size fractions. A well-known example is the distribution of the activity between ingot, slag, and fume during the smelting of contaminated metal. Depending on technical parameters and on the chemical properties of the radionuclides, a substantial enrichment of the activity concentration may be found in the slag or in the fume.

For much other material not arising from thermal processes, higher activity concentrations in fine fractions may be observed. This phenomenon can occur for material consisting of individual particles by the transfer of dissolved radionuclides into the material with a fluid phase (e.g., contamination from spills). A non-uniform activity concentration over particle size may also be caused or further enhanced by a redistribution of the activity in the material through leaching by fluids. An enhanced activity concentration of the fine fraction also obviously results when the activity is brought into the material with fine particles (e.g., deposition of dust or fumes on surfaces).

A higher activity concentration in the fine fraction has to be considered in assessments of the inhalation pathway. It is also relevant for the direct ingestion of contaminated material because this also refers to the fine fraction.

Several investigations have been performed concerning the smelting of metal. On the basis of these studies, element specific enrichment factors in the fumes between 1 and 70 have been derived [9]. These are applied in the calculations performed here for the foundry scenarios WF and RF.

For material other than metal, the situation is more complicated. The investigation of the processes that may lead to an enriched activity in the fine fraction shows that the actual activity distribution over particle size will depend on many factors, such as the type of material, its physical and chemical properties, and the origin and possible later redistribution of the contamination. This obviously causes difficulties for a generic assessment. Nevertheless, it is considered more appropriate to take account of this phenomenon even in a crude fashion rather than ignoring it in total.

On this basis, it is assumed for material other than metal, the activity concentrations in the respirable fine fraction are a factor of four higher than in the average of the material. For the dust that is subject to direct ingestion, a factor of two is used because this pathway on the average refers to coarser particles. These numbers are based on comprehensive investigations carried out on soil-like material in Germany [10]. It should be noted that the chosen factors do not correspond to the maximum values observed in these studies. But they are considered reasonable as an assumption covering the broad majority of material.

3.9 SURFACE CONTAMINATION

The activity in a material is not in all cases fully characterized by the activity concentration. Apart from particle size effects discussed above, a major portion of the activity may be concentrated on the surface of the material. This is in particular relevant for metals and buildings, but also other material may exhibit a surface contamination depending on their nature and on the origin of the contamination.

The difference between contaminants present preferentially on the surface as compared to the bulk of a material only plays a minor role for the important pathways of external irradiation and of food ingestion, and does not affect exposure estimates significantly. For the inhalation and ingestion of contaminated dust, however, this difference can become very important. A well-known example is the massive release of surface-bound radionuclides during the thermal cutting of metals, giving rise to a multiple of the doses that are to be expected if the radionuclides are evenly distributed throughout the bulk of the material.

This aspect has been intensively considered in several studies relating specifically to the clearance of material from nuclear installations [9, 11, 12]. For the purpose of the generic derivation of activity concentration levels; however, such factors cannot be taken into account. Therefore, it has to be recognized that for specific situations such as the clearance of metal or the reuse of buildings from nuclear installations, additional criteria relating to the surface contamination may have to be applied which are not reflected in the derived activity concentration levels. This may lead to the decision of the regulatory body not to release some material even if the activity concentration levels are not exceeded for the bulk quantity.

4. DEVELOPMENT OF ACTIVITY CONCENTRATION LEVELS FOR ARTIFICIAL RADIONUCLIDES

4.1. OVERVIEW

An overview of the scenarios considered in the derivation of activity concentration limits for artificial radionuclides and the relevant pathways is given in Table II. The basis for the exposure estimates and the parameters used for the realistic and low probability cases are described in the following sections. Section 4.2 presents scenario specific assumptions on exposure and decay times as well as dilution factors. Section 4.3 discusses the specific approaches for the modeling of the relevant exposure pathways.

TABLE II. EXPOSURE SCENARIOS CONSIDERED AND RELEVANT PATHWAYS

Scenario	Description	Exposed Individual	Relevant Exposure Pathways
WL	Worker on landfill or in other facility (other than foundry)	worker	External exposure on landfill
			Inhalation on landfill
			Direct ingestion of contaminated material
WF	Worker in foundry	worker	External exposure in foundry from equipment or scrap pile
			Inhalation in foundry
			Direct ingestion of contaminated material
WO	Other worker (e.g., truck driver)	worker	External exposure from equipment or truck load
RL-C	Resident near landfill or other facility	child (1-2a)	Inhalation near landfill or other facility
			Ingestion of contaminated foodstuff grown on contaminated land
RL-A		adult (>17 a)	Inhalation near landfill or other facility
			Ingestion of contaminated foodstuff grown on contaminated land
RF	Resident near foundry	child (1-2a)	Inhalation near foundry
RH	Resident in house constructed of contaminated material	adult (>17 a)	External exposure in house
RP	Resident near public place constructed with contaminated material	child (1-2a)	External exposure on place
			Inhalation of contaminated dust
			Direct ingestion of contaminated material
RW-C	Resident using water from private well or consuming fish from contaminated river	child (1-2a)	Ingestion of contaminated drinking water, foodstuff and fish
RW-A		adult (>17 a)	

4.2. GENERAL PARAMETERS FOR SCENARIOS

For each scenario, general parameters are defined that characterize the exposure situation:

- Exposure time;
- Decay time allowed before the scenario starts; and
- Decay time during the scenario.

The decay time before the scenario addresses the period of time between the determination of compliance with the activity concentration levels for the material in question and the actual start of the exposure.

The decay time during a scenario defines the time intervals at which new material is brought into a facility or used for construction purposes. Since exposures in individual years are considered, a maximum of 365 days of decay can be taken into account during a scenario, even if the deposition of material is a single event or if there is no new material used as in the case of a building after the construction is finished.

Decay times for the growing of foodstuff on contaminated land are treated separately because the material in this case has to be present in the area concerned for a considerable period of time before the growing of plants is expected to start.

The following values for these parameters for the realistic assumptions and for the low probability case (see Section 2) are used:

- Exposure time:

  - For all workplace scenarios except WO a range between a quarter of a working year (realistic assumption) and a full working year (low probability assumption) is used. For scenario WO an exposure time of 900 hours, corresponding to half a working year, is used in order to cover the case that a piece of equipment cleared from a nuclear facility is re-used.

  - The realistic time residents are exposed from a facility is set to 1000 hours per year. But since the dust within a building very close to a facility may also be impacted, a low probability assumption of a continuous exposure throughout a year is made. This covers, for example, a child spending most of the time in the house or in its vicinity.

  - With similar arguments, the low probability assumption for the scenario of living in a house constructed from the material is set to a continuous exposure (8760 hours). As a realistic assumption, 4500 hours are used.

  - For the case of children playing on a public place covered with the material, exposure times are assumed as 400 (about 1 hour per day) to 1000 hours, the upper bound being sufficient to address children playing on this place for about 3 hours every day.

- Decay times:

  - Decay times are chosen identically for all scenarios in which the exposure is due to material brought into a facility for processing or deposition. For the realistic case, a decay time before the scenario of 30 days and a decay during the scenario of 365 days is used. The latter corresponds to the assumption that the facility receives such material only once or at least infrequently. A facility processing such material on a routine basis is covered by the low probability assumptions with only a one day decay time before the scenario and no decay during the scenario.

  - The two considered scenarios where the material has been used for construction purposes (building or public place) assume a decay time before the start of the scenario of 100 days. This allows for the preparation of the building material and the construction phase. Since no new material will be brought in after the construction is complete, a 365 day decay time during the scenario is assumed.

  - For the growing of foodstuff on an area contaminated by the material a decay time of 365 days before the start of the scenario is assumed. Since new material will not be added (or only a infrequent basis as, for example, in the case of wood chips), the decay time during the scenario is also set to 365 days.

- For the water pathways, decay times are considered within the model applied (see Section 4.3.4). General assumptions are therefore not required.

The parameter values are provided in Table III.

TABLE III. GENERAL PARAMETER OF EXPOSURE SCENARIOS

Parameter	Unit	Case	Scenario						
			WL	WF	WO	RL	RF	RH	RP
			worker landfill	worker foundry	other worker	resident landfill	resident foundry	resident house	resident place
Exposure time (t_e)	h/a	realistic	450	450	900	1000	1000	4500	400
		low prob.	1800	1800	1800	8760	8760	8760	1000
Decay time before scenario (t_1)	d	realistic	30	30	30	30	30	100	100
		low prob.	1	1	1	1	1		
Decay time during scenario (t_2)	d	realistic	365	365	365	365	365	365	365
		low prob.	0	0	0	0	0		
Decay time before food scenario (t_{f1})	d	realistic	N/A	N/A	N/A	365	N/A	N/A	N/A
Decay time during food scenario (t_{f2})	d	realistic	N/A	N/A	N/A	365	N/A	N/A	N/A

4.3. MODELLING OF EXPOSURE PATHWAYS

In the following, the exposure models and the parameters used are described for all pathways relevant to the exposure scenarios considered. The results of the calculations are shown in Appendix II. The activity concentration levels are shown in Tables XV and XVI.

4.3.1. External Exposure

Exposure situations in which external exposure is relevant are quite varied and may include exposure on a landfill or garden where waste that has been released from regulatory control is disposed (landfill worker), working near a large piece of cleared equipment or while staying in a building that is constructed using building rubble or other material (e.g., slag or fly ash) that has been released from regulatory control as aggregate for the new concrete or as substitute for cement in the concrete. The scenarios considered are defined to cover these and similar situations.

The dose from external exposure is calculated according to equation (1):

$$E_{ext,C} = \dot{e}_{ext} \cdot t_e \cdot f_d \cdot e^{-\lambda t_1} \frac{1-e^{-\lambda t_2}}{\lambda \cdot t_2} \qquad (1)$$

where

$E_{ext,C}$ [(μSv/a)/(Bq/g)] individual effective dose in a year from external exposure per unit activity concentration in the material,

$\dot{e}_{ext}$ [(μSv/h)/(Bq/g)] average effective dose rate per unit activity concentration in the material, depending on geometry, distance, shielding, age group etc.,

t_e [h/a] exposure time,
f_d [-] dilution factor,
λ [1/a] radioactive decay constant,
t_1 [a] decay time before start of scenario, and
t_2 [a] decay time during scenario.

External exposures are assessed for four different situations as required by the definition of the scenarios in Table II with the following parameters:

- Dilution factor:
  - In a realistic situation, a dilution of at least 1:10 is reasonable to assume for the landfill scenario while the low probability approach assumes no dilution.
  - For the external irradiation in a foundry processing the material it is assumed that a worker is in contact with a larger piece of equipment or a pile of scrap. This also covers a truck driver bringing material to a foundry or a landfill. The same range for the dilution factor is assumed as for the landfill scenario.
  - In scenario RH it is assumed that a person spends time in a room or enclosure that is partially made from the material (e.g., by using building rubble, slag or ash as aggregate or cement substitute in concrete). It is assumed that the material of which the room or enclosure is constructed, will in realistic circumstances, be mixed 1:10 with other material. Since the construction material can, for technological reasons, contain only a certain percentage of building rubble, ashes or similar material, an upper limit for the dilution of 0.5 is assumed for the low probability case.
  - The scenario RP considers playing children on a public place partially made from the material. The dilution factor for realistic parameters is assumed at 0.1. For the low probability case a factor of 0.5 is chosen, because the public place is not likely to be covered with a deep cover of the material – either the cover will consist only of a relatively thin layer of, for example, ashes or slag, or there will be some mixing with other material. A factor of 0.5 is felt to provide a sufficiently conservative upper estimate.
- Density of material:
  - The density of the material only has a relatively small effect on the results. For a higher density, more activity is present per volume of the material (with a given mass specific activity concentration). This increases the number of photons emitted; however, self-absorption of the gamma radiation by the material increases as well.
  - On these grounds, a homogeneously distributed source in the material is assumed for which a density of 1.5 g/cm³ is used for the dose calculations in all scenarios.

- Geometry:
  - In the landfill scenario and for the public place, doses are calculated for rotational exposure geometry at 1 m height above the ground.
  - To estimate exposures from a large item (equipment, pile of scrap, truckload) the exposure geometry is chosen to be a slab 5 m × 2 m × 1 m thick. The dose coefficients for this exposure situation are almost identical to those for a smaller piece of equipment (5 m × 2 m × 1 m) made of steel (density 7.8 g/cm^3) considered in other models set up for the derivation of clearance values. Thus, the scenario presented here covers both situations.
  - For the building constructed of contaminated material, the exposure geometry chosen is a room[9] of 3 × 4 m^2 with a height of 2.5 m. The calculations are based on 2 walls and a ceiling that are 20 cm thick. It is assumed that windows and doors account for the other 2 walls and that the floor would be made of other material. Doses are calculated for a rotational geometry in the middle of the room at a height of 1 m. Doses calculated in clearance studies for the use of steel plates cleared from nuclear facilities are considerably smaller than those in the case considered here. Thus, this scenario is covered here as well.
- Dose coefficients:
  - Doses are calculated for adults in the workplace scenarios and for the resident in the house. For the public place, dose calculations are performed for children between 1 and 2 years of age.[10]

The parameter values are provided in Table IV.

[9] The actual size of the room is of minor importance. If, e.g., the room is much longer in one dimension, say 8 m instead of 3 m, the dose coefficient increases by only 10%.

[10] The inclusion of children between one and two years of age in the reference groups is consistent with a strict interpretation of the exemption criterion (10 μSv/a) as relating to any single year of exposure; in terms of radiological risk from protracted low level exposure a much longer integration period could be considered so that children of a specific age group would normally not be in the most restrictive age group.

TABLE IV. PARAMETERS FOR EXTERNAL IRRADIATION SCENARIOS

Parameter	Unit	Case	WL	WF/WO	RH	RP
			worker landfill	foundry or other worker	resident house	resident place
Dilution factor (f_d)	[-]	realistic	0.1	0.1	0.1	0.1
		low prob.	1	1	0.5	0.5
Density of material	g/cm³		1.5	1.5	1.5	1.5
Geometry			1 m above ground, semi-infinite source	1 m from load or item 5x2x1m³, no shielding	Ceiling, 2 walls, 3x4 m², 2.5 m height, 20 cm wall thickness.	1 m above ground, semi-infinite source
Dose rate coefficient ($\dot{e}_{ext}$)	µSv/h/(Bq/g)		(adult)	(adult)	(adult)	(child 1-2 a)
			Depending on radionuclide and geometry			

4.3.2. Inhalation

Inhalation of contaminated dust can occur in many exposure situations. Therefore, representative exposures for workplaces and for the general population are considered. A child (age group 1-2 a) is chosen as the reference age group in the latter case.

Doses from inhalation are calculated according to equation (2):

$$E_{inh,C} = e_{inh} \cdot t_e \cdot f_d \cdot f_c \cdot C_{dust} \cdot \dot{V} \cdot e^{-\lambda t_1} \frac{1-e^{-\lambda t_2}}{\lambda \cdot t_2} \qquad (2)$$

where

- $E_{inh,C}$ [(µSv/a)/(Bq/g)] individual effective dose in a year from inhalation per unit activity concentration in the material,
- e_{inh} [µSv/Bq] effective dose coefficient for inhalation (see section 3.1.),
- t_e [h/a] exposure time,
- f_d [-] dilution factor,
- f_c [-] concentration factor of specific activity in the fine fraction,
- C_{dust} [g/m³] effective dust concentration in the air,
- $\dot{V}$ [m³/h] breathing rate,
- λ [1/a] radioactive decay constant,
- t_1 [a] decay time before start of scenario, and
- t_2 [a] decay time during scenario.

The inhalation pathway is relevant for most of the scenarios considered. The following parameters are used:

- Dilution factor:
  - For the landfill, the same range (0.1 to 1) for the dilution factor is used as for external irradiation.

- The dilution factor for the foundry is chosen as 0.02 in the realistic case, accounting for the fact that typical foundries process large amounts of scrap material. For the low probability case, a factor of 0.1 is used.[11]

- For the residents living in the vicinity of a landfill or facility, the dilution factors are reduced by a factor of 10 as compared to the assumptions within the facility. This takes into account that several other sources will contribute to the airborne dust outside the facility.

- On the public place a realistic dilution factor of 0.1 is assumed in accordance with the assumptions for the external exposure. However, the low probability assumption of the external exposure pathway of 0.5 dilution is not used for the inhalation pathway, because the material may have been used for covering the place with a thin layer (e.g. ash). Since the airborne dust in this case would be almost completely generated from the cover layer, no dilution is assumed in the low probability case.

• Dust concentration in air:

  - For the workplaces, a realistic dust concentration in air of 0.5 mg/m^3 and a low probability value of 1 mg/m^3 is assumed.

  - The range for the dust concentration in air for the scenarios outside the facilities are reduced to 10^{-4} for realistic assumptions and to $5x10^{-4}$ for low probability assumptions.

• Concentration factor of specific activity in the fine fraction:

  - The higher activity in the fine fraction as compared to the material average is taken into account according to the discussion in Section 3.7. For metal smelting an element dependent range between 1 and 70 is used, while for other materials a factor 4 is used.

• Breathing rate:

  - The breathing rate for workers and other adults is set to 1.2 m^3/h (accounting for moderate physical activity). For children between one and two years of age a breathing rate of 0.22 m^3/h is applied.

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[11] It should be noted that for the external irradiation in the foundry, a dilution factor in the range of 0.1 to 1 is used, corresponding to the landfill scenario. The reason for adopting a lower factor for the inhalation pathway is as follows: A worker in the foundry may be specialized on processing certain material types in preparation to smelting (e.g., stainless steel). Consequently, this worker may be exposed to the material of concern on a frequent basis, which is accounted for by the lower dilution considered for the external exposure as well as for the material ingestion scenarios. The radionuclide concentrations in the fumes present in the foundry, on the other hand, will be determined by the overall dilution of the material processed in the facility, which is expected to be considerably higher.

- Dose coefficients:
    - Dose coefficients for workers are taken from the BSS [2] for 5 μm AMAD (Activity Median Aerodynamic Diameter). For the public, dose coefficients are taken from the BSS [2] for the default lung retention class and the appropriate age group.

Parameter values are provided in Table V.

TABLE V. PARAMETERS FOR INHALATION SCENARIOS

Parameter	Unit	Case	WL	WF	RL-A	RL-C	RF	RP
			worker landfill	worker foundry	resident landfill		resident foundry	resident place
Dilution factor (f_d)	[-]	realistic	0.1	0.02	0.01	0.01	0.002	0.1
		low prob.	1	0.1	0.1	0.1	0.01	1
Dust concentration in air (C_{dust})	g/m³	realistic	5×10^{-4}	5×10^{-4}	10^{-4}	10^{-4}	10^{-4}	10^{-4}
		low prob.	10^{-3}	10^{-3}	5×10^{-4}	5×10^{-4}	5×10^{-4}	5×10^{-4}
concentration factor (f_c)	[-]		4	1 – 70	4	4	1 – 70	4
Breathing rate ($\dot{V}$)	m³/h		1.2	1.2	1.2	0.22	0.22	0.22
Dose coefficient (e_{inh})	μSv/Bq		5 μm, worker, see 3.1.	5 μm, worker, see 3.1.	adult, see 3.1.	child (1-2a), see 3.1.	child (1-2a), see 3.1.	child (1-2a), see 3.1.

4.3.3. Ingestion

Two types of exposure pathways are considered for ingestion:

- Inadvertent direct ingestion of dust (e.g. via hand-to-mouth-pathway), and
- Ingestion of crops which are grown in the material in question (e.g. soil) which the nuclides enter via the roots of the plants.

The growing of plants in soil that contains material that has been released from regulatory control might occur in the following situations: released building rubble which is present in soil in small fractions, released soil from a nuclear site which is used in a garden or which has been used for covering an old landfill site which later on is used as a recreational area, or even reuse of a former nuclear site for general purposes. The foodstuff scenario RL-A accounts for an adult who will consume vegetables grown in the material, RL-C covers the exposure of children in the same situation.

The dose from ingestion is calculated according to equation (3):

$$E_{ing,C} = e_{ing} \cdot q \cdot f_d \cdot f_c \cdot f_t \cdot e^{-\lambda t_1} \frac{1-e^{-\lambda \cdot t_2}}{\lambda \cdot t_2} \qquad (3)$$

where $E_{ing,C}$ [(μSv/a)/(Bq/g)] individual effective dose in a year from ingestion per unit activity concentration in the material,

e_{ing} [μSv/Bq] dose coefficient for ingestion see section 3.1.,
q [g/a] ingested quantity per year,
f_d [-] dilution factor,
f_c [-] concentration factor in fine fraction,
f_t [-] root transfer factor,
λ [1/a] radioactive decay constant,
t_1 [a] decay time before start of scenario, and
t_2 [a] decay time during scenario.

The factor f describes the transfer of elements from soil to plants for those circumstances where growing of foodstuff in soil mixed with material that has been released from regulatory control is considered. This factor accounts for the fact that the uptake of radionuclides in plants depends on the element. Values for f are given in Bq/kg in the plant per Bq/kg in the soil (i.e., they are dimensionless) and are provided in Safety Report 19 [13].

The following parameters are used for the ingestion scenarios:

- Dilution factor:
  - Assumptions for the dilution of dust ingested inadvertently by a resident near a landfill are identical to those for the inhalation pathway. For the growing of foodstuff a realistic dilution of 0.01 and a low probability dilution of 0.1 is used. This dilution covers the fact that only part of the soil will consist of the material. It is also assumed that only a portion of the annual dietary intake will be grown in the garden. With the combination of these two factors, the assumed range is considered to be adequate.
- Concentration factor of specific activity in the fine fraction:
  - This factor is only relevant for the direct ingestion of material. For the particle size fraction that may be subject to direct ingestion a concentration factor of 2 is used according to the discussion in Section 3.8.
- Root transfer factor:
  - This factor is only relevant for the ingestion of foodstuff. Root transfer factors describing the transfer of radionuclides from the soil to the plants are provided in [13].
- Annually ingested quantity:
  - For the direct ingestion of a worker a quantity of 10 g/a is assumed. A low probability approach is to use 50 g/a.
  - The amount of dirt and dust which a small child may inadvertently swallow when playing on a public place covered with the material could amount under realistic assumptions to 25 g/a. The low probability approach is to assume an ingested quantity of 50 g/a.

- For the foodstuff pathway the annual consumption of vegetables and fruits is considered that may be grown in the garden.[12] Consumption quantities used are for the realistic case 68 kg per year for children and 88 kg per year for adults. In the low probability scenarios consumption rates of 204 kg per year for children and 264 kg per year for adults are used. The derivation of these assumptions is provided in connection with other consumption parameters required for the water pathway model in Section 4.3.4. A dilution with foodstuff from other sources already has been taken into account in the assumptions for the dilution factor.

- Dose coefficients:
  - The ingestion dose coefficients are taken from the Basic Safety Standards [2] for workers or the appropriate age group of the public.

Parameter values are provided in Table VI.

TABLE VI. PARAMETERS FOR INGESTION SCENARIOS

Parameter	Unit	Case	WL/WF	RP	RL-A	RL-C
			landfill or foundry worker	resident place	resident landfill	
Dilution factor (f_d)	[-]	realistic	0.1	0.1	0.01	0.01
		low probab.	1	1	0.1	0.1
Concentration factor (f_c)	[-]		2	2	N/A	N/A
Root transfer factor (f_t)	[-]		N/A	N/A	[12]	[12]
Annually ingested quantity (q)	g/a or kg/a	realistic	10 g/a	25 g/a	88 kg/a	68 kg/a
		low probab.	50 g/a	50 g/a	264 kg/a	204 kg/a
Dose coefficient (e_{ing})	μSv/Bq		worker; see 3.1.	child (1-2a), see 3.1.	adult, see 3.1.	child (1-2a), see 3.1.

4.3.4. Water Pathway

Water pathways are included in radiological assessments in those cases where large quantities of material that has been removed from regulatory control are disposed or stored in a single place where rain can reach the material and dissolve its residual contamination that is then carried away to a groundwater layer or to surface water. The radionuclides can enter the human food chain if the water is used as drinking water or for irrigation purposes. In the case of groundwater contamination, it is conceivable that the water is taken from a private well that is not subject to any legal requirements concerning the water quality, while in the case of surface water contamination, the water might be used by municipal water works. Various investigations have demonstrated that the private well supplying groundwater to a family is the most restrictive of the various water pathways. If the contaminated water is discharged

[12] This scenario does not consider other agricultural products like grain, meat, or milk. Such products would require substantially larger areas as compared to the growing of vegetables or fruit in a private garden. This would lead to substantially higher dilution factors because it cannot reasonably be assumed that large agricultural areas are contaminated in total with the material. Therefore, the consideration of a private garden with limited types of foodstuff produced represents the covering scenario for the food pathway.

into surface water an additional exposure pathway to be taken into account is the ingestion of contaminated fish.

Modeling a water pathway requires assumptions about the quantity of material that is stored or disposed, the location (landfill site, public area, etc.) where it is placed and the characteristics of the environment (e.g., hydrogeology). These factors are highly site-specific making the generic modeling of the water pathway difficult. Nevertheless, it is considered more appropriate to include the water pathway into the assessment in spite of this difficulty than to disregard this pathway in total.

The model used for the water pathway is described in the following. In line with the overall approach a realistic case and a low probability case are considered. Assumptions for the latter case represent unfavorable site and exposure conditions, so that the modeling results are considered to cover all situations that are reasonably to be expected.

The models developed are based on the RESRAD computer model developed for radiation dose estimates arising from residual radioactive material [14]. This computer model has been widely used for exposure assessments and has been benchmarked against other models. A direct use of RESRAD for modeling the water pathway, however, was not possible because not all of the nuclides relevant here are considered in RESRAD. Moreover, only a small subset of the models implemented in RESRAD actually are required here. Therefore, it was decided to develop a new model based on algorithms and assumptions provided in the RESRAD documentation. In order to verify the model developed, its results were checked against RESRAD results for selected radionuclides.

4.3.4.1. Model equations

The modeling of the water pathway assumes an extended source of the material present in the catchment area of a groundwater aquifer. This could be a landfill or the consequence of the use of the material in a landscape construction project.

The model assumes conservatively that the whole inventory of radionuclides in the material is available for migration. The rate at which the radionuclides are released is determined using a K_d model [14]. Within this model the leach rate of the radionuclide i from the source L_i is given as:

$$L_i = \frac{I}{\theta^{cz} \cdot z^{cz} \cdot R_i^{cz}} \qquad (4)$$

where

- I [m/a] infiltration rate,
- θ^{cz} volumetric water content of the contaminated zone (dimensionless),
- z^{cz} [m] thickness of contaminated zone,
- R_i^{cz} retardation factor for radionuclide i (dimensionless),

The retardation factor is given by:

$$R_i^{cz} = 1 + \frac{\rho^{cz} \cdot K_{d_i}}{\theta^{cz}} \tag{5}$$

where

ρ^{cz} [g/cm^3] density of contaminated zone, and

K_{d_i} [cm^3/g] distribution coefficient for radionuclide i.

The decisive parameter determining the leaching of different radionuclides from the contaminated zone is the distribution coefficient. This quantity is dependent on the chemical characteristics of the radionuclide and the geochemical properties of the soil. Values provided for different elements in the literature vary considerably. For the purpose of the generic model developed here it is therefore necessary to select conservative estimates from the values published for different elements.

For the realistic scenario the default values used in the RESRAD model are used. These are already reasonably conservative in comparison to other values published (see Table E.4 in [14]). For some nuclides, however, lower values are reported in this table. The low probability scenario therefore uses the minimum values for the distribution coefficients provided in Table E.4 of [14].

For some elements no measurements of distribution coefficients are available. In this case the approximation given in Appendix H of [14] is used, estimating the distribution coefficient from the root transfer factor f_i (see Section 4.3.3) as

$$\ln K_{d_i} = a + b \cdot \ln f_i \tag{6}$$

with a = 2.11 (valid for sandy soil) and b = -0.56.

The values of the distribution coefficient used for the different elements are given in Table VII. Values derived from Equation (6) are indicated. The remaining values are based on measurements.

TABLE VII. DISTRIBUTION COEFFICIENTS (cm^3/g)

Element	realistic	low probability
Ag	0	0
Am	20	20
Ba	50	44*
Bi	0	0
Bk	213*	213*
C	0	0
Ca	50	5
Cd	0	0
Ce	1000	500
Cf	109*	109*
Cl	3*	3*
Cm	395*	395*
Co	1000	60
Cs	1000	270
Es	213*	213*
Eu	268*	240
Fe	1000	160
Gd	182*	182*
H	0	0
Ho	182*	182*
I	0.1	0.1
La	213*	213*
Mn	200	50
Mo	20*	10
Na	10	10
Nb	0	0
Ni	1000	300
Np	50*	5
Pd	30*	30
Pm	268*	240
Pt	12*	12*
Pu	2000	550
Rb	20*	20*
Rh	44*	44*
Ru	0	0
Sb	0	0
Se	0	0
Sm	182*	182*
Sn	0	0
Sr	30	15
Tb	182*	182*
Tc	0	0
Te	0	0
Th	60000	1378
Tl	0	0
Tm	213*	213*
U	50	15
Zn	0	0
Zr	395*	280

* *value calculated using Equation 6*

It should be noted that K_d values in concrete situations may be considerably different from the numbers given in Table VII. It may also be the case that the linear K_d model is not adequate for certain site conditions (e.g. because of the presence of other chemical substances or because of adsorption saturation effects). Therefore, it cannot be assumed that leach rates in all cases are covered by the model presented. This possibility, however, has to be seen in the overall context of relatively conservative assumptions used, so that a higher leach rate for some radionuclides under specific site conditions does not necessarily mean that eventual exposures are higher than predicted by the model.

The radionuclide concentration in the seepage C_i^s for radionuclide i can be calculated from the leach rate L_i as:

$$C_i^s = \frac{M \cdot c_i \cdot L_i}{U^s} \tag{7}$$

where

- M [g] total mass of contaminated material,
- c_i [Bq/g] specific activity of radionuclide i in the contaminated material,
- L_i [1/a] leach rate for radionuclide i according to Equation (4), and
- U^s [m^3/a] volume of seepage through contaminated zone.

The volume of the seepage through the contaminated zone U^s is given by:

$$U^s = I \cdot A^{cz} \tag{8}$$

where

I [m/a] infiltration rate,

A^{cz} [m^2] surface area of contaminated zone.

It is assumed that the seepage from the source is discharged into an aquifer. For the realistic scenario, it is assumed that there is an unsaturated zone between the contaminated material and the aquifer. Its presence will only have an effect on the eventual contaminant concentration in the seepage reaching the aquifer through radioactive decay of the radionuclides while migrating through the unsaturated zone. The transport time through this zone is given by the following equation:

$$t_i = \frac{z^{uz} \cdot R_i^{uz} \cdot p^{uz} \cdot R_s^{uz}}{I} \tag{9}$$

where

I [m/a] infiltration rate,

z^{uz} [m] thickness of contaminated zone,

p^{uz} effective porosity of the unsaturated zone (dimensionless),

R_s^{uz} saturation ratio of the unsaturated zone (dimensionless), and

R_i^{uz} retardation factor for radionuclide i in the unsaturated zone (dimensionless).

The unsaturated zone retardation factor is given by:

$$R_i^{uz} = 1 + \frac{\rho^{uz} \cdot K_{di}}{\theta^{uz}} \tag{10}$$

where

ρ^{uz} [g/cm^3] density of unsaturated zone,

K_{di} [cm^3/g] distribution coefficient for radionuclide i, and

θ^{uz} volumetric water content of the unsaturated zone (dimensionless).

Distributions coefficients are chosen identical to the contaminated zone (see Table VII).

The transport time given by Equation (9) will only be valid if the transport can be described as flow through a porous medium with the K_d concept being applicable. This will not be the case in all situations. For example, transport mechanisms like fracture flow or colloidal transport may lead to a substantially faster transport of the radionuclides through the unsaturated zone. Therefore, the low probability model does not take account of the presence of an unsaturated zone at all. This covers the situation where there is a direct contact of the contaminated zone with the groundwater aquifer as well as the presence of fast transport mechanisms through an unsaturated zone.

The exposure assessment assumes a private well downstream of the source. This well is conservatively assumed to be so close to the source that no dilution with groundwater that has not been impacted by the source takes place. The transport modeling of the radionuclides in the aquifer does not consider dispersion or diffusion effects. This is also a conservative assumption.

Within these assumptions the radionuclide concentration in the well water is given by the dilution with the groundwater volume U_{gw} flowing underneath the area of the contaminated zone:

$$U^{gw} = z^{gw} \cdot w^{gw} \cdot v^{gw} \cdot p^{gw} \tag{11}$$

where

z^{gw}	[m] thickness of aquifer,
w^{gw}	[m] width of contaminated zone perpendicular to flow rate of aquifer,
v^{gw}	[m/a] pore water velocity of groundwater, and
p^{gw}	effective porosity of aquifer (dimensionless).

From Equations (7), (8), (9), and (11) the concentration of the radionuclide i in the well water c_i^w is given by:

$$c_i^w = \frac{U^s}{U^{gw} + U^s} \cdot C_i^s \cdot e^{-\lambda_i \cdot t_i} \tag{12}$$

From this result the ingestion dose arising from the use of the well water as drinking water can be calculated.

For the assessment of the radiological impact of using this water for the irrigation of foodstuff grown in a private garden the transfer of the radionuclides from the water to the plants has to be considered. This is performed using the transfer factor given in the following equation derived in [14] assuming an overhead irrigation of the plants:

$$f_t = \frac{I_{rr} \cdot f_r \cdot T_f \cdot \left(1 - e^{-\lambda_w \cdot t_e}\right)}{Y_w \cdot \lambda_w} + \frac{I_{rr} \cdot \left(1 - f_r\right) \cdot f_l \cdot \left(1 - e^{-L_i \cdot t_e}\right)}{\rho^e \cdot L_i} \tag{13}$$

where (with default assumptions used according to [13])

I_{rr}	[m/a] irrigation rate,
f_r	fraction of deposited radionuclides retained on vegetation (0.25),

T_f foliage-to-food transfer coefficient (0.1 for fruit and non-leafy vegetables and 1 for leafy vegetables),

λ_w weathering removal constant for vegetation (20 a^{-1}),

t_e time of exposure during growing season (0.17 a for fruit and non-leafy vegetables and 0.25 a for leafy vegetables),

Y_w wet-weight crop yield (0.7 kg/m^2 for fruit and non-leafy vegetables and 1.5 kg/m^2) for leafy vegetables,

f_i root transfer factor for radionuclide i (dimensionless, see Section 4.3.3),

L_i [1/a] leach rate for radionuclide i according to Equation (4), and

ρ^e effective surface density of soil (225 kg/m^2)

The eventual discharge of the groundwater into a surface water body will also give rise to exposures if the surface water is used as drinking water or for irrigation. However, because of dilution effects doses will be lower in this case as compared to the private well. Therefore, it is not necessary to consider the use of surface water explicitly in the model. An additional exposure pathway arises, however, through the ingestion of fish from this surface water body. In analogy to Equation 12, the radionuclide concentration in the river water (c_i^r) is determined from the flow rate of the river (U^r)as:

$$c_i^r = \frac{U^s}{U^r + U^s} \cdot C_i^s \cdot e^{-\lambda_i t_i} \tag{14}$$

From this concentration the radionuclides transferred into fish can be calculated using transfer factors given in Table D.5 of [14].

4.3.4.2. *Conditions at model site*

For the realistic scenario, the amount of material present on the site is assumed as 25,000 m^3, and for the low probability case, a total volume of 100,000 m^3 is considered. The thickness of the contaminated zone is assumed to be 5 m in both cases. These assumptions are considered to cover all cases of material containing artificial radionuclides.[13]

In analogy to the foodstuff scenarios, a decay time before the start of the scenario of one year is assumed. During the scenario the decay depends on the migration time of the contaminant calculated according to Section 4.3.4.1. After the water reaches the well or the river, no further decay is considered because the dominating pathway is the direct ingestion of drinking water, which would occur instantaneously.

The infiltration rate is chosen as 0.2 m per year corresponding to the default assumptions in RESRAD. This value is sufficient for a moderate climate. In cases of wet regions and appropriate soil conditions, higher infiltration rates are possible. However, in this case flow rates of aquifers and surface water are to be expected to be higher too, so that the eventual dilution factor between the seepage from the contaminated material and ground or surface water should remain approximately the same.

[13] For material with elevated levels of natural radionuclides (NORM), higher masses are possible (e.g. in connection with mining operations). However, the models developed are not applied to natural radionuclides in this report.

For realistic assumptions an unsaturated zone of 2 m thickness is assumed between the contaminated zone and the top of the aquifer. The low probability scenario assumes direct contact of the contaminated zone and the aquifer.

The pore water velocity of the groundwater in the aquifer is taken as 1000 m per year in the realistic case and 500 m per year in the low probability case. Lower groundwater velocities and consequently a lower dilution may occur at some sites. However, within the overall context of the assumptions applied to the model site, this range is considered to be sufficiently conservative.

The groundwater in the private well is assumed to be used as drinking water and for irrigation purposes in a private garden. The irrigation rate is assumed as 0.2 m per year.

The river considered in the model is assumed to have a flow rate of 5 m^3/s, which is considered high enough to support a sufficient fish population to cover the annual fish consumption of the exposed persons.

The model calculations consider adults and children of the age group 1-2a in accordance with the ingestion scenarios presented in Section 4.3.3. Dietary parameters are also chosen consistent with these scenarios. The model presented requires input parameters for the consumption of

- drinking water;
- leafy vegetables;
- non-leafy vegetables and fruits; and
- fish.

Safety Report 19 [13] provides only aggregate numbers on consumption (410 kg per year of fruits, vegetables, and grain for adults). Since this is not sufficient for the models developed here, the ingestion quantities are based on detailed parameters provided in the German Radiation Protection Ordinance [15], giving ingestion quantities for average cases and for low probability cases (approximately corresponding to 95% percentiles). These parameters are used for the realistic and the low probability scenarios, respectively. They are shown in Table VIII. Considering that the overall consumption given in [13] of 410 kg per year also includes grain, the assumptions are consistent.

TABLE VIII. INGESTION PARAMETERS

Type	consumption of children (1-2a) [kg/a]		consumption of adults (>17 a) [kg/a]	
	realistic	**low prob.**	**realistic**	**low prob.**
drinking water	100	200	350	700
leafy vegetables	6	18	13	39
non-leafy vegetables	17	51	40	120
fruits	45	135	35	105
total vegetables and fruits	68	204	88	264
fish	0.6	3	1.5	7.5

For the realistic scenario, it is assumed that 25 % of the annual consumption of drinking water and foodstuff are affected by the radionuclides from the contaminated material and that the remainder is obtained from other sources. In the low probability scenario, the assumption is used that the total consumption of drinking water and foodstuff as specified above is affected from the contaminated material.

A summary of the site parameters used is presented in Table IX.

1026 TABLE IX. SITE PARAMETERS FOR WATER PATHWAY MODEL

Parameter	Unit	realistic	low probability
Contaminated zone			
decay time before scenario	a	1	1
area of contaminated zone	m^2	5000	20000
thickness of contaminated zone	m	5.00	5.00
density of contaminated area	g/cm^3	1.80	1.80
infiltration rate	m/a	0.20	0.20
irrigation rate	m/a	0.20	0.20
seepage through contaminated zone (calculated)	m^3/a	1000	4000
total porosity of contaminated area		0.40	0.40
saturated hydraulic conductivity	m/a	5000	5000
volumetric water content		0.16	0.16
Unsaturated zone			
thickness of unsaturated zone	m	2.00	0.00
density of unsaturated zone	g/cm^3	1.80	1.80
total porosity of unsaturated zone		0.40	0.40
effective porosity of unsaturated zone		0.20	0,.20
volumetric water content		0.16	0.16
Groundwater aquifer			
thickness of aquifer	m	5.00	5.00
width of contaminated zone perpendicular to aquifer	m	100	100
groundwater pore water velocity	m/a	1000	500
effective porosity of aquifer		0.25	0.25
flow rate of aquifer (calculated)	m^3/a	1.25E+05	6.25E+04
dilution factor between seepage and groundwater (calculated)		7.94E-03	6.02E-02
Surface water			
flow rate of river	m^3/s	5.00	5.00
dilution factor between seepage and river (calculated)		6.34E-06	2.54E-05
Irrigation parameter			
length of growing season for non-leafy vegetables	a	0.17	0.17
length of growing season for leafy vegetables	a	0.25	0.25
weathering removal constant for vegetation	1/a	20	20
fraction of radionuclides retained on vegetation		0.25	0.25
foliage-to-food transfer coefficient for non-leafy vegetables		0.1	0.1
foliage-to-food transfer coefficient for leafy vegetables		1	1
effective surface density of soil	kg/m^2	225	225
wet-weight crop yield for non-leafy vegetables	kg/m^2	0.7	0.7
wet-weight crop yield for leafy vegetables	kg/m^2	1.5	1.5
Ingestion parameter			
consumption of drinking water (1-2a)	kg/a	100	200
consumption of drinking water (> 17a)	kg/a	350	700
consumption of non-leafy vegetables (1-2a)	kg/a	17	51
consumption of non-leafy vegetables (> 17a)	kg/a	40	120
consumption of leafy vegetables (1-2a)	kg/a	6	18
consumption of leafy vegetables (> 17a)	kg/a	13	39
consumption of fish (1-2a)	kg/a	0.6	3
consumption of fish (> 17a)	kg/a	1.5	7.5
fraction of contaminated drinking water consumed		0.25	1
fraction of contaminated vegetables consumed		0.25	1
fraction of contaminated fish consumed		0.25	1

4.3.4.3. Radionuclides considered

Modeling is performed only for radionuclides with a half-life greater than 0.5 years because radionuclides with a shorter half-life will not contribute significantly to the water pathway doses. Ingestion doses incurred by these short-lived radionuclides will be dominated by the ingestion scenarios and/or other pathways presented in Section 4.3.3

The ingrowth of daughter nuclides is considered according to Section 3.2. However, for the water pathway it has to be considered that the leachability and groundwater mobility of a daughter nuclide may be higher than those of its parent nuclides. To account for this effect the following approach is used:

- Daughter nuclides with a half-life less then 0.05 years are treated in equilibrium with their parent nuclides in the water and foodstuff consumed because the processes relevant for the migration of the radionuclides and the plant uptake are slow enough to at least nearly achieve a radioactive equilibrium in this case.

- Longer-lived daughter nuclides are modeled independently and their dose contribution is added to the dose incurred by the parent nuclide. The ingrowth of daughter nuclides is considered in analogy to the other pathways using the model presented in Chapter 2.

4.3.4.4. Time scales

In the realistic scenario, an unsaturated zone is assumed to be present between the contaminated material and the groundwater aquifer. In this situation, migration processes of contaminants with a high K_d value are very slow. The time span between the deposition of the material and their arrival in the well or the river may be hundreds or even thousands of years. The consideration of such long-term exposures may be seen as contradicting the assumption concerning the ingrowth of daughter nuclides (see Chapter 2), where a period of 100 years has been used.

The examination of the results for those nuclides dominated by the water pathway within the realistic scenario showed, however, that the resulting activity concentration levels do not change if a cut-off after 100 years is applied. Therefore, the question of which time scale to used is not of practical relevance in this case.

4.3.4.5. Discussion of results

The results from the water pathway model presented in Appendix II show that only for some radionuclides the water pathway dominates the activity concentration level. These are mobile nuclides with a considerably long half-live, high ingestion dose factors and low external dose factors.

The exposures from these nuclides over the water pathway in real situations will depend on actual site conditions. As discussed already, the model used for the derivation of activity concentration levels does not cover all potentially occurring individual site parameters. Nevertheless, the results are considered to be sufficiently conservative to cover the vast majority of cases:

- The volumes of contaminated material considered in the model are quite high.
- The exposure situation of residents using the contaminated groundwater downstream of the landfill without any additional dilution corresponds to unfavorable conditions.
- The model used does not take account of effects like dispersion that would lead to lower exposures.
- An intensive use of the contaminated water is assumed for drinking water and for irrigation purposes.

On this basis, the derived activity concentration levels are considered appropriate also for sites where some of the relevant site factors are more unfavorable as assumed here.

4.3.5. Skin Contamination

Skin contamination by dust containing radionuclides can only occur with some significance at workplaces in dusty environments. Those workplaces may be at a scrap yard or metal recycling facility where metal is segmented or at a landfill site where workers come into contact with the dumped material.

The skin dose is calculated according to equation (15):

$$E_{skin,C} = \dot{e}_{skin} \cdot t_e \cdot L_{dust} \cdot f_d \cdot f_c \cdot \rho \cdot e^{-\lambda t_1} \frac{1-e^{-\lambda \cdot t_2}}{\lambda \cdot t_2} \qquad (15)$$

where

$E_{skin,C}$ [(μSv/a)/(Bq/g)] skin equivalent dose in a year from skin contamination with beta and gamma emitters per unit activity concentration in the material,
$\dot{e}_{skin}$ [(μSv/h)/(Bq/cm²)] sum of skin equivalent dose rate coefficients for beta emitters (4 mg/cm² skin density) and for gamma emitters [6] per surface specific unit activity,
t_e [h/a] exposure time (time during which the skin is contaminated),
L_{dust} [cm] layer thickness of dust loading on the skin,
f_d [-] dilution factor,
f_c [-] concentration factor,
ρ [g/cm³] density of surface layer,
λ [1/a] radioactive decay constant,
t_1 [a] decay time before start of scenario, and
t_2 [a] decay time during scenario.

Contamination of the skin is assumed to occur during the entire working year (1800 h/a). The layer thickness of the dust is assumed to 100 μm (0.01 cm) which is a thickness that would not be significantly disturbing while working and therefore would be removed by the worker only at the end of his working time.

No dilution has been assumed. This is a conservative assumption, but it is consistent with the low probability parameter used for the landfill scenario. In order to account for a higher activity concentration in the fine fraction a concentration factor 2 is used (see Section 3.7). As the material causing skin contamination might always be recently cleared, no decay before or during the scenario is assumed. The density of the dust on the skin is set to 1.5 g/cm³.

Parameter values provided in Table X.

TABLE X. SCENARIO PARAMETERS FOR SKIN CONTAMINATION

Parameter	Unit	Scenario SKIN
Exposure time (t_e)	h/a	1800
Layer thickness (L_{dust})	cm	0.01
Dust density (ρ)	g/cm³	1.5
Dilution factor (f_d)	[-]	1
Concentration factor (f_c)	[-]	2
Decay time before scenario (t_1)	d	0
Decay time during scenario (t_2)	d	0
Dose rate coefficient ($\dot{e}_{skin}$)	(µSv/h)/(Bq/cm²)	depending on radionuclide

The parameter values defined are in total quite conservative. Therefore, the estimation of the skin dose has to be seen as a low probability scenario. The resulting dose therefore could be converted into an effective dose with the skin weighting factor of 0.01 and the fraction of the total skin being exposed (choosing this fraction as 0.1 would correspond to an exposure of about 2000 cm^2, approximately equivalent to the forearms and hands). The resulting effective dose could then be compared to the 1 mSv/a dose criterion.

However, this would not yield a compliance with the skin dose limit of 50 mSv/a, corresponding only to an effective dose of 0.05 mSv/a with an assumption of an uncovered skin area of 2000 cm^2. Therefore, it is necessary to use the BSS dose limit for the skin of 50 mSv/a as the criterion for the assessment of the skin dose. This limit compared to the equivalenty dose of the exposed skin area (for which size no assumptions are required) is given by equation 15.

4.4. FLUIDS

Liquids of concern generally carry radionuclides in a water-borne or organic-liquid-borne form. Radionuclides can be in the form of suspended solids or dissolved in solution from solids, liquids or gases. Typically, liquids can be considered on the same basis as solids for the external exposure pathway. However, ingestion and inhalation exposures require consideration of likely mechanisms of intake, for example, vaporization, drinking, etc. These mechanisms apply, in turn, to the physical-chemical properties of the specific liquid and the processes commonly associated with it. Processes that tend to concentrate the small concentrations are, for example, water processing or incineration and recycle of organic liquids. These processes can lead to concentration exceeding the activity concentration levels in filters, sludges, resins, residues, ashes, and combustion gases. Finally, the volume of liquids is an unstable quantity, strongly depending on the ambient physical conditions, especially temperature. In particular, liquids evaporate and concentrate as the temperature rises. Therefore, it is necessary to adopt exclusion levels that cannot be considered as inappropriate when the ambient physical conditions are modified. For this reason, the following is recommended:

- For pure liquids, in the case the radionuclide is part of the molecule of the liquid, the concentration level level applies to the liquid as such.

- For dissolved radionuclides, i.e., in case of solutions, the activity concentration level applies to the solid residue after evaporation of the liquid or, at least, to the best concentrate of the solution.

4.5. GASES

Calculations were not undertaken explicitly for gases. However, scenarios representing exposure from gas cylinders were taken into account in deriving the exemption concentrations for Schedule I in the BSS [2]. These calculations took account of exposure from a limited volume of gas whereas exposure from larger quantities of gas would, in principle, occur during transport or storage of gas cylinders. These exposures were taken into account in establishing exempt levels for purposes of the Transport Regulations [16] and it was decided to adopt the Schedule I values of the BSS [2] into the Transport Regulations [16]. Therefore it was considered appropriate to use the Schedule I values for the activity concentration levels.

5. DEVELOPMENT OF ACTIVITY CONCENTRATION LEVELS FOR NATURALLY OCCURRING RADIONUCLIDES

The objective in defining material that contains naturally occurring radioactive substances that should be regulated is to identify that material of significant radiological risk where regulation can achieve real improvements in protection. At the same time, the number of materials involved should not be so great as to make regulation essentially unmanageable. The application of a dose criterion of 10 μSv/a is not practical. In selecting levels for material that contains naturally occurring radioactive material (NORM), a major issue is that high levels that would exclude the majority of natural material in the environment would also allow a number of situations such as release of phosphate slags to be excluded without further considerations. Conversely, selecting a low value would trigger an unnecessary application of the BSS [2]. Therefore, the levels should be derived from consideration of the worldwide distribution of concentrations of naturally occurring radionuclides.

In considering activity concentration levels for naturally occurring radionuclides, the intention is to exclude from regulation virtually all soils, but not exclude from regulation ores, mineral sands, industrial residues and wastes which are recognized as having significant activity considerations.

Tables XI present data from UNSCEAR for concentrations of naturally occurring radionuclides in normal soil material. The values for ^{238}U and ^{232}Th are for 'head of chain' assuming that daughters are in equilibrium.

1173 TABLE XI: NATURAL RADIONUCLIDES IN SOIL [8]

Region/Country	Population in 1996 (10^6)	Concentration in soil (Bq/kg)							
		^{40}K		^{238}U		^{226}Ra		^{232}Th	
		Mean	Range	Mean	Range	Mean	Range	Mean	Range
Africa									
Algeria	28.78	370	66-1,150	30	2-110	50	5-180	25	2-140
Egypt	63.27	320	29-650	37	6-120	17	5-64	18	2-96
North America									
Costa Rica	3.50	140	6-380	46	11-130	46	11-130	11	1-42
United States [M7]	269.4	370	100-700	35	4-140	40	8-160	35	4-130
South America									
Argentina	35.22	650	540-750						
East Asia									
Bangladesh	120.1	350	130-610			34	21-43		
China [P16,Z5]	123.2	440	9-1,800	33	2-690	32	2-440	41	1-360
- Hong Kong SAR [W12]	6.19	530	80-1,100	84	25-130	59	20-110	95	16-200
India	944.6	400	38-760	29	7-81	29	7-81	64	14-160
Japan [M5	125.4	310	15-990	29	2-59	33	6-98	28	2-88
Kazakhstan	16.82	300	100-1,200	37	12-120	35	12-120	60	10-220
Korea, Rep. of	45.31	670	17-1,500						
Malaysia	20.58	310	170-430	66	49-86	67	38-94	82	63-110
Thailand	58.70	230	7-712	114	3-370	48	11-78	51	7-120
West Asia									
Armenia	3.64	360	310-420	46	20-78	51	32-77	30	29-60
Iran (Islamic Rep. of)	69.98	640	250-980			28	8-55	22	5-42
Syrian Arab Republic	14.57	270	87-780	23	10-64	20	13-32	20	10-32
North Europe									
Denmark	5.24	460	240-610			17	9-29	19	8-30
Estonia	1.47	510	140-1,120			35	6-310	27	5-59
Lithuania	3.73	600	350-850	16	3-30			25	9-46
Norway	4.35	850		50		50		45	
Sweden	8.82	780	560-1,150			42	12-170	42	14-94
West Europe									
Belgium	10.16	380	70-900			26	5-50	27	5-50
Germany	81.92		40-1,340		11-330		5-200		7-134
Ireland [M6]	3.55	350	40-800	37	8-120	60	10-200	26	3-60
Luxembourg	0.41	620	80-1,800			35	6-52	50	7-70
Netherlands [K2]	15.58		120-730		5-53	23	6-63		8-77
Switzerland	7.22	370	40-1,000	40	10-150	40	10-900	25	4-70
United Kingdom [B2]	58.14		0-3,200		2-330	37			1-180
East Europe									
Bulgaria	8.47	400	40-800	40	8-190	45	12-210	30	7-160
Hungary	10.05	370	79-570	29	12-66	33	14-76	28	12-45
Poland [K2]	38.60	410	110-970	26	5-120	26	5-120	21	4-77
Romania [K2]	22.66	490	250-1,100	32	8-60	32	8-60	38	11-75
Russian Federation	148.1	520	100-1,400	19	0-67	27	1-76	30	2-79
Slovakia	5.35	520	200-1,380	32	15-130	32	12-120	38	12-80
South Europe									
Albania	3.40	360	15-1,115	23	6-96			24	4-160
Croatia	4.50	490	140-710	110	83-180	54	21-77	45	12-65
Cyprus	0.76	140	0-670			17	0-120		
Greece	10.49	360	12-1,570	25	1-240	25	8-65	21	1-190
Portugal	9.81	840	220-1,230	49	26-82	44	2-210	51	22-100
Slovenia	1.92	370	15-1,410			41	6-250	35	2-90
Spain	39.67	470	25-1,650			32		33	2-210
Median		400	140-850	35	16-110	35	17-60	30	11-64
Population-weighted average		420		33		32		45	

Table XII shows typical activity concentrations in various ores and mineral sands that are used in industrial processes.

TABLE XII: ACTIVITY CONCENTRATIONS IN ORES AND MINERAL SANDS IN (Bq/Kg)

Ore/mineral sand	^{238}U	^{226}Ra	^{232}Th	^{40}K
Phosphate ore	30-5000	30-5000	25-2000	3-200
Monazite sand	370		1800	160
Monazite	6000-40000		8000-900000	
Bastnaesite			400	
Xenotime	3500-500000		180000	
Thorianite			2500000-5500000	
Tin ores	1000		300	
Pyrochlore	10000		80000	
Titanium ores	70-9000		70-9000	
Ilmenite	2000		1000	
Zircon sands	10000	3000-4000	10000	
Bauxite	400-600		400-600	
Coal		soil concentrations typically		
Iron ore	15			

Residues from industrial processes may have elevated levels of natural radionuclides. Phosphogypsum, a by-product from phosphate rock processing can have activity concentrations of ^{226}Ra up to 3 Bq/g. Residues from ore processing industries generally can have elevated levels of natural radionuclides but if these industries are subject to regulation because of the activity concentration in the feedstock, this will not be an issue. Examples are given in Table XIII.

Although not explicitly considered, elevated levels of isotopes of polonium and lead can also occur in residues from industrial processes. For example, tin rich residues from metal extraction processes can contain up to 10 Bq/g of ^{210}Pb and ^{210}Po. Filter dusts from metal processing can also contain elevated concentrations of ^{210}Po due to volatilization during heating. For example, concentrations of ^{210}Po of up to 200 Bq/g have been observed in collected fumes from tin smelting.

TABLE XIII: ACTIVITY CONCENTRATIONS IN INDUSTRIAL RESIDUES AND WASTES IN (Bq/Kg)

Material	^{238}U	^{226}Ra	^{232}Th	^{40}K
Tin slag		1000-4000	230-340	
Oil scale (old process)		up to 4000		
Oil scale (new process, scale inhibition techniques)		40-100		
Rare earth extraction byproducts		3000-450000		
T_iO_2 production residues from ilmenite		up to 400,000		
Monazite processing residues		up to 450,000		
Zircon processing				
residues		2000-50,000		
sludge		200-7000		
Copper slag		500-2000		
Aluminium processing sludge	260-540	150-330		
Fly ash	400			
Blast furnace slag from steel production	150		150	

Some products from processing of natural radioactive materials may in themselves be radioactive. Examples are given in Table XIV. The main issues appear to surround thorium-containing materials.

TABLE XIV: PRODUCTS FROM PROCESSING NATURAL MATERIALS IN (Bq/Kg)

	^{238}U	^{226}Ra	^{232}Th	^{40}K
Phosphate fertilisers	300-3000	200-1000		Up to 6000
Thorium				
Thoriated welding electrodes			up to ~100,000	
Special alloys (jet engines)			35,000	
Gas mantles			500,000	
Thoriated glass			200,000	
Titanium oxide pigment			30,000	
Construction materials containing fly-ash		70-170	70-170	

Unmodified concentrations of radionuclides in most raw materials are deemed to be excluded from the Standards by the BSS [1]. In this Report, it has been taken to mean virtually all unmodified soils, but not ores or mineral sands that are recognized as having significant activity concentrations. Activity concentration levels have been chosen as the optimum boundary between, on the one hand, the ubiquitous unmodified soil concentrations (Table XI) and, on the other hand, activity concentrations in ores, mineral sands, industrial residues and wastes (Tables XII, XIII, and XIV) is judged to be 0.5 Bq/g for naturally occurring radionuclides. The only exceptions are ^{40}K where the level is 5 Bq/g and ^{235}U where the level is 0.05 Bq/g based on the natural ratio between the two decay chains of ^{238}U and ^{235}U.

It can be seen that these levels are around a factor of 10 higher than the population-weighted average activity concentrations in Table XI, and are therefore unlikely to result in an unwarranted regulatory burden. Scenario-based calculations done by the European Union demonstrate convergence with these numbers.

For indoor radon in air, the "action levels" established in the BSS [2], namely 1000 Bq/m^3 for work places and within the range of 200-600 Bq/m^3 for dwellings, shall apply.

6. ACTIVITY CONCENTRATION LEVELS

Table XV provides the activity concentration levels for artificial and Table XVI provides the levels for natural radionuclides.

TABLE XV. ACTIVITY CONCENTRATION LEVELS FOR ARTIFICIAL RADIONUCLIDES

Radionuclide	Concentration Level (Bq/g)	
H-3	100	
Be-7	10	
C-14	1	
F-18	10	*
Na-22	0.1	
Na-24	1	*
Si-31	1000	*
P-32	1000	
P-33	1000	
S-35	100	
Cl-36	1	
Cl-38	10	*
K-42	100	
K-43	10	*
Ca-45	100	
Ca-47	10	
Sc-46	0.1	
Sc-47	100	
Sc-48	1	
V-48	1	
Cr-51	100	
Mn-51	10	*
Mn-52	1	
Mn-52m	10	*
Mn-53	100	
Mn-54	0.1	
Mn-56	10	*
Fe-52	10	*
Fe-55	1000	
Fe-59	1	
Co-55	10	*
Co-56	0.1	
Co-57	1	
Co-58	1	
Co-58m	10000	*
Co-60	0.1	
Co-60m	1000	*
Co-61	100	*
Co-62m	10	*
Ni-59	100	
Ni-63	100	
Ni-65	10	*
Cu-64	100	*
Zn-65	0.1	
Zn-69	1000	*
Zn-69m	10	*
Ga-72	10	*
Ge-71	10000	
As-73	1000	
As-74	10	*
As-76	10	*
As-77	1000	
Se-75	1	
Br-82	1	

Radionuclide	Concentration Level (Bq/g)	
Rb-86	100	
Sr-85	1	
Sr-85m	100	*
Sr-87m	100	*
Sr-89	1000	
Sr-90	1	
Sr-91	10	*
Sr-92	10	*
Y-90	1000	
Y-91	100	
Y-91m	100	*
Y-92	100	*
Y-93	100	*
Zr-93	10	*
Zr-95	1	
Zr-97	10	*
Nb-93m	10	
Nb-94	0.1	
Nb-95	10	
Nb-97	10	*
Nb-98	10	*
Mo-90	10	*
Mo-93	10	
Mo-99	10	
Mo-101	10	*
Tc-96	1	
Tc-96m	1000	*
Tc-97	10	
Tc-97m	100	
Tc-99	1	
Tc-99m	100	*
Ru-97	10	
Ru-103	10	
Ru-105	10	*
Ru-106	0.1	
Rh-103m	10000	*
Rh-105	100	
Pd-103	1000	
Pd-109	100	
Ag-105	10	
Ag-110m	0.1	
Ag-111	100	
Cd-109	1	
Cd-115	10	
Cd-115m	100	
In-111	10	
In-113m	100	*
In-114m	10	
In-115m	100	*
Sn-113	1	
Sn-125	10	
Sb-122	10	
Sb-124	1	
Sb-125	0.1	

Radionuclide	Concentration Level (Bq/g)	
Te-123m	1	
Te-125m	1000	
Te-127	1000	
Te-127m	10	
Te-129	100	*
Te-129m	100	
Te-131	100	*
Te-131m	10	
Te-132	1	
Te-133	10	*
Te-133m	10	*
Te-134	10	*
I-123	10	
I-125	1000	
I-126	10	
I-129	0.1	
I-130	10	*
I-131	10	
I-132	10	*
I-133	10	*
I-134	10	*
I-135	10	*
Cs-129	10	
Cs-131	1000	
Cs-132	10	
Cs-134	0.1	
Cs-134m	10	*
Cs-135	100	
Cs-136	1	
Cs-137	0.1	
Cs-138	10	*
Ba-131	10	
Ba-140	1	
La-140	1	
Ce-139	1	
Ce-141	100	
Ce-143	10	
Ce-144	10	
Pr-142	100	*
Pr-143	1000	
Nd-147	100	
Nd-149	100	*
Pm-147	1000	
Pm-149	1000	
Sm-151	10000	
Sm-153	100	
Eu-152	0.1	
Eu-152m	100	*
Eu-154	0.1	
Eu-155	1	
Gd-153	10	
Gd-159	100	*
Tb-160	1	
Dy-165	1000	*

TABLE XV. ACTIVITY CONCENTRATION LEVELS FOR ARTIFICIAL RADIONUCLIDES

Radionuclide	Concentration Level (Bq/g)	
Dy-166	100	
Ho-166	100	
Er-169	1000	
Er-171	100	*
Tm-170	100	
Tm-171	1000	
Yb-175	100	
Lu-177	100	
Hf-181	10	
Ta-182	0.1	
W-181	10	
W-185	1000	
W-187	101000	
Re-186	1000	
Re-188	100	*
Os-185	1	
Os-191	100	
Os-191m	1000	*
Os-193	100	
Ir-190	1	
Ir-192	1	
Ir-194	100	*
Pt-191	10	
Pt-193m	1000	
Pt-197	1000	*
Pt-197m	100	*
Au-198	10	
Au-199	100	
Hg-197	100	
Hg-197m	100	
Hg-203	10	
Tl-200	10	
Tl-201	100	
Tl-202	10	
Tl-204	1	
Pb-203	10	
Bi-206	1	
Bi-207	0.1	
Po-203	10	*
Po-205	10	*
Po-207	10	*
At-211	1000	1217
Ra-225	10	1218
Ra-227	100	
Th-226	1000	
Th-229	0.1	
Pa-230	10	
Pa-233	10	
U-230	10	
U-231	100	
U-232	0.1	
U-233	10	
U-236	10	
U-237	100	

Radionuclide	Concentration Level (Bq/g)	
U-239	100	*
U-240	100	*
Np-237	1	
Np-239	100	
Np-240	10	*
Pu-234	100	*
Pu-235	100	*
Pu-236	1	
Pu-237	100	
Pu-238	1	
Pu-239	1	
Pu-240	1	
Pu-241	100	
Pu-242	1	
Pu-243	1000	*
Pu-244	0.1	
Am-241	1	
Am-242	1000	*
Am-242m	1	
Am-243	1	
Cm-242	10	
Cm-243	1	
Cm-244	10	
Cm-245	1	
Cm-246	1	
Cm-247	0.1	
Cm-248	1	
Bk-249	100	
Cf-246	1000	
Cf-248	10	
Cf-249	0.1	
Cf-250	1	
Cf-251	1	
Cf-252	10	
Cf-253	100	
Cf-254	1	
Es-253	100	
Es-254	0.1	
Es-254m	10	
Fm-254	10000	*
Fm-255	100	*

* indicates half life less than 1 day

TABLE XVI. ACTIVITY CONCENTRATION LEVELS FOR RADIONUCLIDES OF NATURAL ORIGIN

Radionuclide	Concentration Level (Bq/g)
Radionuclides in the ^{235}U decay series	0.05
^{40}K	5
All other naturally occuring radionuclides	0.5

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