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Clearance levels for radionuclides in solid materials

Application of exemption principles

Interim report for comment

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CLEARANCE LEVELS FOR RADIONUCLIDES IN SOLID MATERIALS:
APPLICATION OF EXEMPTION PRINCIPLES
INTERIM REPORT FOR COMMENT

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FOREWORD

Radioactive waste is produced during the generation of nuclear power and the use of radioactive materials in industry, research and medicine. The importance of the safe management of radioactive waste for the protection of human health and the environment has long been recognized.

This report proposes levels of radionuclides in solid materials below which regulatory control may be relinquished on the grounds that the associated radiation hazards are trivial. The radiological basis for the guidance is the international consensus on principles for the exemption of radiation sources and practices from regulatory control reached in 1988 and published in IAEA Safety Series No. 89. The levels are intended as international reference values. They may find use in agreements between Member States on the transboundary movement of materials for which the final destination and final use cannot be known in advance.

The levels may be seen as those below which release from regulatory control is 'automatic' without further consideration being needed. Release from regulatory control may, of course, be allowed under other conditions; regulatory authorities may decide, on the basis of a generic or site specific optimization subject to dose constraints, to select other, less restrictive, release levels. This optimization process includes consideration of factors other than those associated with radiation protection, for example, those concerned with the health, social, environmental and economic benefits and risks of implementing the practice.

Even when the radiological basis for release remains the same as that used in this report, that is, trivial radiation risk, other types of clearance are possible. When the intended use and destination of the materials for release can be clearly identified, more specific values can be derived. However, the clearance will be conditional on the material being used in the way specified or being sent to the prescribed destination after release (conditional clearance). This is in contrast with the clearance levels contained in the present report, which have been derived assuming that the final destination and use of the released materials are not known in advance (unconditional clearance). The most likely uses and destinations for material being released from regulatory control are recycling, reuse and near surface disposal. International consensus on appropriate conditional clearance levels for release via each of these routes would also be beneficial.

This report is intended for eventual publication in the IAEA's Radioactive Waste Safety Standards (RADWASS). For the present, it is being issued on an interim basis and will be revised after about three years to take account of comments received, of experience gained in its application and of any new information which may emerge during this period of time.

The report was developed with the help of consultants and through two Advisory Group meetings held in 1992. Since then, the text has been revised following its circulation to experts in Member States on two occasions during 1993 and 1994.

The final revision of the report was made after review by the Extended International Radioactive Waste Management Advisory Committee (INWAC) in May 1995. G. Linsley of the Division of Nuclear Fuel Cycle and Waste Management was the responsible officer at the IAEA.

EDITORIAL NOTE

In preparing this publication for press, staff of the IAEA have made up the pages from the original manuscript(s). The views expressed do not necessarily reflect those of the governments of the nominating Member States or of the nominating organizations.

Throughout the text names of Member States are retained as they were when the text was compiled.

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CONTENTS

1. INTRODUCTION

BACKGROUND

101. Many uses of radioactive materials are beneficial to mankind, for example, in electric power production, in cancer treatment, and in medical diagnosis. However, radioactive materials are also potentially harmful to health and their use must therefore be regulated. The accepted regulatory approach is based on a system of notification and authorization such as that described in the International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources [1]. Some types of sources of ionizing radiation are not subject to regulatory control, either because they are not amenable to such control (e.g. cosmic rays) and are therefore excluded from the regulatory process, or because they present such a low risk that control by regulatory processes would be a waste of resources. In the latter case, two categories can be distinguished:

(a) Radiation sources which never enter the regulatory control regime, i.e. control is not imposed, and
(b) Radiation sources which are released from regulatory control, i.e. control is removed.

Sources in the first category are excluded from regulatory control by a process called exemption. Exempted sources typically include small sources of radiation such as tracers used in research, calibration sources and some consumer products containing small sources or low levels of activity per unit mass. The corresponding levels of activity or activity concentration are called exemption levels. In the second category the release of sources from control is called clearance. Cleared sources include waste materials and materials for recycling from within the nuclear fuel cycle and wastes from other regulated facilities such as hospitals, research laboratories and industry. The amounts of material involved in clearances can be substantial and are generally greater than those involved in exemptions of man-made sources. The corresponding levels of activity or activity concentration are called clearance levels. The derivation of these levels is the concern of the present document. The distinction between exemption and clearance has been made only recently and so in the literature terms such as 'exemption' and 'exempt' are often used in circumstances where terms such as 'clearance' and 'cleared' would currently be used.

102. Regulations containing specifications for exemption or clearance levels relevant to the application of radionuclides in hospitals, research institutes and industry exist in many industrialized countries but the radiological basis for the values in national regulations is often not well defined and the values differ from one country to another. A few countries have generic rules or regulations relevant to the nuclear

fuel cycle which include specifications for clearance, but in most cases clearance from regulations by safety authorities has been on the basis of case by case evaluations.

103. This interim report is eventually intended to be published within the RADWASS series, the lead document of which is the Safety Fundamentals [2], in which concepts of exemption and clearance of materials from regulatory control are introduced. The clearance levels presented in this report are based on the radiological principles for exemption given in IAEA Safety Series No. 89 [3]. The values are derived by drawing on other IAEA documents in which clearance levels have been calculated, i.e. Safety Series No. 111-P-1.1 [4] and IAEA-TECDOC-401 [5] as well as on national documents. The use of the concepts of exemption and clearance in this publication is consistent with their use in Refs [1, 6].

OBJECTIVE

104. The objective of this report is to provide a set of nuclide specific clearance levels applying to solid materials irrespective of the use to which those materials are put or of their destination after control has been relinquished. They are called 'unconditional clearance levels'. The basis for these levels is the guidance on radiological principles for exemption in IAEA Safety Series No. 89 [3] and the Basic Safety Standards [1].

105. One intended application of the unconditional clearance levels is related to international trade, for example the import and export of scrap metals of mixed types, or whose end users are not well defined. Either conditional or unconditional clearance levels could form the basis of an international agreement on levels that would be acceptable for materials being transferred between countries. The need for such clearance levels is identified in the Code of Practice on the International Transboundary Movement of Radioactive Waste which was adopted by the General Conference of the IAEA in September 1990 [7].

SCOPE

106. This report provides only values for the unconditional clearance of solid materials from regulatory control. It addresses materials that can be potentially moved out of the originating facility and recycled, reused or disposed of within the Member State or in another State or country without restriction. The following materials are not included in the term 'solid material' used in this publication: sealed

radiation sources, contaminated land and buildings[1], and materials which can be used as food products. This report does not apply to the disposal of solid materials in the marine environment.

The unconditional clearance levels were derived for application to materials originating from controlled practices; they do not necessarily apply to materials resulting from uncontrolled activities, such as accidents or previous unregulated practices. Actions intended to reduce exposures in these cases are termed 'interventions' [1]. The guidance does not apply to naturally occurring radioactive materials at unmodified concentrations. Exposure to such sources is normally excluded from regulatory consideration [1]. Where the concentrations of natural radionuclides are enhanced, leading to a potential for significant human exposures, it may be judged that intervention is required [1]. This is discussed further in paras 504–507.

Conditional clearance values that could serve as a basis for international trade may be established by agreement between national authorities or by the appropriate international organizations.

STRUCTURE

107. Following this introduction, the radiological basis for deriving clearance levels is set out in Section 2 and the important distinction between unconditional and conditional clearance is explained. Section 3 describes the method by which the unconditional clearance levels are derived. The derived values are presented and their application and interpretation is discussed. Section 3 draws on the more detailed analysis of national and international studies contained in Appendix I. Section 4 contains a discussion of approaches to verifying that the clearance levels have been complied with. In Section 5 regulatory aspects of clearance are discussed. In both Sections 4 and 5 the problem of applying the clearance levels to naturally occurring radionuclides is discussed.

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[1] Although renovation and reuse of contaminated buildings was considered in some of the studies reviewed in establishing the unconditional clearance levels, the data reviewed are insufficient to develop firm guidance and the values are not generally recommended for use in this context. The values can, however, be applied to materials removed from contaminated buildings.

2. EXEMPTION AND CLEARANCE PRINCIPLES

INTRODUCTION

201. The concept of exemption was pursued for several years through IAEA working groups under a general heading of 'de minimis', mainly in relation to radioactive waste disposal in marine and terrestrial environments [8, 9]. In 1984, in co-operation with the Nuclear Energy Agency of the OECD (OECD/NEA), a new programme was started with the specific objectives of developing principles for exemption of radiation sources and practices from regulatory control and of developing guidance on the application of the principles to practical problems. This culminated in 1988 with the publication of Safety Series No. 89 [3], which contains results representing an international consensus on the subject.

202. The IAEA has subsequently issued publications dealing with the problems involved in applying exemption principles to the recycling and reuse of materials from the nuclear industry [4] and to the wastes arising from the use of radionuclides in hospitals and research laboratories [10]. In both these publications, methods for deriving exemption levels (now to be called clearance levels) are described and example values presented. Various national and regional groups have also been studying exemption principles and their application; the guidance given in 1988 by the Commission of the European Communities on clearance levels for the recycling of steel scrap is particularly relevant [11]. Thus a substantial base of experience now exists and clearance levels have been proposed in various countries for application to the most important low activity level waste streams from the nuclear fuel cycle and from the application of radioisotopes in medicine, research and industry.

203. Most of the above studies were concerned with the clearance of materials whose fate is predetermined, for example by recycling or by disposal via incineration, and therefore the clearance in such cases is conditional. National regulatory authorities may issue such clearances with clauses limiting the scope of the relevant practice. There is, however, also a need for unconditional clearance. This is because it is not always possible to be sure of the fate of a material once it is cleared and control is lost; for example, it may be disposed of, but it may just as easily be recycled. An unconditional clearance should take account of any reasonably possible scenario by which humans could be exposed to radiation after control of a material has been relinquished. In the event of transboundary movements of released materials such as scrap and waste, guidance on internationally applicable unconditional clearance levels would be needed.

INTERNATIONAL GUIDANCE ON EXEMPTION AND CLEARANCE PRINCIPLES

204. Sources and practices involving the potential exposure of humans to ionizing radiation are normally controlled by a system of notification and authorization as exemplified in Ref. [1]. Users are required to notify the regulatory authority of their intentions and to apply for an authorization in the form of a registration or a licence. However, specific sources and practices may be exempted from the control, if, following an analysis of the health hazards involved, the regulatory authority considers the inclusion of the respective sources or practice in the system to be unnecessary [1]. Similarly, sources and practices already under regulatory control may be cleared from regulatory requirements if the regulatory authority considers that this is warranted [1].

205. The two basic criteria for determining, from a radiation protection standpoint, whether or not a source can be exempted[2] from regulatory control are contained in Refs [1, 3]; they are as follows:

— Individual risks must be sufficiently low not to warrant regulatory concern; and
— Radiation protection must be optimized, taking the cost of regulatory control into account.

In addition, exempted sources and practices must be inherently safe, with no appreciable likelihood of scenarios that could lead to a failure to meet the above criteria.

206. The first criterion, individual risk, is addressed by defining a level of individual dose[3] that can be regarded as 'trivial'. The second criterion is usually addressed by using optimization analysis techniques such as cost–benefit analysis, intuitive or formal, or other methods.

207. There are two main approaches that can be considered in deciding whether a level of dose is trivial: first, to choose a level of risk and a corresponding dose which have no significant effect as regards individuals; and secondly, to use the exposure

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[2] Although the general principles were established for exemption, it is clear that the same principles apply to clearance.

[3] Unless otherwise stated, the term 'dose' refers to the sum of the effective dose from external exposure in a given period and the committed effective dose from radionuclides taken into the body in the same period.

to natural background radiation, to the extent that it is normal and unavoidable, as a relevant reference level. In Ref. [3] these approaches are evaluated and it is concluded that for the purpose of exemption a level of individual dose of some tens of microsieverts in a year can reasonably be regarded as trivial.

208. Because an individual may be exposed to radiation from several exempted practices, it is necessary to ensure that the total dose does not rise above the trivial dose level. It is therefore recommended that each exempt practice should contribute only a part of the identified trivial dose [3]. The apportionment suggested could lead to individual doses to average members of the critical group of the order of 10 μSv/a from each exempt practice [3]. The value of 10 μSv/a is used in this report as the basis for evaluating unconditional clearance levels.

209. In 1990, after the publication of Safety Series No. 89, the International Commission on Radiological Protection (ICRP) issued a revised set of recommendations in its Publication 60 [12]. The main changes that are relevant in this context are as follows: first, the risk per unit dose has been increased; secondly, the organ weighting factors have been changed, leading to a new quantity 'effective dose' (in place of 'effective dose equivalent') and thirdly, the concept of potential exposures has been introduced. For the purposes of this report, the 10 μSv/a individual dose level has been retained, as it relates to a level of dose that is still within the range of risks considered in Safety Series No. 89 to be trivial and is also small in comparison with doses due to natural background radiation. The numerical differences between results expressed as 'effective dose equivalent' and 'effective dose' are normally small compared with other uncertainties in the analysis and so no adjustment has been made to those results originally reported in terms of effective dose equivalent. The concept of potential exposures is addressed in the next section.

210. In relation to the optimization of protection, each practice should be assessed as if it were to be subjected to a formal optimization procedure [3]. A study of the available options (including various kinds of regulatory action) should be made. If exemption is found to be the option that optimizes radiation protection, then exemption is the appropriate course of action, provided individual doses are trivial. If, however, a preliminary analysis shows that the practice gives a collective dose commitment of less than about 1 man·Sv per year of practice, then the total detriment is low enough to permit exemption without more detailed examination of other options.

211. In order to avoid regulatory problems, clearance levels should not be greater than the prescribed levels for exemption from regulatory control [1]. This is to ensure that material, once cleared from regulatory control, does not immediately become liable for regulation once more.

The need for derived quantities

212. The guiding radiological criteria for exemption and clearance are expressed in terms of dose and cannot be used directly for establishing exemption or clearance levels. Hence it is necessary to convert them into practical quantities. In this context, for solid materials, useful quantities are mass activity concentration (Bq/g), surface contamination (Bq/cm^2), total activity per unit time (Bq/a) and total mass per unit time (t/a). The derivation of these quantities requires a thorough examination of the reasonably possible routes by which humans may be exposed to radiation from the proposed exempt or cleared radioactive materials. The radiation doses associated with each route of exposure and for each radionuclide considered must be evaluated. On the basis of these evaluations it is possible to calculate a quantity, either in Bq/a, Bq/g, Bq/cm^2, t/a, or some combination of these, which will satisfy the exemption criteria specified in Safety Series No. 89. Details of the methodology have been described for landfill disposal and incineration [5] as well as for recycling and reuse [4].

213. In considering the pathways by which humans may be exposed to radiation, it will always be possible to envisage some circumstances in which exposure may occur but is not certain to occur. Such exposures are termed 'potential exposures'. They may be foreseen and their probability of occurrence estimated, but they can never be accurately predicted. Potential exposures should be considered as a part of the assessment process. However, dose limits do not apply directly to potential exposures; ideally they should be replaced by risk limits which take account of both the probability of incurring a dose and the detriment associated with that dose if it were to be received. It has been suggested in Ref. [12] that a simpler approach be taken, provided that the doses are less than the dose limits even if the event occurs. In such a case it will suffice to use the product of the expected dose and its probability of occurrence as if this were a dose that is certain to occur. This approach can be applied to scenarios which have a low probability of occurrence and would result in annual individual doses above 10 μSv but below the relevant dose limit. The difficulty with this approach is in estimating the probability of occurrence of the scenarios. The approach adopted in this report is to constrain doses from 'likely' scenarios to 10 μSv/a and for 'unlikely' scenarios to 100 μSv/a. Thus, even for unlikely scenarios doses would still be close to the dose region described in Safety Series No. 89 as trivial.

Unconditional and conditional clearance

214. The concept of clearance from regulatory control implies a removal of restrictions so that the cleared materials can be treated without any consideration of their

radiological properties. However, the removal of restrictions may not always be complete; there is also the possibility of clearing material under specified conditions. The application of conditions ensures that conditional releases will also provide an adequate degree of radiological protection to the public.

215. The full and complete clearance of a material requires that all reasonably possible exposure routes are examined and taken into account in the derivation of the clearance levels, irrespective of how that material is used and to where it may be directed. Such clearances are here called 'unconditional clearances'.

216. Alternatively, the clearances may be constrained in some way, usually because the fate of the material being considered in the clearance is known, so that only a limited number of reasonably possible exposure routes have to be considered in deriving the clearance levels. The clearance may then be granted with certain conditions, for example, it may prescribe a definite fate for the material being considered. Such clearances are here called 'conditional clearances'.

Considerations in deriving unconditional clearance levels

217. As discussed above, the derivation of unconditional clearance levels must take into account radiation exposure during all the reasonably possible uses and movements of the material intended for clearance. For a given radionuclide, the derived quantity will be determined by the scenario and exposure pathway which give rise to the highest radiation dose. Since the value of the unconditional clearance level must be acceptable everywhere, it must be based on consideration of generic scenarios and data. To ensure that the derived values are widely applicable it will usually be necessary to err on the conservative side of the range of observed data in the choice of assumptions and parameter values. For these reasons, the values derived for use in unconditional clearances will tend to be conservative, i.e. in most cases the actual doses received will be well below the individual dose criterion.

218. The individual dose criterion of 10 μSv/a is appropriate for use in cases where there is no or insufficient knowledge about other exposures of the critical group. However, it should be borne in mind that the level of 10 μSv/a is not to be considered as a limit; hence the requirement to comply with the individual dose criterion is not as strict as the requirement to comply with dose limits.

Considerations in deriving conditional clearance levels

219. When the practice which is a candidate for clearance is well defined, such as disposal to a landfill or the recycling of steel scrap by melting, it will usually be possible to take account of the known features of the practice. The likelihood of critical

group exposure due to overlapping practices should be taken into account in interpreting the recommendations of Safety Series No. 89. If it is clear that the likelihood of accumulating doses from more than one cleared practice is small, then a more liberal apportionment of the trivial dose (a few tens of microsieverts) may be considered. Also, in deriving the clearance levels, if there is good knowledge of the practice being considered, it may be possible to limit the number of exposure scenarios which need to be considered, and to introduce practice specific data into the dose calculation.

220. These considerations may be expected, in general, to lead to higher clearance levels than in the case of unconditional clearance. The values will, of course, depend on certain assurances being given on the fate or use of the radioactive material. If such assurances cannot be given, clearance level values determined under unconditional clearance considerations would be more appropriate.

3. DERIVED UNCONDITIONAL CLEARANCE LEVELS FOR RADIONUCLIDES IN SOLID MATERIALS

METHOD OF DERIVATION

301. The approach used to derive unconditional clearance levels is based on an analysis of published reports, reviewing assessment studies in which potential radiation doses from various possible uses or operations involved in the disposal, incineration, recycling or reuse of contaminated materials were estimated. A summary of these reports and the analysis is given in Appendix I.

302. The individual dose criterion used in the Appendix as the basis for calculating the unconditional clearance levels was 10 μSv/a, as discussed in Section 2, and nuclide specific values of clearance levels, in terms of Bq/g and/or Bq/cm^2, were obtained for each of the dose assessment scenarios considered in each of the reviewed studies. By this means, it was possible to produce a range of clearance level values for each radionuclide considered. This range reflects, in part, the assumptions made by each of the assessment groups involved in producing the reports.

303. Not all the data were included when constructing the ranges of values. The general criteria for accepting the data were that the data sets should be based on credible exposure scenarios having a reasonable likelihood of occurring, with realistic assumptions concerning transfer parameters, exposure times, etc. Some of the

reviewed assessments, or components of them, were judged, in the context of the present study, to be:

— incomplete, with important exposure scenarios missing;
— unrealistic, because of the assumptions made.

In such cases, the data were excluded from further consideration.

304. For the remaining data, a procedure was devised for determining unconditional clearance levels from the ranges of values for each radionuclide. The aim of the procedure was to select values which would provide a high degree of assurance that doses would not exceed 10 μSv/a.

305. The values determined from the analysis are presented in the following paragraphs together with notes on how they should be interpreted. A description of the data reviewed and of the procedure followed to obtain the unconditional clearance levels is given in Appendix I.

DERIVED VALUES AND THEIR INTERPRETATION

306. Compliance with the derived values for unconditional clearance from regulatory control of solid materials (Table I) will provide a high degree of assurance that the individual dose criterion of 10 μSv/a will not be exceeded, irrespective of the use or application of material after its release. The values have been derived assuming the equal possibility of disposal, incineration, recycling or reuse. Furthermore, it is assumed that the cleared materials could be used anywhere, e.g. in another country as a result of transboundary movement. The analyses of potential radiological impact are, therefore, necessarily generic and conservative. The levels may be regarded as those below which release from regulatory control is 'automatic', without further consideration being needed. As explained in the Foreword, release from regulatory control may also be allowed under other conditions.

307. Table I should be interpreted as follows:

(a) The uncertainties in the results of studies used to develop the classification do not allow a single number to be attached to each radionuclide but only categorization by order of magnitude.
(b) Where a single value of the clearance level is required by regulators, the log-mean values for each category are proposed for use as representative clearance level values. The clearance levels are then 0.3, 3, 30, 300 and 3000 Bq/g for the five classes.
(c) The levels in Table I are those below which the unconditional clearance of material containing the relevant radionuclides from facilities under regulatory control may be allowed.

TABLE I. DERIVED UNCONDITIONAL CLEARANCE LEVELS

Ranges of activity concentration (Bq/g)	Radionuclides[a]			Representative single values of activity concentration (Bq/g)
0.1	Na-22	Cs-134	U-234	
	Na-24	Cs-137	U-235	
	Mn-54	Eu-152	U-238	
	Co-60	Pb-210	Np-237	
	Zn-65	Ra-226	Pu-239	0.3
	Nb-94	Ra-228	Pu-240	
	Ag-110m	Th-228	Am-241	
	Sb-124	Th-230	Cm-244	
		Th-232		
< 1.0				
≥ 1.0				
	Co-58	In-111		
	Fe-59	I-131		
	Sr-90	Ir-192		3
	Ru-106	Au-198		
		Po-210		
< 10				
≥ 10				
	Cr-51			
	Co-57	I-129		
	Tc-99m	Ce-144		30
	I-123	T1-201		
	I-125	Pu-241		
< 100				
≥ 100				
	C-14	Sr-89		
	P-32	Y-90		300
	Cl-36	Tc-99		
	Fe-55	Cd-109		
< 1000				
≥ 1000				
	H-3	Ni-63		
	S-35	Pm-147		
	Ca-45			3000
< 10 000				

[a] Radon-220 and radon-222 were not considered in this classification.

(d) In the absence of other guidance, the clearance level values for surface contamination (Bq/cm^2) may be taken to be the same in unit terms as for activity concentration (Bq/g). Where appropriate, mass and surface criteria should be applied simultaneously, for example for metal objects and buildings. For many materials it will only be possible to apply activity concentration values, for example for materials with uneven, rough or porous surfaces.

(e) In nearly all practical cases, more than one radionuclide will be involved. To determine if a mixture of radionuclides is at or below the clearance level a simple ratio expression can be used:

$$\sum_{i=1}^{n} \frac{C_i}{C_{Li}} \leq 1$$

where C_i is the concentration of radionuclide i in the material being considered (Bq/g); C_{Li} is the clearance level of radionuclide i in the material (Bq/g); and n is the number of radionuclides in the mixture.

In the above expression, the ratio of the concentration of each radionuclide to the clearance level is summed over all radionuclides in the mixture. If this sum is less than or equal to one, the material complies with the clearance requirements.

(f) For radionuclides not presented in Table I it is recommended that the following expression be used to categorize the desired nuclide.

$$\text{Minimum} \left\{ \frac{1}{E_\gamma + 0.1E_\beta}, \frac{ALI_{inh}}{1\,000}, \frac{ALI_{ing}}{100\,000} \right\}$$

where E_γ is the effective γ energy in MeV and E_β is the effective β energy in MeV, as given in ICRP Publication 38 [13]; ALI_{inh} is the most restrictive value of the annual limit on intake by inhalation in Bq; and ALI_{ing} is the most restrictive value of the annual limit on intake by ingestion in Bq as given in ICRP Publication 61 [14]. (See Appendix I for more details):

(g) Some relaxation of the values in Table I should be possible under certain conditions, e.g. where the fate of the released material is well defined, or where the size or surface area of objects to be cleared is small. Such clearances are termed conditional clearances. Administrative conditions and numerical criteria for these conditional clearances must be established by national authorities.

4. VERIFICATION OF CLEARANCE LEVELS

401. From a regulatory viewpoint, it is necessary to be able to verify that the cleared material conforms to the applicable clearance levels. This can be done by direct measurement on the material in question, by laboratory measurements on representative samples, by the use of properly derived scaling factors or by other means which are accepted by the competent national authority.

402. The goal of keeping individual doses below about 10 μSv/a implies that dose rates have to be detected which are a small fraction of natural background and so it is necessary to operate at the limits of detectability.

MEASUREMENT METHODS

403. A number of publications exist dedicated fully or partially to the measurement methods, devices and techniques required to verify clearance levels [15–24]. Although it is not the intention here to give a full review of these and other literature, it may be concluded that clearance levels for the most frequently used radionuclides as well as for radionuclide compositions typical for decommissioning wastes from nuclear power plants can be determined with direct measurements. In industrial and other routine practices where economic and practical considerations apply, the choice of measurement strategy and appropriate measurement instruments are important. Depending on the radionuclides present it may be necessary to supplement direct measurements on the material with laboratory analysis of suitably selected samples.

404. In deciding on a measurement strategy, the following two steps are important:

- To group the material to be cleared so that it is as homogeneous as possible in relation to both material and origin (and thus radionuclide spectrum).
- To assess the radionuclide spectrum for the material to be cleared by analyses of samples, taking into account all available information about the operational history of the material.

On the basis of this information, the measurement method can be selected and suitable instruments can be chosen and appropriately calibrated.

405. Because it is unlikely that unconditionally cleared materials will contain truly uniform levels of contamination, procedures must be developed for determining the appropriate mass or material surface over which radionuclide quantities may be averaged. In this context, a distinction is made between averaging reasonable variations in activity throughout a contaminated mass or across a surface, and reaching clear-

ance levels by deliberately adding relatively high specific activity materials to low activity or uncontaminated materials. The latter practice, dilution, is generally not appropriate.

406. Averaging procedures should be an integral part of the clearance system and need to be selected according to the type of material leaving the site. The destination of the released material is unknown and it is possible that breakup of the released material could result in pieces which have a significantly higher specific activity than the recommended unconditional levels. The probability of this occurring can be reduced by averaging over small quantities and areas.

407. For large volumes of material or large areas and in cases where measurements covering the complete surface of the material are not feasible, it is important to apply proper statistical methods to optimize the measurements, including the sampling process (see, for example, Refs [16, 17]).

408. The distribution of radionuclides in scrap metal is generally inhomogeneous and the geometry may make measurement difficult. However, the melting process has the effect of producing a homogeneous distribution of radionuclides in the metal and the activity of the melted batch can then be directly obtained from the laboratory analysis of a single small sample.

409. There are a number of radionuclides which are difficult to measure directly on material to be cleared, for example, alpha- and (weak) beta-emitting radionuclides. However, many of these can be related to other radionuclides in specific waste streams. For example ^{55}Fe and ^{63}Ni can often be correlated to ^{60}Co, and ^{90}Sr to ^{137}Cs, both of which are easy to measure. When using scaling factors to verify levels of radionuclides which cannot be directly measured on the material, it is necessary to have a well founded base for the scaling factor and use the factor only on the waste streams for which the scaling factor has been established. It is also necessary to be aware of any changes in the processes giving rise to the waste stream, as these may affect the ratios of radionuclides present in the material. If it can be reasonably demonstrated through a record of irradiation, contamination, handling or utilization, or an appropriate combination of these, that a type or group of radionuclides is not present in the source of the waste to be cleared, most competent authorities would accept that these radionuclides need not be investigated.

410. For naturally occurring radionuclides, such as ^{226}Ra, ^{232}Th and ^{238}U, verification can present problems. This is not due to difficulties of detection but rather to the problems caused by the prevalence of these radionuclides in materials of the natural environment. Materials used in building construction frequently contain natural radionuclides at levels close to or in excess of the recommended clearance

levels. The Annex presents data on measured levels of some commonly occurring radionuclides and shows that it may be difficult to distinguish artificially enhanced levels from variations in the natural background levels of radiation (see also paras 504–507).

5. REGULATORY ASPECTS

REGULATORY METHODS

501. Two mechanisms can be envisaged for including clearance criteria within a regulatory framework:

(a) By conditions placed in the licence or authorization of each individual licence or authorization holder to whom the clearance applies. (This assumes that the legislation allows for this approach.)
(b) By legislation, either primary (Acts or Bills, etc.) or secondary (Regulations, Statutory Instruments, etc.) if suitable primary legislation is in place.

Option (a) would allow the clearance requirements to be produced to meet the requirements of the individual, while still remaining within the overall framework. Option (b) would ensure that all operators work to the same standards, and that regulators only have to concern themselves with one general set of clearance criteria. This approach would also ensure that no one site operator gains at the expense of another.

502. As discussed in Section 2, the clearance may be conditional or unconditional:

Conditional clearance

For conditional clearance, materials will be free for release from the site subject to certain conditions such as disposal route or destination. These would be checked by regulators to ensure compliance with the conditions.

Unconditional clearance

For unconditional clearance, the materials may be released from the site without regard to their destination. It is assumed that regulators do not carry out follow-up checks once the material has left the site, since the material is deemed to be outside regulatory concern.

For both conditional and unconditional clearances, it is expected that the regulators will audit those releasing cleared material to ensure compliance with clearance criteria.

COMPLIANCE ASSURANCE AND QUALITY CONTROL

503. All the activities required of a licensee or otherwise authorized person (or organization) to verify compliance with the established levels for clearance should be performed within the framework of a quality control system, set up in accordance with pertinent, recognized quality assurance requirements. In the definition of such a system, the potential amount of cleared materials and the complexity of the practice should be taken into account. The system should, in particular, include the keeping of records on quantities of released material and on the related activity concentrations. Regulatory authorities should periodically review the cleared practices to assure themselves that the radiological considerations and analysis continue to be valid, i.e. that the relevant parameters determining the conditions of a clearance authorization have not changed significantly.

NATURALLY OCCURRING RADIONUCLIDES

504. Natural sources of radiation are normally excluded from regulations because they are not amenable to control. In the Basic Safety Standards [1], exposure to natural sources is normally excluded, except where specified by the regulatory authority. In cases where chronic exposure to natural sources is significantly enhanced above natural levels, the regulatory authority may judge that intervention is required, for example, for radon in the home. In such cases remedial actions would normally be invoked if an action level specified by national authorities is exceeded.

505. The Basic Safety Standards place strong emphasis on the role of the regulatory authority in determining when controls are necessary on exposures to natural sources and since there are usually a number of source and site specific factors influencing such judgements, it may be necessary for the regulatory authority to regulate on a case by case basis.

506. The clearance levels for the naturally occurring radionuclides in Table I are applicable to materials intended for clearance but contaminated with natural radionuclides, for example, scrap metals, concrete, etc., rather than to naturally occurring materials, such as rocks and stones containing such nuclides. The values in Table I apply to the components which have been added to the natural levels in the materials concerned. It is recognized, however, that there may be problems in determining an

'average' natural level and, in some cases, on judging whether there is any real enhancement (see also para. 410).

507. It is clear that the application of clearance concepts to materials contaminated with naturally occurring radionuclides is not straightforward and some interpretation and judgement by national regulatory authorities is likely to be needed.

OTHER CONSIDERATIONS

508. Regulatory authorities should ensure that use of the clearance option does not absolve the disposer from responsibility for complying with other pertinent legal requirements for disposal of waste with other hazardous characteristics (e.g. bio-hazardous waste).

509. Different clearance levels may be appropriate if the basis for concern is other than health. For example, if cleared materials were to be used in radiosensitive industries there would be a need to take potential effects into account in deriving levels.

510. From consideration of the collective doses which could be accumulated as a result of the unconditional clearance of materials it seems that fairly substantial quantities in terms of mass would have to be involved before collective dose commitments of more than 1 man·Sv per year of practice would be delivered (see Appendix II). Nevertheless, this is an aspect which national regulatory authorities need to keep under review in granting clearances.

Appendix I

DERIVATION OF UNCONDITIONAL CLEARANCE LEVELS FOR SOLID MATERIALS

REVIEW OF ASSESSMENT STUDIES

I.1. In recent years, studies have been conducted both at international and national levels concerned with the derivation of clearance levels. These studies have been directed towards the low activity streams of material generally considered to be the most likely candidates for clearance from regulatory control. These are:

- Low level solid wastes from the nuclear fuel cycle, for example, the lightly contaminated paper, plastics and clothing which arise in work with radioactive materials;
- The slightly contaminated ferrous and non-ferrous metals and concrete which arise mainly in the decommissioning and refurbishing work at nuclear facilities;
- The low level wastes generated during the application of radioisotopes in industry, hospitals and research laboratories.

The solid wastes are normally disposed of to landfills or by incineration (and residual ash to landfills). Metals and concrete may also be considered for recycling and reuse.

I.2. In the above mentioned studies, potential individual radiation exposures have been evaluated for a range of scenarios linked to each of the practices considered. The following list summarizes the main scenarios considered, although it should be noted that the detailed assumptions are different in each case.

Landfill disposal	—	transport workers
	—	landfill site workers
	—	disturbance of the site after closure
	—	radionuclide transfer via groundwater
	—	fires in the landfill
Incineration	—	operators
	—	emissions
	—	ash (to landfill)
Recycling (steel)	—	scrap transport workers
	—	scrap processing workers
	—	workers at smelter and fabrication plant
	—	consumer use

— emissions
— use of slag

(Similar groups of scenarios are considered for non-ferrous metals and concrete.)

Reuse
— small tools and equipment
— large equipment
— buildings (use and renovation)

The evaluation of radiation exposure in each of the scenarios takes account, as necessary, of exposure due to external irradiation, and to inhalation and ingestion of radionuclides.

Results of previous studies

I.3. While all the studies considered address the problem of relating the activity concentration or total activity in materials to radiation dose, they were not all conducted for the same purpose. Some studies were intended only to derive a relationship between activity concentration and radiation dose. Some of the reported values are based on dose criteria other than those given in Section 2 of the main report. Also, the dosimetry used in the studies varies; in most of the studies ICRP 30 dosimetry [25] was used but in the remainder the dosimetry is based on the concepts of ICRP 60 [2]. The approaches taken in relation to these issues are discussed in the subsequent text.

I.4. In Tables I.1 to I.5 the results of the reviewed studies are summarized. Unless otherwise noted, the clearance levels have been derived on the basis of an individual dose criterion of 10 μSv/a. The values presented are the most limiting results obtained from each of the studies considered.

Landfill disposal

I.5. Table I.1 summarizes the results of seven studies on this subject. A brief description of each of the studies is given below.

IAEA [5]

I.6. Part II of the IAEA study is intended to provide guidance on the methods for establishing exempt quantities for the terrestrial disposal (landfill and incineration) of low level radioactive wastes but 'example exempt quantities' for a range of radionuclides are also derived. The radiological basis is an individual dose of 10 μSv/a; collective doses are also considered. The study uses ICRP 30 dosimetry [25].

TABLE I.1. SUMMARY OF RESULTS OF STUDIES ON CLEARANCE LEVELS FOR LANDFILL DISPOSAL (Bq/g)

Radio-nuclide	IAEA [5]	Müller-Neumann [27]	Poschner et al. [29]	Sumerling and Sweeney [26]	Guetat et al. [28]	Elert et al. [30]	Harvey et al. [31]
H-3	3×10^4	2×10^3		4×10^3	4×10^3	(1×10^{-1})	(6×10^5)
C-14	1×10^2	6×10^2		1×10^2	8×10^2	(1×10^{-1})	(2×10^4)
Na-22	6×10^{-1}	2×10^0	2×10^1		6×10^{-1}		(3×10^3)
Na-24					3×10^{-1}		(2×10^4)
P-32	1×10^3	3×10^3	1×10^4	7×10^2	1×10^3		(4×10^3)
S-35	5×10^3	5×10^3	2×10^3		1×10^4		(3×10^4)
Cl-36		(2×10^{-2})	(2×10^{-2})		6×10^1	(4×10^{-4})	(1×10^4)
Ca-45	2×10^5	8×10^3			8×10^3	(6×10^2)	(1×10^4)
Cr-51		1×10^2		3×10^2	4×10^1		(2×10^5)
Mn-54	2×10^0	4×10^0			1×10^0	(5×10^{-2})	(1×10^4)
Fe-55					6×10^5	(3×10^1)	(3×10^4)
Fe-59		3×10^0			1×10^0		(3×10^3)
Co-57		3×10^1			1×10^1		(3×10^3)
Co-58		4×10^0			1×10^0	(5×10^{-2})	(1×10^3)
Co-60	5×10^{-1}	1×10^0		1×10^1	5×10^{-1}	(2×10^{-2})	(1×10^2)
Ni-63		1×10^4			5×10^4	(1×10^0)	(5×10^4)
Zn-65		6×10^0			2×10^0	(8×10^{-2})	(3×10^3)
Sr-89		4×10^3			4×10^3		(3×10^3)
Sr-90	3×10^2	2×10^0		3×10^1	1×10^1	(8×10^{-5})	(4×10^2)
Y-90		2×10^4			1×10^3		(2×10^3)
Nb-94		2×10^0			7×10^1	(5×10^{-3})	(4×10^3)
Tc-99m				9×10^1	2×10^1		(5×10^5)
Tc-99		(3×10^{-2})			4×10^1	(1×10^{-3})	(1×10^4)
Ru-106	6×10^0	2×10^2			6×10^0	(1×10^{-2})	(9×10^2)
Ag-110m		1×10^0			4×10^1		(3×10^3)
Cd-109		4×10^2			2×10^2		(5×10^3)
In-111					3×10^0		(3×10^4)
I-123					1×10^1		(5×10^4)
I-125		7×10^2		3×10^2	1×10^2		(7×10^2)
I-129		(2×10^{-2})			5×10^1	(2×10^{-3})	(9×10^1)
I-131	4×10^0	9×10^0		3×10^1	3×10^0		(5×0^2)
Sb-124		2×10^0			7×10^1		(3×10^3)

TABLE I.1. (cont.)

Radio-nuclide	IAEA [5]	Müller-Neumann [27]	Poschner et al. [29]	Sumerling and Sweeney [26]	Guetat et al. [28]	Elert et al. [30]	Harvey et al. [31]
Cs-134		2×10^{0}			8×10^{1}	(3×10^{-2})	(5×10^{2})
Cs-137	1×10^{0}	6×10^{0}		2×10^{1}	2×10^{0}	(8×10^{-3})	(8×10^{2})
Ce-144	1×10^{2}	1×10^{2}			2×10^{1}	(3×10^{-2})	(1×10^{3})
Pm-147		4×10^{3}		2×10^{7}	1×10^{4}	(1×10^{2})	(2×10^{4})
Eu-152		3×10^{0}			1×10^{0}	(4×10^{-2})	(5×10^{3})
Ir-192					2×10^{0}		(5×10^{3})
Au-198					3×10^{0}		(6×10^{3})
Tl-201					2×10^{1}		(1×10^{5})
Pb-210			1×10^{1}		2×10^{1}		(5×10^{0})
Po-210			2×10^{1}		2×10^{2}		(2×10^{1})
Th-228			3×10^{-1}		8×10^{1}		(4×10^{1})
Th-230			5×10^{-1}		8×10^{0}		(3×10^{1})
Th-232			1×10^{-1}	1×10^{1}	2×10^{0}		(2×10^{0})
Ra-226			3×10^{-1}	1×10^{1}	7×10^{1}		(5×10^{0})
Ra-228			3×10^{-1}		1×10^{0}		(4×10^{1})
U-234			1×10^{0}		1×10^{1}		(3×10^{2})
U-235			1×10^{0}		8×10^{0}		(3×10^{2})
U-238			1×10^{0}	1×10^{1}	1×10^{1}		(2×10^{2})
Np-237		3×10^{-1}			3×10^{0}		(2×10^{1})
Pu-239	6×10^{-1}	3×10^{-1}		4×10^{2}	6×10^{0}	(2×10^{-3})	(2×10^{1})
Pu-240		3×10^{-1}			6×10^{0}	(2×10^{-3})	(2×10^{1})
Pu-241					1×10^{2}	(5×10^{-1})	(9×10^{2})
Am-241	6×10^{-1}	3×10^{-1}		2×10^{2}	6×10^{0}	(3×10^{-3})	(2×10^{1})
Cm-244		6×10^{-1}			1×10^{1}	(2×10^{-2})	(3×10^{1})

Notes: The values listed from each of the studies are the most limiting from the various exposure scenarios considered.

Values are based on an individual dose level of 10 μSv/a.

( ) means not considered in the summary of results (see text).

For landfill disposal the annual amount of radioactive waste is taken to be 1.68×10^{4} t/a, which is 10% of the annual total of waste received at the landfill. For incineration the annual amount of radioactive waste is assumed to be 1.5×10^{5} t/a; again this is 10% of the total waste received.

I.7. Comparison of the results of the assessment with those of other similar landfill assessments shows that the 'landfill residential scenario' gives clearance levels at least two orders of magnitude lower than other comparable scenarios in other studies. Further investigation shows this to be mainly due to the very small dilution of the radioactive waste with non-active material assumed in the study. National experiences show that the assumed dilution is unrealistic. The IAEA ratio of radioactive to non-radioactive material of 0.44 compares with 0.002 [26], 0.01 [27] and 0.06 [28]. The IAEA landfill residential scenario has therefore been omitted from the present analysis. The radionuclides affected are tritium, ^{60}Co, ^{90}Sr and ^{137}Cs; the values for these radionuclides in Table I.1 are, in each case, from the next most limiting scenario in the IAEA analysis.

Müller-Neumann et al. [27], *Poschner et al.* [29]

I.8. In these reports[4] exempt activity concentrations are derived for the disposal of low level waste by landfill and incineration on the basis of an individual dose rate of 10 μSv/a. ICRP 30 dosimetry is used [25]. Three sizes of landfill site are considered, on the basis of national experience in Germany. The smallest size (1×10^4 t/a) gives the most restrictive results and these are used in the present analysis. The assumed ratio of radioactive to non-radioactive material is 0.01. The incinerator plant is assumed to burn 7.5×10^4 t/a, of which a fraction of 0.0013 is radioactive.

I.9. The landfill study gives results for the radionuclides ^{36}Cl, ^{99}Tc and ^{129}I, which are very low, compared with those from the other studies. The limiting pathway for the radionuclides was ingestion following release of seep water from the landfill. The values from these studies are being discussed as a basis for German regulatory recommendations for the disposal of slightly radioactive waste. Within this process the models and parameters used are being re-evaluated and tested. In particular, the ingestion pathway via groundwater is being recalculated using an improved model and more realistic assumptions. While the calculated exemption levels for most of the radionuclides are not expected to change significantly, it is expected that the revised values for ^{36}Cl, ^{99}Tc and ^{129}I, which are controlled by the groundwater pathways, will be considerably less restrictive. For these reasons, the Table I.1 values for these radionuclides have been omitted from the rest of the analysis. Since the same model was used for incineration (disposal of ashes to landfill), the values in Table I.2 for these radionuclides are also excluded.

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[4] The Poschner et al. [29] report is a follow-up study of the Müller-Neumann et al. [27] study. It applies the same model to other radionuclides, mainly naturally occurring ones.

Sumerling and Sweeney [26]

I.10. This study was commissioned by the United Kingdom Department of the Environment to examine the existing national exemption orders and to evaluate the implied radiological impact of the associated exempt practices. Both individual and collective doses were examined. Results are presented in terms of doses per unit activity disposal per year. The exemption orders of interest covered waste disposal to small landfills and special precautions burial. The wastes originate mainly from research and medical applications and from the demolition of buildings. The study uses ICRP 30 dosimetry [25]. The assumed ratios of radioactive to non-radioactive wastes are 0.002 for so called 'dustbin disposal' (total waste 1×10^4 t/a), 0.0001 for special disposal (total waste 2.5×10^5 t/a) and 0.1 for dustbin disposal of waste from demolition.

Guetat et al. [28]

I.11. This study was carried out as part of a project to define reporting levels for the revision of the Commission of the European Communities Regulations on Radiation Protection. It considers exposure to radiation in the use of concentrated sealed and unsealed sources and the disposal, incineration, recycling and reuse of larger amounts of diluted sources. Calculations are made of limiting activity concentrations and total activities, generally based on 1 mSv/a. For atmospheric releases 10 μSv/a is used. The results have been normalized for the use in the present study to be consistent with 10 μSv/a. The study uses dosimetry based on ICRP 60 [12]. The quantity of radioactive waste considered is of the order of 1000 t/a and the assumed ratios of radioactive to non-radioactive waste are 0.02 or 0.06 for landfill disposal, and 0.03 for incineration. For recycling the assumed ratio depends on the product and the step of the practice. For some scenarios it is unity.

Elert et al. [30]

I.12. This study was intended to provide technical background material as a basis for future decisions on exemption levels for decommissioning wastes. Relationships are provided to evaluate individual doses from concentrations of activity in steel and concrete which may be disposed of in a landfill repository or recycled. The study uses dosimetry based on ICRP 30 [25].

I.13. The results for landfill disposal are based on the assumption that an entire nuclear power plant is dismantled within a year and that the concrete and other materials are disposed of at a single site. It is assumed that the whole landfill is contaminated at the same level with no dilution by other non-radioactive materials. Thus, although the assumed quantity of material is realistic, neither the assumed

time-scale for disposal nor the assumption that the entire nuclear power plant would be contaminated at the exempt level is likely. The parameter values used for assessing radiation doses are rather conservative, e.g. a high value for the dust concentration at the landfill (5 mg/m^3) is assumed for the entire exposure period. The values derived for long lived beta emitters such as ^{36}Cl and ^{90}Sr are based on the drinking water pathway. The scenario assumes that the entire water supply for the exposed individuals is obtained from a well which collects all leachate from the landfill and with no treatment prior to consumption. The overall result is that the calculated values are generally two to four orders of magnitude more restrictive than in the other studies.

I.14. The authors have pointed out (in private communications) that the study was not intended to assess radiological consequences but was mainly intended to provide a calculated method and to give values of dose per unit concentration. They expect that more appropriate and realistic assumptions regarding the disposal and recycling scenarios would be made by those applying the method. The results have been excluded from the subsequent analyses.

Harvey et al. [31]

I.15. This study derived activity concentrations and total activities for use in the revision of the Commission of the European Communities Regulation on Radiation Protection. It considers the use and disposal of radionuclides in relatively small operations outside the nuclear industry. In the first stage of the study, a very wide range of exposure scenarios is considered for a limited range of radionuclides. The results of the first stage were used to identify the most important exposure scenarios, and the chosen scenarios were then used in the second stage of the study for a much more extensive range of radionuclides. The scenarios studied in detail are: normal use by workers within the operation; misuse or accidents involving workers; exposure of the public following disposal of material to a landfill. The dosimetry used is based on ICRP 60 [12].

I.16. The scope of the study by Harvey et al. is slightly different from the scope of this report since it was used to establish exemption levels for application outside the nuclear industry. Therefore it considered only moderate amounts of radioactive materials and did not take account of the disposal of large volumes of material at one site. In addition, the landfill results, which are given in Table I.1, were not intended to be used on their own (the study also considered the use and misuse of the material and this influenced the choice of pathways and parameter values). Thus, exemption concentrations derived solely on the basis of this landfill scenario (which, for many radionuclides, was not the limiting scenario) cannot be compared directly with those calculated in other studies. They do, however, provide an indication of the range of exemption levels that might be appropriate.

I.17. The results of Harvey et al. have been excluded from the subsequent analysis in paras I.37 ff.

Summary of results of landfill studies

I.18. The first five studies considered in Table I.1 produced results which are in reasonable agreement with each other; that is, the values are generally within a factor of 10 for a given radionuclide. The agreement is further improved when the landfill residential scenario values of the IAEA [5] and the values for ^{36}Cl, ^{99}Tc and ^{129}I of Müller-Neumann et al. [27] and Poschner et al. [29] are excluded from consideration, as discussed earlier.

I.19. Reviewing the results of the first five studies of Table I.1 it can be seen that there is considerable variation in the values, depending upon the radionuclide; a range of about 10^4 is seen between the values for tritium and those for ^{239}Pu and ^{60}Co. The results consistently show the lowest values as being associated with the long lived alpha emitters (^{232}Th, ^{239}Pu, ^{241}Am, etc.) and energetic photon emitters (^{60}Co).

Incineration

I.20. Table I.2 summarizes the results of five studies on the derivation of exempt levels for disposal by incineration. All five studies also included landfill disposal and have been described previously. Where comparison is possible the agreement between the results of the studies is good; there is a substantial range of values, depending on the radionuclide. In this case, the range is about 10^5. The lowest values are obtained for the long lived alpha emitters (^{232}Th, ^{239}Pu, ^{241}Am, etc.), and for the energetic photon emitters such as ^{60}Co.

Recycling

I.21. Table I.3 summarizes the results of ten studies on the recycling of steel, aluminium, copper and concrete. Brief descriptions of the studies are given below.

IAEA [4]

I.22. This report provides guidance on methods for deriving exempt concentrations of radionuclides in materials from the nuclear industry intended for recycling and reuse. Derived values are given for the recycling of steel, aluminium and concrete and for the reuse of rooms, tools and equipment. In the case of reuse, both activity concentrations (Bq/g) and surface contamination (Bq/cm^2) are evaluated. The radiological basis is 10 μSv/a to individuals. Collective doses are also evaluated. The

TABLE I.2. SUMMARY OF RESULTS OF STUDIES ON CLEARANCE LEVELS FOR DISPOSAL BY INCINERATION (Bq/g)

Radio-nuclide	IAEA [5]	Müller-Neumann [27]	Poschner et al. [29]	Sumerling and Sweeney [26]	Guetat et al. [28]
H-3	7×10^5	1×10^6		4×10^4	3×10^4
C-14	9×10^2	3×10^3		1×10^2	1×10^3
Na-22	6×10^{-1}	1×10^1	2×10^0		1×10^0
Na-24					2×10^3
P-32	8×10^3	2×10^4	8×10^4		9×10^2
S-35	3×10^4	4×10^5	2×10^5		2×10^3
Cl-36			(2×10^0)		4×10^2
Ca-45	2×10^4			4×10^3	1×10^3
Cr-51					1×10^2
Mn-54	2×10^0				4×10^0
Fe-55					1×10^5
Fe-59					3×10^0
Co-57					3×10^1
Co-58					3×10^0
Co-60	5×10^{-1}	9×10^1			1×10^0
Ni-63		4×10^5			9×10^4
Zn-65					5×10^0
Sr-89					7×10^2
Sr-90	8×10^1	5×10^1			3×10^2
Y-90					4×10^3
Nb-94					2×10^0
Tc-99m				2×10^{10}	1×10^{10}
Tc-99		(6×10^0)			1×10^3
Ru-106	8×10^0				2×10^1
Ag-110m					1×10^0
Cd-109		8×10^3			5×10^2
In-111					5×10^1
I-123					2×10^5
I-125				1×10^3	3×10^2
I-129		(4×10^0)			7×10^1
I-131	8×10^0	1×10^2		2×10^3	2×10^1
Sb-124					2×10^0
Cs-134					2×10^0
Cs-137	3×10^0	4×10^1			5×10^0

TABLE I.2. (cont.)

Radio-nuclide	IAEA [5]	Müller-Neumann [27]	Poschner et al. [29]	Sumerling and Sweeney [26]	Guetat et al. [28]
Ce-144	9×10^{1}				6×10^{1}
Pm-147					1×10^{3}
Eu-152					3×10^{0}
Ir-192					4×10^{0}
Au-198					5×10^{1}
Tl-201					2×10^{2}
Pb-210			1×10^{2}		4×10^{0}
Po-210			1×10^{2}		4×10^{1}
Th-228			4×10^{0}		1×10^{0}
Th-230			4×10^{0}		2×10^{0}
Th-232			8×10^{-1}		4×10^{-1}
Ra-226			2×10^{2}		2×10^{0}
Ra-228			3×10^{2}		3×10^{0}
U-234			1×10^{1}		3×10^{0}
U-235			1×10^{1}		3×10^{0}
U-238			1×10^{1}		3×10^{0}
Np-237					7×10^{-1}
Pu-239	2×10^{-1}	3×10^{0}			1×10^{0}
Pu-240					1×10^{0}
Pu-241					3×10^{1}
Am-241	2×10^{-1}				1×10^{0}
Cm-244					2×10^{0}

Notes: The values listed from each of the studies are the most limiting from the various exposure scenarios considered.

Values are based on an individual dose level of 10 μSv/a.

( ) means not considered in the subsequent analysis (see text).

dosimetry is based on ICRP 30 [25]. The basis for the calculations is the recycling of 100 t of material, with no dilution by uncontaminated material and no partitioning of activity during melting, although the sensitivity of the results to varying dilution, partitioning and the quantity recycled is examined.

CEC [11]

I.23. This study evaluates the doses arising from the recycling and reuse of scrap steel from nuclear power plants contaminated at specified clearance levels. Both individual and collective doses are estimated. The main emphasis is on specific activity with only limited consideration given to doses related to surface activities. ICRP 30 dosimetry has been used [25]. The total quantity of material recycled per year is assumed to be 1×10^4 t. In evaluating the doses from the contaminated steel scrap it is assumed that during the steel melting process the scrap is diluted by mixing with larger amounts of uncontaminated scrap. The assumed dilution factor is 10.

Garbay et al. [32]

I.24. This report gives exempt activity concentrations for the recycling of copper and aluminium arising from the dismantling of nuclear installations (reactors and enrichment facilities). It is based on characteristics of French industry. The radiological criteria used as a basis for the derived activity concentrations are 10 μSv/a for groups for which it is judged that there is a possibility of exposures from other cleared practices and 50 μSv/a for groups where exposures from other cleared practices are considered unlikely. Normalization of the results to a basis of 10 μSv/a is complicated, and the values in Table I.3 are uncorrected. The dosimetry is based on ICRP 30 [25]. The quantities of contaminated material considered in the analysis are 500 t of copper and 20 t of aluminium per reactor. The ratios of radioactive to non-radioactive material vary depending on the step of the process but the ratios are relatively small. The assessment does not take surface contamination into account. The limiting scenarios are related to nuclear power plants, except for the uranium isotopes, where the limiting scenarios are related to enrichment facilities.

Elert et al. [30]

I.25. This study is discussed in paras I.12–I.14. For steel and concrete recycling it is assumed that all the steel and concrete from a nuclear power plant is homogeneously contaminated at the exempt level, that the steel is processed by one scrapyard and a single foundry in one year. The assumptions relating to time-scales for dismantling and the assumed procedure for recycling are considered to be unrealistic. Furthermore, in reality not all of the steel and concrete would actually be contaminated; large parts of buildings would be almost free of contamination.

I.26. The assumptions made in the assessments are unrealistic and this is reflected in the results obtained. They have been excluded from the subsequent analysis in paras I.37 ff.

TABLE I.3. SUMMARY OF RESULTS OF STUDIES ON CLEARANCE LEVELS FOR RECYCLING (Bq/g)

Radio-nuclide	Steel			Aluminium		Copper	Concrete			All four materials
	IAEA[a] [4]	CEC[b] [11]	Elert et al. [30]	IAEA[a] [4]	Garbay et al.[c] [32]	Garbay et al.[c] [32]	IAEA [4]	Haristoy et al. [33]	Elert et al. [30]	Guetat et al. [28]
H-3								8×10^{5}	(5×10^{-2})	1×10^{4}
C-14								7×10^{3}	(3×10^{1})	2×10^{3}
Na-22								1×10^{0}		7×10^{-2}
Na-24								∞		6×10^{-1}
P-32								2×10^{5}		2×10^{2}
S-35								3×10^{4}		3×10^{4}
Cl-36							2×10^{4}	1×10^{3}	(1×10^{-3})	1×10^{2}
Ca-45								8×10^{3}	(6×10^{2})	3×10^{3}
Cr-51								6×10^{3}		7×10^{1}
Mn-54	4×10^{-1}	1×10^{1}	(9×10^{-2})	1×10^{0}	2×10^{1}	6×10^{1}	1×10^{0}	4×10^{0}	(4×10^{-2})	2×10^{-1}
Fe-55	1×10^{4}	2×10^{5}	$(2 \times 10^{2)}$	1×10^{3}	2×10^{5}	2×10^{5}	2×10^{5}	2×10^{4}	(2×10^{1})	4×10^{2}
Fe-59								3×10^{1}		2×10^{0}
Co-57								6×10^{1}		3×10^{0}
Co-58			(3×10^{0})					2×10^{1}	(3×10^{-2})	1×10^{0}
Co-60	1×10^{-1}	2×10^{0}	(1×10^{-1})	3×10^{-1}	5×10^{0}	2×10^{1}	3×10^{-1}	7×10^{-1}	(1×10^{-2})	6×10^{2}
Ni-63	2×10^{4}	2×10^{5}	(2×10^{3})	4×10^{4}	1×10^{6}	5×10^{4}	1×10^{5}	2×10^{4}	(1×10^{2})	8×10^{3}
Zn-65	6×10^{-1}		(2×10^{0})	2×10^{0}	3×10^{1}	8×10^{1}	2×10^{0}	6×10^{0}	(5×10^{-2})	5×10^{1}
Sr-89								3×10^{3}		6×10^{1}

Sr-90	5×10^{1}	4×10^{1}	(2×10^{0})	2×10^{2}	8×10^{2}	3×10^{2}	3×10^{2}	9×10^{1}	(2×10^{-3})	7×10^{0}
Y-90								1×10^{14}		3×10^{2}
Nb-94	2×10^{-1}		(2×10^{-2})	5×10^{-1}	7×10^{0}	3×10^{1}	5×10^{-1}	1×10^{0}	(2×10^{-2})	7×10^{-2}
Tc-99m								∞		2×10^{1}
Tc-99	7×10^{3}		(3×10^{1})	2×10^{4}			5×10^{4}	3×10^{3}	(5×10^{-2})	2×10^{3}
Ru-106			(4×10^{-3})		4×10^{1}	7×10^{1}		1×10^{1}	(2×10^{-2})	9×10^{-1}
Ag-110m					6×10^{0}	2×10^{1}		1×10^{0}	9×10^{-2}	
Cd-109								2×10^{2}		1×10^{1}
In-111								4×10^{12}		6×10^{0}
I-123								∞		2×10^{1}
I-125								3×10^{3}		3×10^{1}
I-129								2×10^{2}	(8×10^{-4})	1×10^{1}
I-131								4×10^{5}		6×10^{0}
Sb-124								7×10^{0}		9×10^{-1}
Cs-134		5×10^{0}	(1×10^{-2})		1×10^{1}	3×10^{1}		1×10^{0}	(2×10^{-2})	9×10^{-2}
Cs-137	5×10^{-1}	8×10^{0}	(6×10^{-3})	1×10^{0}	2×10^{1}	9×10^{1}	1×10^{0}	3×10^{0}	(3×10^{-2})	2×10^{-1}
Ce-144			(2×10^{0})					7×10^{1}	(2×10^{-2})	6×10^{0}
Pm-147			(2×10^{2})					2×10^{3}	(1×10^{2})	5×10^{2}
Eu-152	4×10^{-1}		(3×10^{-3})	1×10^{0}	1×10^{1}	5×10^{1}	1×10^{0}	1×10^{0}	(3×10^{-2})	1×10^{-1}
Ir-192								2×10^{1}		1×10^{0}
Au-198								7×10^{12}		6×10^{0}
Tl-201								6×10^{12}		3×10^{1}
Pb-210								7×10^{0}		7×10^{-1}
Po-210								2×10^{1}		3×10^{0}
Th-228								2×10^{-1}		6×10^{-2}
Th-230								3×10^{-1}		1×10^{-1}
Th-232								8×10^{-2}		2×10^{-2}

TABLE I.3. (cont.)

Radio-nuclide	Steel			Aluminium		Copper	Concrete			All four materials
	IAEA[a] [4]	CEC[b] [11]	Elert et al. [30]	IAEA[a] [4]	Garbay et al.[c] [32]	Garbay et al.[c] [32]	IAEA [4]	Haristoy et al. [33]	Elert et al. [30]	Guetat et al. [28]
Ra-226								6×10^{-1}		7×10^{-2}
Ra-228								2×10^{0}		1×10^{-1}
U-234					2×10^{0}	7×10^{0}		5×10^{-1}		2×10^{-1}
U-235					1×10^{0}	7×10^{0}		5×10^{-1}		2×10^{-1}
U-238	1×10^{0}			5×10^{0}	2×10^{0}	7×10^{0}	3×10^{0}	5×10^{-1}		2×10^{-1}
Np-237					1×10^{1}	2×10^{0}		2×10^{-1}		4×10^{-2}
Pu-239	3×10^{-1}	1×10^{-1}	(7×10^{-3})	1×10^{0}	1×10^{1}	2×10^{0}	9×10^{-1}	3×10^{-1}	(9×10^{-3})	8×10^{-2}
Pu-240			(7×10^{-3})					3×10^{-1}	(9×10^{-3})	8×10^{-2}
Pu-241	1×10^{1}		(4×10^{-1})	7×10^{1}			5×10^{1}	2×10^{1}	(5×10^{-1})	2×10^{0}
Am-241	3×10^{-1}	1×10^{-1}	(6×10^{-3})	1×10^{0}	1×10^{1}	2×10^{0}	9×10^{-1}	3×10^{-1}	(9×10^{-3})	8×10^{-2}
Cm-244		2×10^{-1}	(1×10^{-2})		3×10^{1}	3×10^{0}		4×10^{-1}	(2×10^{-2})	1×10^{-1}

Notes: The values listed from each of the studies are the most limiting from the various exposure scenarios considered.

Except where noted, values are based on an individual dose level of 10 μSv/a.

( ) means not considered in the subsequent analysis (see text).

[a] Base case results — dilution and partitioning not included.

[b] Includes a dilution factor of 10.

[c] See text for dose criteria used in these studies.

Haristoy et al. [33]

I.27. Clearance levels for radionuclides in concrete arising from the dismantling of nuclear installations are derived by considering radiation exposures due to concrete disposal, recycling and reuse. The radiological basis is the same as that described in Garbay et al. [32], but the results in Table I.3 have been normalized to an individual dose criterion of 10 μSv/a. The annual quantity of recycled contaminated concrete is assumed to be 2000 t. The ratio of radioactive to non-radioactive material is usually taken to be 0.01 but in some of the scenarios, higher ratios are considered. The dosimetry used is based on ICRP 60 [12].

Guetat et al. [28]

I.28. This study is described in para. I.11.

Summary of the results of recycle studies

I.29. Comparison of the values in Table I.3 is complicated because:

(a) Assumptions in the various studies differ with respect to dilution of active material with inactive material in the recycling processes;
(b) One of the studies uses a dose criterion of 10 μSv/a for groups where there is a likelihood of exposures to other cleared practices and 50 μSv/a for groups where the possibility of exposures to other cleared practices is small.

These factors influence the values derived in the various studies and the agreement between the studies would be better if normalization was possible.

I.30. The following observations can be made on the results of the studies in Table I.3:

(a) The agreement between derived values for a given radionuclide is quite good, despite the complicating factors mentioned above, and is usually within a factor of 100. The range of values among radionuclides is about 10^5.
(b) The most limiting radionuclides are the long lived alpha emitters (^{232}Th, ^{239}Pu, ^{241}Am, etc.) and the high energy photon emitters (e.g. ^{60}Co).

Reuse

I.31. Table I.4 contains the results of four studies on the direct reuse of materials. The IAEA and CEC studies are described in paras I.22 and I.23.

TABLE I.4. SUMMARY OF RESULTS OF STUDIES ON CLEARANCE LEVELS FOR REUSE

Radionuclide	Bq/g		Bq/cm^2				
	IAEA[a] [4]	NUREG[a] [34]	IAEA[b] [4]	IAEA[c] [4]	CEC[d] [11]	Haristoy et al. [33]	NUREG[b] [34]
H-3		1×10^5				1×10^6	3×10^3
C-14		4×10^3				2×10^4	1×10^2
Na-22		5×10^{-1}				2×10^{-1}	2×10^{-1}
Na-24		3×10^{-1}				2×10^{16}	1×10^{-1}
P-32		2×10^2				4×10^3	2×10^1
S-35		2×10^4				2×10^4	4×10^2
Cl-36		1×10^3				2×10^3	6×10^1
Ca-45		2×10^3				8×10^3	6×10^1
Cr-51		3×10^1				2×10^3	1×10^1
Mn-54	4×10^0	1×10^0	4×10^{-1}	4×10^1	2×10^0	8×10^0	6×10^{-1}
Fe-55	9×10^2	2×10^3	9×10^1	5×10^3	4×10^2	4×10^3	8×10^1
Fe-59		1×10^0				4×10^1	4×10^{-1}
Co-57		1×10^1				7×10^1	3×10^0
Co-58		1×10^0				4×10^1	5×10^{-1}
Co-60	1×10^0	4×10^{-1}	1×10^{-1}	1×10^1	5×10^{-1}	2×10^0	2×10^{-1}
Ni-63	2×10^4	1×10^4	3×10^3	1×10^4	5×10^2	3×10^4	3×10^2
Zn-65	6×10^0	2×10^0	6×10^{-1}	5×10^1		1×10^1	7×10^{-1}
Sr-89		3×10^2				9×10^2	2×10^1
Sr-90	7×10^1	8×10^1	1×10^1	4×10^1	2×10^0	2×10^2	1×10^0
Y-90		1×10^2				7×10^6	1×10^1
Nb-94	2×10^0	7×10^{-1}	2×10^{-1}	2×10^1		2×10^0	3×10^{-1}
Tc-99m		1×10^1				∞	4×10^0
Tc-99	9×10^3	2×10^3	1×10^3	5×10^3		6×10^3	5×10^1
Ru-106		5×10^0				3×10^1	2×10^0
Ag-110m		4×10^{-1}				3×10^0	2×10^{-1}
Cd-109		2×10^2				1×10^2	8×10^0
In-111		4×10^0				1×10^6	1×10^0
I-123						3×10^{20}	
I-125		8×10^1				7×10^2	3×10^0
I-129		3×10^1				1×10^2	7×10^{-1}
I-131		3×10^0				3×10^3	1×10^0
Sb-124		6×10^{-1}				1×10^1	2×10^{-1}

TABLE I.4. (cont.)

Radio-nuclide	Bq/g		Bq/cm^2				
	IAEA[a] [4]	NUREG[a] [34]	IAEA[b] [4]	IAEA[c] [4]	CEC[d] [11]	Haristoy et al. [33]	NUREG[b] [34]
Cs-134		7×10^{-1}			4×10^{0}	3×10^{0}	2×10^{-1}
Cs-137	4×10^{0}	2×10^{0}	4×10^{-1}	4×10^{1}	5×10^{0}	7×10^{0}	8×10^{-1}
Ce-144		7×10^{1}				1×10^{2}	6×10^{0}
Pm-147		5×10^{3}				2×10^{3}	1×10^{3}
Eu-152	4×10^{0}	9×10^{-1}	4×10^{-1}	4×10^{1}		3×10^{0}	4×10^{-1}
Ir-192		1×10^{0}				4×10^{1}	6×10^{-1}
Au-198						2×10^{6}	
Tl-201						3×10^{6}	
Pb-210		1×10^{0}				9×10^{0}	7×10^{-2}
Po-210		4×10^{0}				1×10^{1}	9×10^{-2}
Th-228		2×10^{0}				2×10^{-1}	5×10^{-2}
Th-230		2×10^{0}				4×10^{-1}	6×10^{-2}
Th-232		5×10^{-1}				1×10^{-1}	1×10^{-2}
Ra-226		7×10^{0}				2×10^{0}	2×10^{-1}
Ra-228		8×10^{0}				4×10^{0}	2×10^{-1}
U-234		6×10^{0}				7×10^{-1}	1×10^{-1}
U-235		4×10^{0}				6×10^{-1}	1×10^{-1}
U-238	1×10^{0}	6×10^{0}	1×10^{0}	4×10^{0}		7×10^{-1}	2×10^{-1}
Np-237		7×10^{-1}				3×10^{-1}	3×10^{-2}
Pu-239	3×10^{-1}	2×10^{0}	2×10^{-1}	7×10^{-1}	6×10^{-2}	3×10^{-1}	6×10^{-2}
Pu-240		2×10^{0}				3×10^{-1}	6×10^{-2}
Pu-241	2×10^{1}	1×10^{2}	1×10^{1}	4×10^{1}		2×10^{1}	3×10^{0}
Am-241	3×10^{-1}	9×10^{-1}	2×10^{-1}	7×10^{-1}	6×10^{-2}	3×10^{-1}	2×10^{-2}
Cm-244		2×10^{0}			1×10^{-2}	5×10^{-1}	4×10^{-2}

Notes: The values listed from each of the studies are the most limiting from the various exposure scenarios considered.

Values are based on an individual dose level of 10 μSv/a.

[a] Building renovation.

[b] Occupancy in contaminated building.

[c] Handling tools, small motors, large pumps and other equipment.

[d] Leaching of total surface contamination into drinking water tank.

Haristoy et al. [33]

I.32. Clearance levels for surface contamination are derived for the reuse of buildings as offices. The radiological criterion used as the basis for the clearance levels was 50 μSv/a. The values have been normalized to a basis of 10 μSv/a. The dosimetry is based on ICRP 60 [12].

NUREG [34]

I.33. This report was prepared by a contractor for the United States Nuclear Regulatory Commission (NRC) to provide a technical basis for calculating the residual radioactive contamination limits allowable for the unrestricted release of decommissioned facilities. It was issued as a draft report for comment. It examines, in a generic way, the exposure scenarios which could be related to the decommissioning of buildings and land. For buildings, two scenarios were considered; these are building renovation (subsurface or volume sources) and normal occupancy (thin layer sources and removable surface sources). Results are presented in terms of annual dose per unit activity concentration or surface activity concentration. Clearance levels have been derived on the basis of an individual dose criterion of 10 μSv/a. The study uses ICRP 30 dosimetry [25].

Summary of the results of reuse studies

I.34. In the case of direct reuse of materials, for example, tools, equipment and buildings, surface contamination gives rise to the main routes of exposure and the exempt levels are expressed in units of Bq/cm^2 as well as Bq/g. The values of surface activity concentration (Bq/cm^2) refer to total surface activity (fixed plus removable). The following observations can be made regarding Table I.4:

(a) Although the reuse scenarios are very different from each other, the results for a given radionuclide are generally within a factor of 100.
(b) The alpha emitters ^{232}Th, ^{239}Pu and ^{241}Am are limiting. The most restrictive levels for these radionuclides usually result from the inhalation pathway, but in the CEC study [11] a scenario involving ingestion also results in low calculated clearance levels.

Unconditional clearance (Guetat et al. [35])

I.35. Table I.5 summarizes the results of a study to establish unconditional clearance levels for materials from nuclear fuel cycle installations. The study proposes activity concentration as well as surface contamination levels. The radiological criteria used as a basis for the derived clearance levels are 10 μSv/a for groups for which it is

TABLE I.5. SUMMARY OF RESULTS OF A STUDY ON UNCONDITIONAL CLEARANCE [35]

Radionuclide	Bq/g	Bq/cm^2
H-3	1×10^4	1×10^4
C-14	1×10^3	1×10^3
Na-22	3×10^0	1×10^1
Na-24	2×10^0	1×10^1
P-32	1×10^4	3×10^2
S-35	2×10^5	1×10^3
Cl-36	2×10^4	4×10^2
Ca-45	7×10^4	6×10^2
Cr-51	2×10^2	1×10^3
Mn-54	7×10^0	4×10^1
Fe-55	2×10^5	1×10^4
Fe-59	5×10^0	4×10^1
Co-57	5×10^1	3×10^2
Co-58	6×10^0	5×10^1
Co-60	2×10^0	1×10^1
Ni-63	1×10^5	2×10^4
Zn-65	1×10^1	7×10^1
Sr-89	1×10^4	4×10^2
Sr-90	2×10^3	1×10^2
Y-90	7×10^3	4×10^2
Nb-94	4×10^0	2×10^1
Tc-99m	8×10^1	5×10^2
Tc-99	6×10^4	6×10^2
Ru-106	3×10^1	1×10^2
Ag-110m	2×10^0	1×10^1
Cd-109	9×10^2	4×10^2
In-111	2×10^1	1×10^2
I-123	5×10^1	3×10^2
I-125	5×10^1	5×10^2
I-129	8×10^2	7×10^1
I-131	2×10^1	1×10^2
Sb-124	3×10^0	2×10^1
Cs-134	4×10^0	2×10^1
Cs-137	1×10^1	4×10^1
Ce-144	1×10^2	2×10^2
Pm-147	1×10^4	7×10^2

TABLE I.5. (cont.)

Radionuclide	Bq/g	Bq/cm^2
Eu-152	5×10^0	2×10^1
Ir-192	8×10^0	5×10^1
Au-198	2×10^1	1×10^2
Tl-201	8×10^1	4×10^2
Pb-210	3×10^1	4×10^0
Po-210	5×10^1	9×10^0
Th-228	1×10^0	3×10^{-1}
Th-230	1×10^0	2×10^{-1}
Th-232	3×10^{-1}	5×10^{-2}
Ra-226	6×10^0	1×10^1
Ra-228	7×10^0	2×10^1
U-234	3×10^0	6×10^{-1}
U-235	3×10^0	6×10^{-1}
U-238	4×10^0	7×10^{-1}
Np-237	1×10^0	2×10^{-1}
Pu-239	1×10^0	1×10^{-1}
Pu-240	1×10^0	1×10^{-1}
Pu-241	4×10^1	7×10^0
Am-241	1×10^0	1×10^{-1}
Cm-244	2×10^0	3×10^{-1}

Notes: The values listed from each of the studies are the most limiting from the various exposure scenarios considered.

See text for dose criteria used in this study.

judged that there is the possibility of exposures from other cleared practices and 50 μSv/a for groups where exposures from other cleared practices are unlikely. Normalization of the results to a basis of 10 μSv/a is complicated and the values in Table I.5 are uncorrected. The dosimetry is based upon ICRP 60 [12].

Overall summary of results

I.36. The extremes of the overall range of activity concentrations obtained from Tables I.1 to I.5 for each of the radionuclides are listed in Table I.6. In the next section they are used as the basis for deriving unconditional clearance levels. The lowest assessed values are invariably associated with two types of radionuclide.

— Energetic photon emitters, of which ^{60}Co is the most important example, and
— Alpha emitters such as ^{239}Pu.

DERIVATION OF UNCONDITIONAL CLEARANCE LEVELS FROM THE RESULTS OF THE ASSESSMENT STUDIES

I.37. The ranges of values of activity concentrations obtained from the review of reports on landfill, incineration, recycling and reuse analyses (Table I.6) have been used as the basis for deriving a single set of clearance levels.

I.38. It was noted in the previous paragraphs that some differences exist in the dosimetry used in the studies reviewed. Both ICRP 30 and ICRP 60 based dosimetry has been used. However, the differences introduced by this are not large in relation to the variability of the results of assessments (less than a factor of two) [36] and for present purposes no correction has been made.

I.39. In the following, a procedure is described for deriving clearance levels from the ranges of values summarized in Table I.6. The aim of the procedure is to select values which provide a high degree of assurance that doses from likely scenarios will not exceed 10 μSv/a. At the same time it is considered desirable to avoid having the final values determined by unlikely scenarios. The procedure allows doses from unlikely scenarios up to about 100 μSv/a.

I.40. The assessed values vary from study to study and from scenario to scenario for each radionuclide (except where only one author studied a particular nuclide). A method was therefore developed to categorize the nuclide values into groups by order of magnitude. Such grouping was considered to be appropriate because greater precision is not considered to be warranted by the technical information available. The method for grouping is as follows. For each radionuclide the lowest value is determined. The lowest value is multiplied by 10 and that result compared to the next lowest value reported. The lesser of 10 times the lowest value and the next lowest value is chosen. The radionuclide is then categorized by the order of magnitude of the chosen value. For example, if the chosen value from the method is 700 Bq/g, the category for that nuclide would be from 100 to 1000 Bq/g.

I.41. The above method has certain weaknesses. First, the setting of the category divisions at 0.1, 1, 10, etc., is rather arbitrary. For example, if the clearance level for one radionuclide is calculated to be 0.9 Bq/g and for a second radionuclide as 1.0 Bq/g they would be classified as a factor of 10 apart. However, this is a weakness inherent in all classification schemes. The second weakness lies in the fact that not all authors studied all radionuclides. Therefore if the results of one author are consistently lower than those of the other authors and if the author did not study a particular radionuclide, it may appear from application of the method that the radionuclide which was studied less is allocated a higher than expected clearance level. These two weaknesses lead to a number of cases where radionuclides with similar emission

TABLE I.6. RANGES OF RESULTS OBTAINED FROM LANDFILL, INCINERATION, RECYCLING AND REUSE ANALYSES (Bq/g)

Radionuclide	Range	
	Low	High
H-3	2×10^{3}	1×10^{6}
C-14	1×10^{2}	7×10^{3}
Na-22	7×10^{-2}	1×10^{1}
Na-24	3×10^{-1}	∞
P-32	2×10^{2}	2×10^{5}
S-35	2×10^{3}	4×10^{5}
Cl-36	6×10^{1}	2×10^{4}
Ca-45	1×10^{3}	2×10^{5}
Cr-51	3×10^{1}	6×10^{3}
Mn-54	2×10^{-1}	6×10^{1}
Fe-55	4×10^{2}	6×10^{5}
Fe-59	1×10^{0}	3×10^{1}
Co-57	3×10^{0}	6×10^{1}
Co-58	1×10^{0}	4×10^{1}
Co-60	6×10^{-2}	9×10^{1}
Ni-63	8×10^{3}	1×10^{6}
Zn-65	5×10^{-1}	8×10^{1}
Sr-89	6×10^{1}	1×10^{4}
Sr-90	2×10^{0}	2×10^{3}
Y-90	1×10^{2}	1×10^{14}
Nb-94	7×10^{-2}	3×10^{1}
Tc-99m	1×10^{1}	∞
Tc-99	4×10^{1}	6×10^{4}
Ru-106	5×10^{0}	2×10^{2}
Ag-110m	9×10^{-2}	2×10^{1}
Cd-109	1×10^{1}	8×10^{3}
In-111	3×10^{0}	4×10^{12}
I-123	1×10^{1}	∞
I-125	3×10^{1}	3×10^{3}
I-129	1×10^{1}	8×10^{2}
I-131	3×10^{0}	4×10^{5}
Sb-124	6×10^{-1}	7×10^{0}
Cs-134	9×10^{-2}	3×10^{1}
Cs-137	2×10^{-1}	9×10^{1}

TABLE I.6. (cont.)

Radionuclide	Range	
	Low	High
Ce-144	6×10^{0}	1×10^{2}
Pm-147	5×10^{2}	2×10^{7}
Eu-152	1×10^{-1}	5×10^{1}
Ir-192	1×10^{0}	2×10^{1}
Au-198	3×10^{0}	7×10^{12}
Tl-201	2×10^{1}	6×10^{12}
Pb-210	2×10^{-1}	1×10^{2}
Po-210	3×10^{0}	2×10^{2}
Th-228	6×10^{-2}	4×10^{0}
Th-230	1×10^{-1}	8×10^{0}
Th-232	2×10^{-2}	1×10^{1}
Ra-226	7×10^{-2}	2×10^{2}
Ra-228	1×10^{-1}	3×10^{2}
U-234	2×10^{-1}	5×10^{1}
U-235	2×10^{-1}	3×10^{1}
U-238	2×10^{-1}	5×10^{1}
Np-237	4×10^{-2}	1×10^{1}
Pu-239	8×10^{-2}	4×10^{2}
Pu-240	8×10^{-2}	6×10^{0}
Pu-241	2×10^{0}	1×10^{2}
Am-241	8×10^{-2}	2×10^{2}
Cm-244	1×10^{-1}	3×10^{1}

energies or dose factors appear in different categories. Expert judgement has been used to correct for these two weaknesses.

I.42. It should be noted that for studies of the ^{238}U and ^{232}Th decay series, none of the scenarios included gave explicit consideration to the possibility of exposure arising from indoor radon. Indoor radon could be important if buildings were to be constructed on a landfill disposal site. Furthermore, radon emanations from building materials made of recycled or reused materials were not analysed in the studies. Therefore, the categorization in this report makes no allowance for possible radon exposures, and national authorities should be aware of this when determining the applicability of the categories.

I.43. In some scenarios involving beta emitters, the dominant exposure pathway is expected to be due to skin dose. In only a few of the studies used to develop the categories was skin dose considered. The skin doses that might be delivered by beta emitters were evaluated by comparing the appropriate representative activity concentrations in Table I.7 with the activity concentrations given by Harvey et al. [37], which did consider skin doses. It was found that, at these representative activity concentrations, no radionuclide would give rise to localized skin doses approaching the ICRP skin dose limit for members of the public (50 mSv/a).

I.44. For the sake of simplicity in cases where a single value clearance level is required, use of three times the lower limit of the category range is suggested (the approximate logarithmic mean of the category). For radionuclides that were not categorized in the report it is recommended that the formula described in para. I.48 be applied to substitute for a literature value, and that the radionuclide is then categorized as described above.

I.45. Although the number of derived clearance levels in terms of surface contamination is limited, the results suggest that it may be appropriate to assume numerical equivalence between levels expressed in Bq/g and Bq/cm^2.

I.46. The assessment studies used to develop clearance levels generally lack consideration of decay between the time that a material is cleared and the time of possible human exposure. This assumption is likely to be conservative for very short lived radionuclides such as ^{131}I, although it should not pose an undue burden when implementing the clearance levels, because in most cases it should be straightforward to store temporarily materials to be cleared until very short lived isotopes have decayed. For some isotopes possible doses could increase with time from ingrowth of progeny. The significance of this effect varies depending on the parent isotope and the exposure scenario, and in any case would appear to involve isotopes that are already in the most restrictive clearance category.

I.47. The categories discussed in this publication were derived using studies from 1992 and earlier. In the future, as more information becomes available and more experience with large scale decommissioning and conditional clearance is gained, it will be necessary to review the categorization of the radionuclides presented here.

DERIVATION OF CLEARANCE LEVELS USING A FORMULA

I.48. The method of derivation of clearance levels for the 56 radionuclides listed in Tables I.1 to I.6 has been described in the previous paragraphs. For other radionuclides, a fitting formula may be used for assigning radionuclides to the categories of Table I.

TABLE I.7. APPLICATION OF FORMULA

Radio-nuclide	Effective γ energy (MeV)	Effective β energy (MeV)	ALI_{inh} (Bq)	ALI_{inh} (Bq)	Formula values	Values from Table I
H-3		5.68×10^{-3}	1×10^{9}	1×10^{9}	2×10^{3}	3×10^{3}
C-14		4.95×10^{-2}	4×10^{7}	4×10^{7}	2×10^{2}	3×10^{2}
Na-22	2.19×10^{0}	1.94×10^{-1}	1×10^{7}	7×10^{6}	5×10^{-1}	3×10^{-1}
Na-24	4.12×10^{0}	5.53×10^{-1}	6×10^{7}	5×10^{7}	2×10^{-1}	3×10^{-1}
P-32		6.95×10^{-1}	5×10^{6}	8×10^{6}	1×10^{1}	3×10^{2}
S-35		4.88×10^{-2}	3×10^{7}	7×10^{7}	2×10^{2}	3×10^{3}
Cl-36	1.55×10^{-4}	2.74×10^{-1}	3×10^{6}	2×10^{7}	4×10^{1}	3×10^{2}
Ca-45	4.35×10^{-8}	7.72×10^{-2}	1×10^{7}	2×10^{7}	1×10^{2}	3×10^{3}
Cr-51	3.26×10^{-2}	3.86×10^{-3}	2×10^{8}	4×10^{8}	3×10^{1}	3×10^{1}
Mn-54	8.35×10^{-1}	4.22×10^{-3}	1×10^{7}	3×10^{7}	1×10^{0}	3×10^{-1}
Fe-55	1.69×10^{-3}	4.20×10^{-3}	3×10^{7}	1×10^{8}	5×10^{2}	3×10^{2}
Fe-59	1.19×10^{0}	1.17×10^{-1}	5×10^{6}	1×10^{7}	8×10^{-1}	3×10^{0}
Co-57	1.25×10^{-1}	1.86×10^{-2}	8×10^{6}	6×10^{7}	8×10^{0}	3×10^{1}
Co-58	9.75×10^{-1}	3.41×10^{-2}	7×10^{6}	2×10^{7}	1×10^{0}	3×10^{0}
Co-60	2.5×10^{0}	9.65×10^{-2}	4×10^{5}	3×10^{6}	4×10^{-1}	3×10^{-1}
Ni-63		1.71×10^{-2}	1×10^{7}	1×10^{8}	5×10^{2}	3×10^{3}
Zn-65	5.84×10^{-1}	6.87×10^{-3}	4×10^{6}	5×10^{6}	2×10^{0}	3×10^{-1}
Sr-89	8.45×10^{-5}	5.83×10^{-1}	2×10^{6}	6×10^{6}	2×10^{1}	3×10^{2}
Sr-90[a]	1.69×10^{-6}	1.13×10^{0}	6×10^{4}	6×10^{5}	5×10^{0}	3×10^{0}
Y-90	1.69×10^{-6}	9.35×10^{-1}	7×10^{6}	5×10^{6}	1×10^{1}	3×10^{2}
Nb-94	1.57×10^{0}	1.68×10^{-1}	2×10^{5}	9×10^{6}	6×10^{-1}	3×10^{-1}
Tc-99m	1.26×10^{-1}	1.62×10^{-2}	2×10^{9}	1×10^{9}	8×10^{0}	3×10^{1}
Tc-99		1.01×10^{-1}	8×10^{6}	3×10^{7}	1×10^{2}	3×10^{2}
Ru-106[a]	2.01×10^{-1}	1.4×10^{0}	2×10^{5}	2×10^{6}	3×10^{0}	3×10^{0}
Ag-110m	2.74×10^{0}	7.13×10^{-2}	1×10^{6}	7×10^{6}	4×10^{-1}	3×10^{-1}
Cd-109	2.64×10^{-2}	8.27×10^{-2}	1×10^{6}	9×10^{6}	3×10^{1}	3×10^{2}
In-111	4.05×10^{-1}	3.44×10^{-2}	9×10^{7}	5×10^{7}	2×10^{0}	3×10^{0}
I-123	1.71×10^{-1}	2.8×10^{-2}	2×10^{8}	9×10^{7}	6×10^{0}	3×10^{1}
I-125	4.2×10^{-2}	1.94×10^{-2}	2×10^{6}	1×10^{6}	1×10^{1}	3×10^{1}
I-129	2.46×10^{-2}	6.38×10^{-2}	3×10^{5}	2×10^{5}	2×10^{0}	3×10^{1}
I-131	3.8×10^{-1}	1.9×10^{-1}	1×10^{6}	8×10^{5}	3×10^{0}	3×10^{1}
Sb-124	1.8×10^{0}	3.84×10^{-1}	1×10^{7}	6×10^{6}	5×10^{-1}	3×10^{-1}
Cs-134	1.55×10^{0}	1.63×10^{-1}	2×10^{6}	1×10^{6}	6×10^{-1}	3×10^{-1}
Cs-137[a]	5.63×10^{-1}	2.48×10^{-1}	2×10^{6}	1×10^{6}	2×10^{0}	3×10^{-1}
Ce-144[a]	5.21×10^{-2}	1.28×10^{0}	2×10^{5}	2×10^{6}	6×10^{0}	3×10^{1}

TABLE I.7. (cont.)

Radio-nuclide	Effective γ energy (MeV)	Effective β energy (MeV)	ALI_{inh} (Bq)	ALI_{inh} (Bq)	Formula values	Values from Table I
Pm-147	4.37×10^{-6}	6.2×10^{-2}	2×10^{6}	5×10^{7}	2×10^{2}	3×10^{3}
Eu-152	1.14×10^{0}	1.36×10^{-1}	4×10^{5}	1×10^{7}	9×10^{-1}	3×10^{-1}
Ir-192	8.11×10^{-1}	2.14×10^{-1}	3×10^{6}	1×10^{7}	1×10^{0}	3×10^{0}
Au-198	4.04×10^{-1}	3.26×10^{-1}	2×10^{7}	1×10^{7}	2×10^{0}	3×10^{0}
Tl-201	9.32×10^{-2}	4.33×10^{-2}	4×10^{8}	3×10^{8}	1×10^{1}	3×10^{1}
Pb-210[a]	4.81×10^{-3}	4.27×10^{-1}	1×10^{4}	2×10^{4}	2×10^{-1}	3×10^{-1}
Po-210	8.5×10^{-6}	8.18×10^{-8}	1×10^{4}	9×10^{4}	9×10^{-0}	3×10^{0}
Th-228[a]	1.74×10^{0}	8.70×10^{-1}	2×10^{2}	3×10^{5}	2×10^{-1}	3×10^{-1}
Th-230	1.55×10^{-3}	1.46×10^{-2}	4×10^{2}	3×10^{5}	4×10^{-1}	3×10^{-1}
Th-232	1.33×10^{-3}	1.25×10^{-2}	9×10^{1}	5×10^{4}	9×10^{-2}	3×10^{-1}
Ra-226[a]	1.71×10^{0}	9.40×10^{-1}	9×10^{3}	9×10^{4}	6×10^{-1}	3×10^{-1}
Ra-228[a]	9.30×10^{-1}	4.77×10^{-1}	2×10^{4}	7×10^{4}	7×10^{-1}	3×10^{-1}
U-234	1.73×10^{-3}	1.32×10^{-2}	6×10^{2}	7×10^{5}	6×10^{-1}	3×10^{-1}
U-235[a]	2.27×10^{-1}	2.74×10^{-1}	6×10^{2}	7×10^{5}	6×10^{-1}	3×10^{-1}
U-238[a]	2.19×10^{-2}	8.90×10^{-1}	6×10^{2}	8×10^{5}	6×10^{-1}	3×10^{-1}
Np-237[a]	2.37×10^{-1}	2.63×10^{-1}	3×10^{2}	3×10^{4}	3×10^{-1}	3×10^{-1}
Pu-239	7.96×10^{-4}	6.65×10^{-3}	3×10^{2}	4×10^{4}	3×10^{-1}	3×10^{-1}
Pu-240	1.73×10^{-3}	1.06×10^{-2}	3×10^{2}	4×10^{4}	3×10^{-1}	3×10^{-1}
Pu-241[a]	6.02×10^{-6}	5.24×10^{-3}	2×10^{4}	2×10^{6}	2×10^{1}	3×10^{1}
Am-241	3.24×10^{-2}	5.19×10^{-2}	3×10^{2}	3×10^{4}	3×10^{-1}	3×10^{-1}
Cm-244	1.7×10^{-3}	8.59×10^{-3}	5×10^{2}	6×10^{4}	5×10^{-1}	3×10^{-1}

[a] Short lived daughter products have been taken into account.

I.49. The formula comprises three relationships; they take account of radiation exposure due to external irradiation, inhalation and ingestion. The clearance level in Bq/g (or Bq/cm^2) may be evaluated from:

$$\text{Minimum} \left\{ \frac{1}{E_\gamma + 0.1E_\beta}, \quad \frac{ALI_{inh}}{1000}, \quad \frac{ALI_{ing}}{100\,000} \right\}$$

where E_γ and E_β are the effective energies in MeV for beta and gamma emissions respectively, as given in ICRP Publication No. 38 [13] and ALI_{inh} and ALI_{ing} are the most restrictive values of the annual limits on intake by inhalation and ingestion in Bq, as given in ICRP Publication 61 [14].

1.50. The numerical terms in the formula were obtained by adjusting their values so as to obtain a fit to the lower end of the ranges of values in Table I.6 for the most important radionuclides.

I.51. For most of the radionuclides considered in Tables I.1 to I.5, the values given by the formula are within the range of maximum and minimum values obtained in the landfill disposal, incineration, recycle and reuse assessments and are not greater than the lowest assessed value by more than a factor of 10. An example of the application of the formula is shown in Table I.7; values from the categorization scheme shown in Table I of the main text are included for comparison.

Appendix II

COLLECTIVE DOSE CONSIDERATIONS

II.1. Following the guidance of Safety Series No. 89 [3], after satisfying the first condition that individual doses are sufficiently low to not warrant regulatory concern, it is also necessary to ensure that radiation protection is optimized before a source or practice can be cleared from regulatory control. In this context, if the collective dose commitment resulting from one year of the practice would be less than about 1 man·Sv, the practice may be automatically cleared without further consideration of other possible management options. Where collective doses exceed 1 man·Sv per year of practice, it is necessary to give consideration to other options. It is still possible to clear, provided that it can be shown that the option chosen optimizes radiation protection and that individual risks remain trivial. The extent and formality of the analysis required for such a demonstration should reflect the magnitude of the collective dose commitment.

II.2. The proper evaluation of collective dose requires a clear specification of the disposal, recycling or reuse practice and site specific information regarding the population at risk, the exposure routes, food sources, etc. Such a specification is problematical for unconditional clearance. Nonetheless, it is possible to explore, in a general way, the likely magnitudes of collective doses which could result from cleared practices being considered in the context of disposal, recycling and reuse. This has been done in a number of studies related to the disposal and incineration of solid wastes [5, 10] to the recycling of steel [4, 11] and of concrete and aluminium [4], and to the reuse of contaminated buildings [4].

II.3. The magnitude of the collective dose is variable, depending upon the nature and size of the assumed practice and it is therefore difficult to provide a general perspective on its potential significance. The approach which has been adopted in the studies reviewed is to provide an estimate of the quantities of material which may be disposed of, recycled or reused as inferred by the 1 man·Sv collective dose value. This is done by evaluating the amount (in units of mass), at the most limiting of the concentrations corresponding to the individual clearance criterion which gives rise to 1 man·Sv of collective dose commitment.

II.4. The general conclusions of the studies with respect to collective dose can be summarized as follows:

(a) In none of the scenarios considered in the studies is collective dose limiting, i.e. the doses are well below 1 man·Sv per year of practice when activity concentrations are limited on the basis of the individual dose criterion of 10 μSv/a.

(b) Substantial quantities of material would have to be disposed of, recycled or reused before the 1 man·Sv criterion would be reached. This is true even when all of the material is assumed to be at the clearance level as determined from individual dose considerations. Clearly, this is unlikely to be the case in practice and larger quantitities would be needed before the collective dose criterion is reached.

II.5. Of the practices considered, the largest collective doses may be expected to arise from recycling of steel and other metals, especially if metals were to be recycled into consumer goods or into motor cars with which humans may come into close contact during their normal use. At the national level (in Europe), typically 1000 t/a of metal suitable for conventional disposal or recycling is currently arising from decommissioning operations. When the decommissioning of nuclear installations enters its main phase these amounts can be expected to be substantially higher, typically of the order of 10 000 t/a. In the case of ^{60}Co, which seems likely to be the most important radionuclide in the context of steel recycling and collective dose, the estimated annual recycled quantity which would give rise to 1 man·Sv is of the order of 10 000 t [4, 11]. However, this calculation assumes all of the steel to be at the level determined on the basis of the individual dose criterion for clearance. In practice, it is likely that only the maximum levels in recycled steel would approach this level and that the average concentration would be substantially lower. Further, it assumes that all of the recycled steel is directed into one or other of the products considered in the analysis. Again this is unlikely. It is more likely that the steel would be used in a variety of products, some of which, such as railway lines or bridge structures, would have little contact with humans during their use.

II.6. In summary, the generic analyses that have been conducted in relation to the disposal, incineration, recycling and reuse of solid materials imply that collective dose is not likely to be a limiting factor in establishing clearance levels except possibly where the practices involved are large scale.

Annex

CONCENTRATIONS OF NATURALLY OCCURRING RADIONUCLIDES IN COMMON MATERIALS

A-1. The purpose of this Annex is to present data on measured levels of naturally occurring radionuclides as a supplement to the discussion in Section 4 of the main text.

ACTIVITY CONCENTRATIONS IN SOIL, ROCK AND ORES

A-2. Concentrations of naturally occurring radionuclides may vary widely in soil and rock, depending on complex interactions among natural physical and chemical processes. Average activity concentrations are summarized in Tables A-I and A-II [38-40].

TABLE A-I. ACTIVITY CONCENTRATIONS IN SOIL (Bq/g)

Nuclide	Average concentration	Typical range
K-40	0.37	0.1–0.7
U-238[a]	0.025	0.01–0.05
Th-232	0.025	0.007–0.05

[a] Ra-226 should normally be similar.

TABLE A-II. AVERAGE RADIUM, URANIUM, THORIUM, AND POTASSIUM CONTENT IN VARIOUS ROCKS (Bq/g)

Rock type	Ra-226	U-238	Th-232	K-40
Igneous	0.048	0.048	0.048	0.81
Sedimentary				
sandstones	0.026	0.015	0.024	0.33
shales	0.040	0.015	0.041	0.81
Limestones	0.016	0.015	0.0052	0.08

TABLE A-III. WORLDWIDE RANGES OF REPORTED ACTIVITY CONCENTRATIONS IN COAL AND PHOSPHATE DEPOSITS (Bq/g)

Mineral	K-40	U-238	Ra-226	Pb-210	Po-210	Th-232	Ra-228
Coal[a]	0.044–0.76	0.002–0.14[b]	0.0005–0.1	0.01–0.05	0.01–0.041	0.0024–0.11	0.013–0.035
Phosphate[c]	0.01–0.23	0.044–4.8	0.03–4.8			0.007-0.11	

[a] Measurements reported by 15 countries.

[b] High value for coal; lignite has been reported up to 0.25 Bq/g.

[c] Measurements reported by 12 countries.

TABLE A-IV. ACTIVITY CONCENTRATIONS MEASURED IN ADDITIONAL ORES AND MINERALS (Bq/g)

Mineral	Country	K-40	Ra-226	Th-232	U-238
Peat	Sweden				0.040[a]
	Finland	0.028			0.016[b]
Oil shale	USA, Estonia	0.5		0.025	0.06
	USA, Morocco				0.35
Ilmenite	Australia		0.075[c]		
Zircon	Australia			>0.5	>0.5
sands	South Africa			>0.5	>0.5
Fireclay	USA			0.07	0.05
Bauxite	USA			0.2	0.25
Copper	USA			0.023–0.11	0.03–0.08

[a] Significantly higher concentrations have also been measured.
[b] Plus Pb-210 (0.030 Bq/g) and Ra-228 (0.0053 Bq/g).
[c] Plus 0.4 Bq/g of Ra-228 and Th-228.

A-3. Much higher concentrations exist in localized areas or deposits. As an extreme example, monazite sand deposits in India contain thorium in concentrations ranging from 8 to 10.5 wt% [38]. Tables A-III and A-IV list ranges of activity concentrations measured worldwide in coal and phosphate deposits [38], and in some additional ores and minerals [38, 39].

A-4. Uranium ores contain uranium and its daughter products in concentrations up to several thousand times higher than average soil levels. Currently exploited ores range from about 0.1 to 3.0% U_3O_8 [38]. Most ores contain little thorium, although exceptions include the Elliot Lake region in Canada where natural thorium averages about 0.2% [38]. Radium-226 exists in uranium ore at a level of about 1 Bq/g per 0.01% U_3O_8 [39].

ACTIVITY REDISTRIBUTION BY HUMAN ACTION

A-5. Human actions often redistribute radioactive elements from the state found in nature. Several examples are provided below.

TABLE A-V. ACTIVITY CONCENTRATIONS MEASURED IN METAL MINING AND PROCESSING WASTES (Bq/g)

Material	Notes	Ra-226	U-238	Th-232
Red mud	Bauxite refinery waste[a]		0.28	0.18
Brown mud	Bauxite refinery waste[a]		0.2	0.46
Furnace slag	Copper smelting waste[b]		0.11	0.19
Tailings	Copper mining waste[b]		0.03, 0.06[c]	0.01 0.11[c]
Leach material	Copper mining waste[b]		0.04–0.11	
Surface waste	Copper mining waste[b]		0.03–0.07	
Amang	By-products from processing tin tailings into concentrated ores[b]	16–18		43–327
Slag	Samples from two tin smelters[b]	0.27–2.0	up to 1.6	up to 0.7
Chloride waste	Slurried waste from titanium tetrachloride producers[b]	0.14–0.91	0.003–1.6	0.004–3.3

[a] Source: Ref. [39].
[b] Source: Ref. [41].
[c] From underground and open pit mines, respectively.

Metal mining and processing. Mineral recovery, concentration and purification processes generate large quantities of by-products such as tailings and slags. Table A-V presents activity concentrations measured in by-products from four metal mining industries.

Uranium mining and milling. Discarded overburden, rock, and very low grade ores may become sources of elevated activity, as may spent leach fields, spillages and other mining residues. One study has estimated ^{226}Ra concentrations in uranium mining overburden ranging from 0.11 Bq/g to two orders of magnitude higher [41]. Milling recovers more than 90% of the uranium in the ore, but leaves almost all of the ^{230}Th and its daughters, including radium, in tailings [38].

TABLE A-VI. REPORTED RANGES OF ACTIVITY CONCENTRATIONS IN PHOSPHATE FERTILIZERS (Bq/g)[a]

Type of fertilizer	U-238	Ra-226	Th-232	K-40
Treated rock phosphates	0.67	0.03–0.48	0.02–0.6	0.073–0.23
One-component phosphate fertilizers	0.529–2.1	0.11–0.91	0.015–0.048	0.052–0.18
PK fertilizers	0.41	0.37	0.015	5.9
NP fertilizers	0.92–2.3	0.02–0.85	0.01–0.063	0.041
NPK fertilizers	0.44–0.47	0.009–0.27	<0.015–0.054	1.2–5.9

[a] Po-210, Pb-210, and Th-230 have also been reported.

Phosphate industry. Phosphate rock is the raw material for all phosphate products, including fertilizers. Table A-VI summarizes activity concentrations measured in fertilizers [38]). By-products of the phosphate industry include large quantities of tailings, calcium silicate slag, and phosphogypsum. Radium-226 in concentrations of about 1.7 Bq/g have been measured in tailings from a US mine [41]. Activity concentrations of ^{226}Ra range from about 1.3 to 2.2 Bq/g in calcium silicate slag, and are typically about 0.9 Bq/g in phosphogypsum [39]. Radium-226 in concentrations up to 400 Bq/g has been measured in scale deposits in piping and vessels at phosphoric acid plants [39].

Energy generation. Coal combustion generates huge quantities of ash and slag in which radionuclide activities become concentrated (see Table A-VII). Combustion of peat concentrates naturally occurring radionuclides in peat ash by a factor of about 20 [39]. Elevated activities have been observed in scale deposited on piping and process equipment at geothermal power plants, and in filter cake generated during the treatment of spent brine [41].

Petroleum exploitation. Activity concentrations up to 1000 Bq/g of ^{226}Ra and ^{228}Ra have been observed on scale deposits within pumps, tubing, and equipment at oil and gas production facilities [39]. Extraction of petroleum from oil shale leaves most of the naturally occurring activity in the spent shale [39].

TABLE A-VII. ACTIVITY CONCENTRATIONS MEASURED IN COAL COMBUSTION BYPRODUCTS (Bq/g) [38]

Nuclide	Bottom ash (slag)		Collected fly ash		Escaping fly ash	
	Low	High	Low	High	Low	High
K-40	0.24	1.2	0.26	1.5	0.26	0.27
U-238	0.017	0.18	0.044	1.0	0.1	0.3
Ra-226	0.004	0.25	0.03	0.2	0.015	0.56
Pb-210	0.03	3.9	0.044	2.0	0.2	3.0
Po-210	0.0074	0.19	0.1	2.0	0.25	5.5
Th-232	0.015	0.12	0.03	0.3	0.04	0.1
Th-228	0.09	0.56			0.1	0.12
Ra-228	0.02	0.067	0.044	0.13	0.1	0.16

Wood ash. Wood ash has long been used as a fertilizer and for other applications. In a survey of wood ash obtained from 14 States in the United States, measured activity concentrations of ^{40}K ranged from 1.0 to 5.7 Bq/g [42].

DISCUSSION

A-6. Any facility, structure, or grounds may contain naturally occurring radionuclides at higher than 'normal' concentrations. A facility might be sited in an area having elevated levels of activity due to mineral or ore deposits, or human actions may have enhanced radionuclide concentrations in soils and construction materials.

A-7. Mining and coal combustion by-products have often been dispersed to the environment or used during the construction of structures and roads. Tailings have been used for construction fill, coal ash and calcium silicate slag for manufacture of concrete, and phosphogypsum for manufacture of cement, wallboard, and plaster [38, 39]. Table A-VIII illustrates activity concentrations measured in construction materials.

A-8. Elevated activities have been traced to mining operations that existed centuries ago, as in examples of sixteenth century mining residues found in Norway and Sweden. In some of these residues, mean activity concentrations for ^{232}Th, ^{226}Ra, and ^{40}K were 0.27, 2.9 and 1.73 Bq/g [43]. Other cases have occurred where radioactive contamination was traced to forgotten industrial activities. In Australia, for

TABLE A-VIII. ACTIVITY CONCENTRATIONS MEASURED IN CONSTRUCTION MATERIALS (Bq/g)[a] [40]

Country	Material	K-40	U-238	Ra-226	Th-232
China	Wood	3.3			
(Taiwan)	Red brick	0.59	0.044		
	Concrete	0.26	0.033		
UK	Clay brick	0.67	0.11	0.052	0.044
	Silicate brick (gravel)	0.37	0.007	0.007	0.004
	Granite	1.1	0.22	0.089	0.081
	Aerated concrete	0.70	0.015	0.089	0.015
	Natural gypsum	0.15	0.015	0.022	0.007
	Concrete block (fly ash)	(0.22–0.59)	(0.037–0.44)	(0.007–0.15)	(0.037–0.044)
USA	Cement	0.13	0.041		0.015
	Silica sand	0.33	0.011		0.019
	Commercial sand	0.26	0.011		0.11
	Red brick	0.67		0.056	0.037
	Silica brick	0.22		0.019	0.015
	Light concrete	0.52		0.074	0.033
	Granite	1.5		0.11	0.17
	Sand	0.26		(<0.015–0.037)	<0.015
Germany	Cement	0.15		0.026	0.015
	Granite	1.3		0.10	0.078
	Brick	0.67		0.063	0.067
	Sand, gravel	<0.26		<0.015	<0.019
	Cement	0.22		0.026	<0.022
	Natural gypsum	<0.074		<0.019	<0.011
	Concrete	0.026		0.022	0.022

[a] Values in parentheses indicate ranges.

example, several houses were found to have been constructed on land that had been used from 1912 to 1915 for the manufacture of radium bromide [43].

A-9. Radionuclide distributions could change dramatically with time. Use of fertilizers, for example, could elevate radionuclide concentrations in soil. Variations in activity distributions could result from the aquisition of materials or equipment. In one reported case, a nuclear facility discovered that sandblasting grit made from coal slag contained ^{226}Ra, ^{228}Ac, ^{228}Th and ^{40}K in concentrations of 0.25, 0.15, 0.22, and 2.10 Bq/g, respectively [44].

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GLOSSARY

The IAEA published a Radioactive Waste Management Glossary as IAEA-TECDOC-264 (1982) and a second edition as IAEA-TECDOC-447 (1988). Over the years, continuing developments in the field of radioactive waste management made it necessary to update or revise individual terms. New terms also needed to be defined or added to the Glossary. The IAEA recently published the third version of the Radioactive Waste Management Glossary, incorporating such updates, revisions and amendments. This Radioactive Waste Management Glossary (1993) serves as a source for the terms included in this Glossary.

analysis, cost–benefit. A systematic economic evaluation of the positive effects (benefits) and negative effects of undertaking an action. Cost–benefit analysis may be used for optimization studies in radiation protection evaluations.

annual limit on intake (ALI). The intake of a given radionuclide in a year by reference man which would result in a committed dose equal to the relevant annual dose limit.

authorization. The granting by a regulatory body of written permission for an operator to perform specified activities. An authorization may be more informal or temporary than a licence.

clearance levels. A set of values, established by the regulatory body in a country or state, expressed in terms of activity concentrations and/or total activities, at or below which sources of radiation can be released from nuclear regulatory control. (See also exemption).

criteria. Conditions on which a decision or judgement can be based. They may be qualitative or quantitative and should result from established principles and standards. In radioactive waste management, criteria and requirements are set by a regulatory body and may result from specific application of a more general principle.

critical group. For a given radiation source and given exposure pathway, a group of members of the public whose exposure is reasonably homogeneous and is typical of individuals receiving the highest dose through the given pathway from the given source.

critical pathway. The dominant environmental route by which members of the critical group are exposed to radiation. For example, the critical pathway for iodine discharged with gaseous effluents is from pasture to cows and then transfer to milk. Consumption of the milk by individuals gives rise to exposure to radiation.

decommissioning. Actions taken at the end of the useful life of a nuclear facility in retiring it from service with adequate regard for the health and safety of workers and members of the public and protection of the environment. The ultimate goal of decommissioning is unrestricted release or use of the site. The time period to achieve this goal may range from a few to several hundred years. Subject to the legal and regulatory requirements of a Member State, a nuclear facility or its remaining parts may also be considered decommissioned if it is incorporated into a new or existing facility, or even if the site in which it is located is still under regulatory or institutional control. This definition does not apply to some nuclear facilities used for mining and milling of radioactive materials (closeout) or for the disposal of radioactive waste (closure).

disposal. The emplacement of waste in an approved, specified facility (e.g. near surface or geological repository) without the intention of retrieval. Disposal may also include the approved direct discharge of effluents (e.g. liquid and gaseous wastes) into the environment with subsequent dispersion.

dose. Absorbed dose, organ dose, equivalent dose, effective dose, committed equivalent dose, or committed effective dose, depending on the context. All these quantities have the dimensions of energy divided by mass. The modifying adjectives are often omitted when they are not necessary for defining the quantity of interest. (See also exposure.) [See also IAEA Safety Series No. 76 and ICRP publications for the numerous technical interpretations of dose to individuals and to populations].

dose, effective. A summation of the tissue equivalent doses, each multiplied by the appropriate tissue weighting factor:

$$E = \sum_T W_T \cdot H_T$$

where H_T is the equivalent dose in tissue T and W_T is the tissue weighting factor for tissue T.

From the definition of equivalent dose, it follows that:

$$E = \sum_T W_T \cdot \sum_R W_R \cdot D_{T,R} = \sum_R W_R \cdot \sum_T W_T \cdot D_{T,R}$$

The unit of effective dose is $J \cdot kg^{-1}$, with the special name sievert (Sv). (See also IAEA Safety Series No. 76 and ICRP publications for the numerous technical interpretations.)

exclusion (from regulatory control). A designation, by the regulatory body in a country or state, of sources of radiation that are not subject to nuclear regulatory control because they are not amenable to control (e.g. cosmic rays and ^{40}K (potassium) in the human body). They are said to be excluded from the regulatory process. (See also exemption.)

exemption or **exempt.** A designation, by the regulatory body in a country or state, for sources of radiation that are not subject to nuclear regulatory control because they present such a low radiological hazard (principles for exemption are presented in IAEA Safety Series No. 89). Under this designation, a distinction can be made between sources which never enter the regulatory control regime (control is not imposed) and sources which are released from regulatory control (control is removed), in both cases because the associated radiological hazards are negligible. The latter is especially pertinent to radioactive waste management, where sources of radiation are released from nuclear regulatory control in accordance with established clearance levels. (See also clearance levels; exclusion.)

exposure. Irradiation of people or materials. Exposure can either be external exposure from sources outside the body or internal exposure from sources inside the body. The exposure can be either normal or potential exposure; occupational, medical or public exposure; and, in intervention situations, temporary, or chronic exposure. (See IAEA Safety Series No. 76 and ICRP Publication 60 for a more technical interpretation of exposure.)

food chain. An expression for depicting the interdependence for food of organisms upon one another in series, beginning with plants or scavenging organisms and ending with the largest carnivores. A web is a network or series of food chains. The series can be represented as compartments in a mathematical model or analysis.

ICRP (International Commission on Radiological Protection). An independent international group of experts, founded in 1928, which provides guidance on principles and criteria in the field of radiation protection. The recommendations of the ICRP are not legally binding, but are generally followed by countries in establishing national regulatory requirements.

incineration. A waste treatment process of burning combustible waste to reduce its volume and yield an ash residue.

intervention. 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.

licence. A formal, legally prescribed document issued to the applicant (i.e. operating organization) by the regulatory body to perform specified activities related to the siting, design, construction, commissioning, operation, decommissioning of a nuclear facility, closure of a disposal facility, closeout of a mining and mill tailings site, or institutional control. (See also authorization.)

practice. 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.

quality assurance. All those planned and systematic actions necessary to provide adequate confidence that an item, process or service will satisfy given requirements for quality, for example, those specified in the licence.

quality control. Action which provides means to control and measure the characteristics of an item, process, facility or person in accordance with quality assurance requirements.

radioactivity. Property of certain nuclides to undergo spontaneous disintegration in which energy is liberated, generally resulting in the formation of new nuclides. The process is accompanied by the emission of one or more types of radiation, such as alpha particles, beta particles and gamma rays.

regulatory body. An authority or a system of authorities designated by the government of a country or State as having legal authority for conducting the licensing process, for issuing licences and thereby for regulating the siting, design, construction, commissioning, operation, closure, closeout, decommissioning and, if required, subsequent institutional control of the nuclear facilities (e.g. near surface repository) or specific aspects thereof. This authority could be a body (existing or to be established) in the field of nuclear related health and safety, mining safety or environmental protection vested and empowered with such legal authority.

risk. The following alternative definitions may be relevant in the field of radioactive waste management:

- In general, risk is the probability or likelihood of a specified event occurring within a specified period or in specified conditions.
- In the safety assessment of radioactive waste repositories, risk may be used as a measure of safety. In this context it is defined as the product of the probability that an individual is exposed to a particular radiation dose and the probability of a health effect arising from that dose.

sealed source. A radioactive source designed in such a form that the probability of dispersion of its radioactive contents is extremely low. Sealed sources may be used, for example, in teletherapy and brachytherapy, and in scientific devices which contain radioactive substances, as well as in a number of medical and industrial applications.

source. Any physical entity that may cause radiation exposure, for example by emitting ionizing radiation or releasing radioactive material.

specific activity. (a) The activity of a radioisotope per unit mass of a material in which the radioisotope occurs. (b) The activity of a radioisotope per unit mass of a material consisting of only that isotope.

waste, radioactive. For legal and regulatory purposes, radioactive waste may be defined as material that contains or is contaminated with radionuclides at concentrations or activities greater than clearance levels as established by the regulatory body, and for which no use is foreseen. (It should be recognized that this definition is purely for regulatory purposes, and that material with activity concentrations equal to or less than clearance levels is radioactive from a physical viewpoint — although the associated radiological hazards are negligible.)

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